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# Nuclear Engineering Handbook

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Edited by  
Kenneth D. Kok



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# *Preface*

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## **Purpose**

The purpose of this handbook is to provide an introduction to nuclear power reactors, the nuclear fuel cycle, and associated analysis tools, to a broad audience including engineers, engineering and science students, their teachers and mentors, science and technology journalists, and interested members of the general public. Nuclear engineering encompasses all the engineering disciplines that are applied in the design, licensing, construction, and operation of nuclear reactors, nuclear power plants, nuclear fuel cycle facilities, and finally the decontamination and decommissioning of these facilities at the end of their useful operating life. This handbook examines many of these aspects in its three sections.

The second edition of this handbook contains some new and updated information including chapters on liquid metal cooled fast reactors, liquid fueled molten salt reactors, and small modular reactors that have been added to the first section on reactors. In the second section, a new chapter on fuel cycles has been added that presents fuel cycle material generally and from specific reactor types. In addition, the material in the remaining chapters has been reviewed and updated as necessary. The material in the third section has also been revised and updated as required with new material in the thermodynamics chapter and economics chapters, and also includes a chapter on the health effects of low level radiation.

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## **Overview**

The nuclear industry in the United States grew out of the Manhattan Project, which was the large science and engineering effort during World War II that led to the development and use of the atomic bomb. Even today, the heritage continues to cast a shadow over the nuclear industry. The goal of the Manhattan Project was the production of very highly enriched uranium and very pure plutonium-239 contaminated with a minimum of other plutonium isotopes. These were the materials used in the production of atomic weapons. Today, excess quantities of these materials are being diluted so that they can be used in nuclear-powered electric generating plants.

Many see the commercial nuclear power station as a hazard to human life and the environment. Part of this is related to the atomic-weapon heritage of the nuclear reactor, and part is related to the reactor accidents that occurred at the Three Mile Island nuclear power station near Harrisburg, Pennsylvania, in 1979, and Chernobyl nuclear power station near Kiev in the Ukraine in 1986. The accident at Chernobyl involved Unit-4, a reactor that was a light water cooled, graphite moderated reactor built without a containment vessel. The accident resulted in 56 deaths that have been directly attributed to it, and the potential for increased cancer deaths from those exposed to the radioactive plume that emanated from the reactor site at the time of the accident. Since the accident, the remaining three reactors at the station have been shut down, the last one in 2000. The accident at Three Mile Island

involved Unit-2, a pressurized water reactor (PWR) built to USNRC license requirements. This accident resulted in the loss of the reactor but no deaths and only a minor release of radioactive material.

In March 2011, a very large earthquake occurred off the coast of Japan that generated a massive tsunami. When the earthquake struck, three of the reactors, Units 1–3, of the Fukushima Daiichi Nuclear Power Plant were operating and Units 4–6 were shut down. The operating units shutdown automatically, and the emergency diesel generators began providing power to the cooling pumps as required. The tsunami swept on shore as a 40 m high wall of water that inundated the emergency power systems knocking them out of operation. With a complete loss of power, the cores of the reactors eventually melted leading to a release of radioactive material both to the air and sea. Cooling was also lost for the spent fuel pools of Units 4–6. When emergency power was restored, sea water was pumped into the reactor systems for cooling purposes. More than 15,000 people were killed by the tsunami, but no deaths were attributed to the failure of the reactors. Five years later, contaminated water is still leaking into the sea, and it will be many years before the site is cleaned and restored.

The commercial nuclear industry began in the 1950s. In 1953, US President Dwight D. Eisenhower addressed the United Nations and gave his famous “Atoms for Peace” speech where he pledged the United States “to find the way by which the miraculous inventiveness of man shall not be dedicated to his death, but consecrated to his life.” President Eisenhower signed the 1954 Atomic Energy Act, which fostered the cooperative development of nuclear energy by the Atomic Energy Commission (AEC) and private industry. This marked the beginning of the nuclear power program in the United States. Earlier on December 20, 1951, 45 kw of electricity was generated at the Experimental Breeder Reactor-I (EBR-I) in Arco, Idaho.

The nuclear reactor in a nuclear power plant is a source of heat used to produce steam that is used to turn the turbine of an electric generator. In that way, it is no different from burning coal or natural gas in a boiler. The difference is that the source of energy does not come from burning a fossil fuel, but from splitting an atom. The atom is a much more concentrated energy source such that a single gram of uranium when split or fissioned will yield 1 MW day or 24,000 kW hours of energy. A gram of coal will yield less than 0.01 kW hours.

Nuclear power plant construction in the United States began in the 1950s. The Shippingport power station in Shippingport, Pennsylvania, was the first to begin operation in the United States. It was followed by a series of demonstration plants of various designs most with electric generating capacity less than 100 MW. During the late 1960s, there was a frenzy to build larger nuclear powered generating stations. By the late 1970s, many of these were in operation or under construction and many more had been ordered. When the accident at Three Mile Island occurred, nuclear power reactor construction activity in the United States essentially ceased and most orders were canceled as well as some reactors that were already under construction.

In 2008, there was a revival in interest in nuclear power. This change was related to the economics of building new nuclear power stations relative to large fossil-fueled plants, and concern over the control of emissions from the latter. Large scale growth of nuclear power is occurring in India and China, but growth in other areas is tempered by slowed economic growth and the availability of natural gas as fuel for generating electricity. However the availability of fossil fuels and their perceived impact on the environment are leading to more interest in nuclear power. This handbook attempts to look at not only the

nuclear power plants, but also the related aspects of the nuclear fuel cycle, waste disposal, and related engineering technologies.

The nuclear industry today is truly international in scope. Major design and manufacturing companies work all over the world. The industry in the United States has survived the 30 years since the Three Mile Island accident, and is resurging to meet the coming requirements for the generation of electric energy. The companies may have new ownership and new names, but some of the people who began their careers in the 1970s are still hard at work and are involved in training the coming generations of workers.

It is important to recognize that when the commercial nuclear industry began, we did not have high-speed digital computers or electronic hand calculators. The engineers worked with vast tables of data and their slide-rules; draftsmen worked at a drawing board with a pencil and ruler. The data were compiled in handbooks and manually researched. The first *Nuclear Engineering Handbook* was published in 1958, and contained that type of information. Today, that information is available on the Internet and in the sophisticated computer programs that are used in the design and engineering process. This handbook is meant to show what exists today, provide a historical prospective, and point the way forward.

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## Organization

The handbook is organized into the following three sections:

- Nuclear Power Reactors
- Nuclear Fuel Cycle Processes and Facilities
- Engineering and Analytical Applications

The first section is devoted to nuclear power reactors. It begins with a historical perspective that looks at the development of many reactor concepts through the research/test reactor stage and the demonstration reactor that was actually a small power station. Today these reactors have faded into history, but some of the concepts are re-emerging in new research and development programs. Sometimes these reactors are referred to as "Generation I." The next chapters in the section deal with the reactors that are currently in operation as well as those that are currently starting through the licensing process, the so-called Generation II and Generation III reactors. This is followed by a discussion of reactor systems that are being proposed to eliminate the high-pressure water cooled systems that require sustained emergency power to shut down. The final chapter in the section introduces the Generation IV reactor concepts. There is no attempt within this section to discuss research and test reactors, military or navel reactors, or space-based reactors and nuclear power systems. There is also no attempt to describe the electric-generating portion of the plant except for the steam conditions passing through the turbines.

Twenty percent of the electrical energy generated in the United States is generated in nuclear power plants. These plants are PWRs and boiling water reactors (BWRs). The Generation II PWRs were manufactured by Westinghouse, Combustion Engineering, and Babcock and Wilcox, whereas the BWRs were manufactured by General Electric. These reactor systems are described in Chapters 2 and 3. The descriptions include the various

reactor systems and components and a general discussion of how they function. The discussion includes the newer systems that are currently being proposed that have significant safety upgrades.

Chapters 4 and 5 describe the CANDU reactor and the high temperature gas cooled reactor (HTGR). The CANDU reactor is the reactor of choice in Canada. This reactor is unique in that it uses heavy water (sometimes called deuterium oxide) as its neutron moderator. Because it uses heavy water as a moderator, the reactor can use natural uranium as a fuel; therefore, the front end of the fuel cycle does not include the uranium enrichment process required for reactors with a light water neutron moderator. The HTGR or gas cooled reactor was used primarily in the United Kingdom. Even though the basic designs of this power generating system have been available since the 1960s, the reactor concept never penetrated the commercial market to a great extent. Looking forward, this concept has many potential applications because the high temperatures can lead to increased efficiency in the basic power generating cycles.

Chapters 6 through 8 give an introductory look at the liquid metal cooled reactor system, the molten salt reactor, and also the small modular reactor systems. Chapter 9 introduces the Gen IV reactor design concepts that have been developed by the United States Department of Energy (USDOE).

The second section is devoted to the nuclear fuel cycle and also facilities processes related to the lifecycle of nuclear systems. The fuel cycle begins with the extraction or mining of uranium ores and follows the material through the various processing steps before it enters the reactor and after it is removed from the reactor core. This section includes nuclear fuel reprocessing, even though it is not currently practiced in the United States, and also describes the decommissioning process that comes at the end of life for nuclear facilities. A separate chapter discusses the fuel cycles that can be used when the reactor fuel is reprocessed.

The first three chapters, Chapters 10 through 12, of this section discuss the mining, enrichment, and fuel fabrication processes. The primary fuel used in reactors is uranium, so there is little mention of thorium as a potential nuclear fuel. The primary enrichment process that was originally used in the United States was gaseous diffusion. This was extremely energy intensive and has given way to the use of gas centrifuges. During fuel fabrication, the enriched gaseous material is converted back to a solid and inserted into the fuel rods that are used in the reactor.

Chapters 13 through 16 discuss the storage of spent fuel, fuel reprocessing, fuel recycle, and waste disposal. Spent fuel is currently stored at the reactor sites where it is stored in spent fuel pools immediately after discharge and can later be moved to dry storage using shielded casks. Fuel reprocessing and fuel recycle are currently not done in the United States, but the chemical separation processes used in other countries are described. Waste disposal of low-level nuclear waste and transuranic nuclear waste are being actively pursued in the United States. The section also includes a discussion of the proposed Yucca Mountain facility for high-level waste and nuclear fuel.

Chapters 17 and 18 describe the transportation of radioactive materials and the processes of decontamination and decommissioning of nuclear facilities.

The third section addresses some of the important engineering analyses critical to the safe operation of nuclear power reactors and also introduces some of the economic considerations involved in the decisions related to nuclear power. These discussions tend to be more technical than those in the first sections.

Chapters 19 and 20 discuss the approaches to safety analysis that are used by the US Nuclear Regulatory Commission (NRC) in licensing nuclear power plants and by the US

Department of Energy (DOE) in the licensing of their facilities. The approach used by the NRC is based on probability and uses probabilistic risk assessment analyses, whereas the DOE approach is more deterministic. Chapters 21 through 23 deal with nuclear criticality, the heat transfer, and thermo-hydraulics and thermodynamic analyses used for nuclear reactors. Criticality is an important concept in nuclear engineering because a nuclear reactor must reach criticality to operate. However, the handling of enriched uranium can lead to accidental criticality, which is an extremely undesirable accident situation. Heat transfer and thermo-hydraulic analyses deal with the removal of heat from the nuclear fission reaction. The heat is the form of energy that converts water to steam to turn the turbine generators that convert the heat to electricity. Controlling the temperature of the reactor core also maintains the stability of the reactor and allows it to function. The thermodynamic cycles introduce the way that engineers can determine how much energy is transferred from the reactor to the turbines.

Chapter 24 introduces the economic analyses that are used to evaluate the costs of producing energy using the nuclear fuel cycle. These analyses provide the basis for decision makers to determine the utility of using nuclear power for electricity generation.

Chapters 25 and 26 discuss radiation protection and the effects of low dose radiation. Persons near or involved in an accidental criticality will receive high radiation exposure that can lead to death. Radiation protection involves the methods of protecting personnel and the environment from excessive radiation exposure. Low dose radiation is discussed to show that the impact of radiation from nuclear power operations is a small fraction of the radiation people receive each day.

**Kenneth D. Kok**



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## *Acknowledgments*

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I also thank my wife, Sharyn Kok, who provided support and encouragement through the process of putting the handbook together. Finally, I want to thank all of my friends and co-workers who encouraged me through this process, with a special thanks to Frank Kreith, who helped make this project possible.



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## *Editor*

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**Kenneth D. Kok** has more than 45 years of experience in the nuclear industry. This includes a wide variety of experience in many areas of nuclear technology and engineering. He served as a senior reactor operator and manager of a research reactor. He planned and managed the decontamination and decommissioning (D&D) of that reactor and has carried out research in neutron radiography, reactor maintainability, fusion reactor systems, advanced nuclear reactor fuel cycles, radioactive material transport systems, and radiation applications. He managed and participated in efforts related to the design and testing of nuclear transport casks, nuclear material safeguards and security, and nuclear systems safety. Kok performed business development efforts related to government and commercial nuclear projects. He performed D&D and organized a successful ASME short course related to D&D of nuclear facilities.

Kok attended Michigan Technological University, where he earned a BS in chemistry, an MS in business administration, and an MS in nuclear engineering. He also did PhD-level course work in nuclear engineering at the Ohio State University. He has more than 25 technical publications and holds two patents. He was a licensed professional engineer before retirement. Kok was elected an ASME fellow in 2003. He presented the Engineer's Week Lecture at the AT&T Allentown Works in 1980. He served as general cochair of the International Meeting of Environmental Remediation and Radioactive Waste Management in Glasgow, Scotland, in 2005, in Liverpool, the United Kingdom, in 2009 and in Brussels, Belgium, in 2013.

Kok is a lifetime member of the ASME, ANS, and the National Defense Industrial Association. He is a past chair of the ASME Nuclear Engineering Division and of the ASME Energy Committee. He was appointed by the American Association of Engineering Societies to serve as the US representative on the World Federation of Engineering Organization's Energy Committee, where he is the vice president for the North American region. He received the ASME 2015 Joseph A. Falcon Energy Award in 2015.



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## *Contributors*

---

**Kurt Colborn**

Waste Control Specialists  
Dallas, Texas

**Kristopher W. Cummings**

Nuclear Energy Institute  
Washington, DC

**Murthy Devarakonda**

Washington TRU Solutions/URS  
Albuquerque, New Mexico

**Brett Dooies**

GE Hitachi Nuclear Energy  
Wilmington, North Carolina

**Chris Ellis**

General Atomics Fission Division  
San Diego, California

**Paul J. Fehrenbach (Retired)**

Atomic Energy of Canada Limited  
Chalk River, Ontario, Canada

**Peter D. Friedman**

Newport News Shipbuilding  
Newport News, Virginia

**Edwin A. Harvego (Retired)**

Idaho National Laboratory  
Idaho Falls, Idaho

**Joseph Heckman**

Energy Solutions  
Oak Ridge, Tennessee

**Nathan (Nate) Hurt (Retired)**

Goodyear Atomic Corporation  
Lake Havasu City, Arizona

**Yehia F. Khalil**

Yale School of Engineering and Applied  
Science  
and  
Yale School of Forestry and Environmental  
Studies  
Yale University  
New Haven, Connecticut

**Stephen W. Kidd**

East Cliff Consulting  
Bournemouth, United Kingdom

**Kenneth D. Kok (Retired)**

Battelle Columbus Division  
URS Corporation  
Richland, Washington DC

**Jay F. Kunze**

College of Science and Engineering  
Idaho State University  
Pocatello, Idaho

**David LeBlanc**

Terrestrial Energy Inc.  
Oakville, Ontario, Canada

**Mark R. Ledoux**

EnergySolutions, LLC  
Salt Lake City, Utah

**Eric Loewen**

GE Hitachi Nuclear Energy  
Wilmington, North Carolina

**Edward S. Lum**

College of Science and Engineering  
Idaho State University  
Pocatello, Idaho

**John Luxat**

Department of Engineering Physics  
McMaster University  
Hamilton, Ontario, Canada

**McLean T. Machut**

AREVA NP Fuel Business Unit  
AREVA Inc.  
Lynchburg, Virginia

**Alistair I. Miller (Retired)**

Atomic Energy of Canada Limited  
Chalk River, Ontario, Canada

**Patricia Paviet**

United States Department of Energy  
Washington, DC

**Ronald E. Pevey**

Department of Nuclear Engineering  
University of Tennessee  
Knoxville, Tennessee

**Maria Pfeffer**

GE Hitachi Nuclear Energy  
Wilmington, North Carolina

**Scott Pfeffer**

GE Hitachi Nuclear Energy  
Wilmington, North Carolina

**Edward G. Price (Retired)**

Atomic Energy of Canada Limited  
Oakville, Ontario, Canada

**Shripad T. Revankar**

School of Nuclear Engineering  
Purdue University  
West Lafayette, Indiana

**Arlen R. Schade (Deceased)**

Bechtel Jacobs LLC  
Oak Ridge, Tennessee

**Richard Schreiber (Retired)**

Westinghouse Electric Co.  
Oak Ridge, Tennessee

**Richard R. Schultz**

Department of Nuclear Science &  
Engineering  
Idaho State University  
Pocatello, Idaho

and

Department of Nuclear Engineering  
Texas A&M University  
College Station, Texas

**Arkal Shenoy (Retired)**

General Atomics Fission Division  
San Diego, California

**Michael F. Simpson**

Department of Metallurgical Engineering  
College of Mines and Earth Sciences  
University of Utah  
Salt Lake City, Utah

**Kevin Theriault**

GE Hitachi Nuclear Energy  
Wilmington, North Carolina

**Brian Triplett**

GE Hitachi Nuclear Energy  
Wilmington, North Carolina

**Cidney B. Voth (Retired)**

United States Department of Energy  
Columbus, Ohio

# Section I

## Introduction

### *Nuclear Power Reactors*

**Kenneth D. Kok**

This section includes a brief early history of the development of nuclear power, primarily in the United States. Individual chapters cover the pressurized water reactor (PWR), boiling water reactor (BWR), and the CANDU Reactor. These three reactor types are used in nuclear power stations in North America, and represent more than 90% of reactors worldwide. Further, this section includes a chapter describing the gas cooled reactor, liquid metal cooled fast reactor, the molten salt reactor, and small modular reactors, and concludes with a discussion of the next generation of reactors, known as “Gen IV.”

The number of reactor concepts that made it past the research and development (R&D) stage to the demonstration stage is amazing. This work was done primarily in the 1950s and early 1960s. Ideas were researched, and small research size reactors were built and operated. They were often followed by demonstration power plants.

Reactor development expanded rapidly during the 1970s. Nuclear power stations were being built all over the United States and in Eastern Canada. On the morning of March 28, 1979, an accident occurred at Three Mile Island Unit 2, Harrisburg, Pennsylvania, that led to a partial core meltdown. All construction on nuclear power plants in the United States halted. There was a significant inflation in the United States economy during this period. The impact of the accident was to increase the need to significantly modify reactors in service as well as those under construction. For the latter, this led to significant cost impacts because of the changes and the inflationary economy. Many reactor orders were canceled and plants already under construction were abandoned or “mothballed.” The public turned against nuclear power as a source of energy to provide electricity.

There has been renewed interest in the construction of new nuclear power stations because of increasing concern over the environmental impact of exhaust fumes from fossil-fueled

power stations and the desire to limit release of these materials. The Watts Bar Unit 2 plant, started in the 1980s by TVA and mothballed at 60% completion stage, has been completed with full power operation expected in 2016. The Watts Bar plant is located on the Tennessee River south of Knoxville, Tennessee. New plants are being ordered in countries around the world. The PWR, BWR, and CANDU chapters in this section address currently operating plants, and the next generation plants being licensed and built today. In the United States, four AP 1000 reactors are under construction in the states of Georgia and North Carolina. The chapter on high temperature gas cooled reactor (HTGR) plants is forward looking and addresses not only electricity generation, but also the production of high-temperature heat for material processing applications. The chapters on the liquid metal cooled fast reactor and the molten salt reactor are steps toward advanced designs that can utilize plutonium from the reprocessing of light water reactor fuel and, in the case of the molten salt reactor, can use thorium. These reactors also operate at low pressure and can be shut down in an emergency and allowed to cool even without emergency cooling systems. The concept of small modular reactors is also introduced. These will allow lower economic investment and also include passive safety systems. Finally, the Generation IV chapter looks at the reactors being investigated as future sources of power for electricity generation. On a historical note, it is interesting to observe that several of the proposed concepts were investigated during the 1950s and 1960s.



# 1

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## *Historical Development of Nuclear Power*

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Kenneth D. Kok

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In the United States, the development of nuclear reactors for nuclear power production began after World War II. Engineers and scientists involved in the development of the atomic bomb could see that the nuclear reactor would provide an excellent source of heat for production of steam that could be used for electricity generation. Work began at Argonne National Laboratory (ANL), Lemont, Illinois, and at Oak Ridge National Laboratory, Tennessee, on various research and demonstration reactor projects.

The director of ANL, Walter Zinn, felt that experimental reactors should be built in a more remote area of the country, so a site was selected in Idaho. This site became known as the National Reactor Testing Station (NRTS) and the Argonne portion was known as ANL-W. The first reactor project at NRTS was the experimental breeder reactor-I (EBR-I). Construction of the reactor began in 1949 and was completed in 1951. On December 20, 1951, a resistance load was connected to the reactor's generator and about 45 kW of electricity generated. This marked the first generation of electricity from a nuclear reactor. The reactor could generate sufficient electricity to supply the power needed for operation of the facility. It is important to note that the first electricity was generated by a sodium-cooled fast-breeder reactor.

In 1953, US President Dwight D. Eisenhower addressed the United Nations and gave his famous "Atoms for Peace" speech where he pledged that the United States would "find the way by which the miraculous inventiveness of man shall not be dedicated to his death, but consecrated to his life." He signed the 1954 Atomic Energy Act, which fostered the cooperative development of nuclear energy by the Atomic Energy Commission and private industry. This marked the beginning of the nuclear power program in the United States.

## 1.1 Early Power and Experimental Reactors

In this section, many types of early reactors will be examined. Many of these were built in the United States as experimental or demonstration projects. Other countries pursued identical and other technologies. Some of these technologies were not developed beyond the experimental stage, but they are now being reconsidered for future use. Many of these reactors are listed in Table 1.1. The primary reference for the information summarized in this section is contained in *Nuclear Power Engineering* by M. M. El-Wakil.

### 1.1.1 BWR Power Plants

Development of the boiling water reactor (BWR) was carried out by the ANL. Following the operation of several experimental reactors in Idaho, the experimental BWR (EBWR) was constructed in Illinois. The EBWR was the first BWR power plant to be built. The plant was initially operated at 5 Megawatts electric (MWe) and 20 Megawatts thermal (MWt). The reactor was operated from 1957 to 1967 at power levels up to 100 MWt.

The first commercial-size BWR was the Dresden Nuclear Power Plant. This plant was owned by the Commonwealth Edison Company and built by the General Electric Company at Dresden, Illinois (about 50 miles southwest of Chicago). The plant was a 200-MWe facility which operated from 1960 to 1978.

The controlled recirculation BWR (CRBWR) was designed by the Allis-Chalmers Manufacturing Company. The reactor was built for the Northern States Power Company and featured an integral steam superheater. The reactor was called the "Pathfinder" and was a 66-MWe and 164-MWt plant. The reactor was built near Sioux Falls, South Dakota, and operated from 1966 to 1967.

Two other BWRs are of interest. The variable moderator boiling reactor was designed by the American Standard Corporation but never built. The second is another plant with an integral superheater built in the USSR. This 100-MWe reactor featured a graphite moderator.

**TABLE 1.1**

Early Reactors in Operation during the Development of Commercial Nuclear Power

Reactor	Type	Date of Operation	Fuel	Coolant	Moderator	Electricity Generation
EBR-1	FBR		U-235/238 metal	Sodium	NA	45 KWe
EBWR	BWR	1957–1967	Enriched uranium metal	Light water	Light water	4.5 MWe
Dresden	BWR	1960–1978	Enriched uranium oxide	Light water	Light water	200 MWe
Pathfinder	CRBR	1966–1967	Enriched uranium oxide	Light water	Light water	66 MWe
Shippingport	PWR	1957–1982	Enriched UO <sub>2</sub>	Light water	Light water	68 MWe
Indian point	PWR	1963–1976	Mixed UO <sub>2</sub> –ThO <sub>2</sub>	Light water	Light water	275 MWe
CVTR	PHWR	1963–1967	Natural UO <sub>2</sub>	Light water	Heavy water	17 MWe
Calder Hall	GCR	1956–2003	Uranium metal	CO <sub>2</sub>	Graphite	50 MWe
THTR	HTGR	1987–1989	Mixed UO <sub>2</sub> –ThO <sub>2</sub>	Helium	Graphite	296 MWe
Piqua	OCR	1963–1966	Enriched uranium metal	Organic liquid	Organic liquid	12 MWe
Hallam	LMGMR	1963–1964	Molybdenum–uranium alloy	Sodium	Graphite	75 MWe
Fermi unit 1	LMFBR	1966–1972	Molybdenum–uranium alloy	Sodium	NA	61 MWe

Note: See text for abbreviations.

### 1.1.2 PWR Power Plants

The first pressurized water reactor (PWR) nuclear power plant built as a central station electrical generating plant was the Shippingport Atomic Power Station near Pittsburgh, Pennsylvania. The reactor was designed and built by the Westinghouse Electric Company and operated by the Duquesne Light Company. The plant produced 68 MWe and 231 MWt. It began operation late in 1957 and operated until 1982. During its lifetime, it operated as a PWR and a light water breeder reactor (LWBR), where it had a core designed with a thorium blanket to breed  $^{233}\text{U}$  as a potential reactor fuel. The Shippingport reactor was based on the reactor system used for naval propulsion.

A second PWR was designed and built at Buchanan, New York, for the Consolidated Edison Company. The reactor was designed by the Babcock & Wilcox Company and had the unique feature of an oil- or coal-fired superheater. The plant was a 275-MWe and 585-MWt plant. The plant used fuel that was a mixture of uranium and thorium oxide.

The pressurized heavy water-moderated reactor is also included in this category. This plant can use natural uranium as fuel. One early plant of this type was built and operated in Parr, South Carolina. It operated at 17 MWe from 1963 to 1967. This is the type of reactor used in Canada.

A final early concept for a PWR was a pebble-bed system. This concept, developed by the Martin Company, was known as the liquid fluidized bed reactor (LFBR). The concept was never realized.

### 1.1.3 Gas-Cooled Reactor Power Plants

Early gas-cooled reactor (GCR) power plants were developed in the United Kingdom. The first ones were cooled with  $\text{CO}_2$  and were known as the Calder Hall type. They used natural uranium metal fuel and were moderated with graphite. The first one began operation in 1956 and was closed in 2003. It was located in Seaside, Cumbria, and generated 50 MWe. Later versions were up to five times larger. Gas-cooled power plants were also built in France, Germany, and other European countries.

A second type of GCR used the pebble bed concept with helium as a coolant. The uranium and thorium fuel was imbedded in graphite spheres and cooled with helium. The high temperature thorium fueled reactor (THTR) operated between 1985 and 1989 in Germany. It produced 760 MWt and 307 MWe. The thorium in the fuel pellets was used to breed  $^{233}\text{U}$ .

Two GCR power plants have been operated in the United States. The first was Peach Bottom Unit 1, which provided 40 MWe. The second was the Fort St. Vrain reactor, which provided 330 MWe.

### 1.1.4 Organic Cooled and Moderated Reactors

The first organic cooled and moderated reactor was an experimental reactor (MORE). It was constructed and operated at the NRTS in Idaho. It was followed by the Piqua OMR Power Plant in Piqua, Ohio. It was a 12-MWe and 45-MWt plant. The reactor included an integral superheater. The plant operated from 1963 to 1966.

### 1.1.5 Liquid Metal-Cooled Reactors

Liquid metal has been used to cool thermal and fast reactors. Sodium-cooled graphite reactors are examples of thermal reactors. The sodium-cooled reactor experiment was

built by Atomics International. Even though it was a small reactor (20 MWt), a steam generator turbine system was added to this reactor and it generated electricity for Southern California Edison Company beginning in July 1957. The Hallam Nuclear Power Facility (HNPF) was subsequently constructed for the consumers of Public Power District near Lincoln, Nebraska. The plant was a 76-MWe and 254-MWt graphite-moderated sodium-cooled reactor system. The plant operated from 1963 to 1964.

The more familiar sodium-cooled reactor is the liquid metal-cooled fast-breeder reactor (LMFBR). The Enrico Fermi nuclear power plant was built in Lagoona Beach, Michigan, in 1966. The reactor operated at 61 MWe until 1972. Reactors of this type have the advantage of operating at relatively low pressure.

### 1.1.6 Fluid-Fueled Reactors

Several fluid-fueled reactors have been built and operated as experiments. The concept is that fuel is contained within the coolant. Systems of this type include aqueous fuel systems, liquid metal-fueled systems, molten salt systems, and gaseous suspension systems. The homogeneous reactor experiment was constructed and operated at Oak Ridge National Laboratory, as was the Molten-Salt Reactor experiment. A liquid metal fuel reactor experiment was operated at Brookhaven National Laboratory. Power reactors of this type have not been built.

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## 1.2 Current Power Reactor Technologies

The major development of nuclear power began in the late 1960s. Power plants rapidly increased in size from a generating capacity of tens of MWe to more than 1000 MWe. Building and operation took place all over the world. Today, nuclear power plants are operating in 33 countries. The data provided in this section have been extracted from the "World List of Nuclear Power Plants" provided by the American Nuclear Society in the March 2015 edition of *Nuclear News*.

The development of nuclear power was in full swing in the 1970s when the accident occurred at the Three Mile Island Unit 2 nuclear power plant near Harrisburg, Pennsylvania, in 1979. The reactor was a PWR supplied by Babcock & Wilcox Corporation. As a result of this accident, reactor construction came to a standstill as the cause of the accident was analyzed, and the design of reactors under construction was modified to meet new licensing requirements. Costs increased dramatically and many orders for reactors were canceled. The impact of this accident was felt primarily in the United States.

In 1986, an accident occurred at the Chernobyl Unit 4 reactor near Kiev in Ukraine. The Chernobyl reactor was a light water-cooled graphite-moderated (LWG) reactor. This accident led to the release of a large amount of airborne radioactivity and the death of many of the responders. As a result of this accident, several countries with smaller nuclear power programs ceased the pursuit of nuclear power electricity generation.

In 2010 the rebirth of the nuclear power industry seemed to be taking off. New power reactors were being discussed and ordered in many countries around the world. However, on March 11, 2011, a major magnitude 9.0 earthquake struck off the coast of Japan followed

less than an hour later by a 15 m tall tsunami wave. Of the six nuclear power reactors at the Fukushima Daiichi, units 1, 2, and 3 were operating as were seven other nuclear power plants in the area. All of the 11 operating nuclear power plants in the region shut down automatically when the quake hit, and subsequent investigation showed no significant damage to any of the plants due to the quake. The problem at Fukushima Daiichi was caused by the loss of all off-site and on emergency power due to the tsunami which disabled the on-site emergency generators which were required to provide cooling water circulation for about 4 days. Units 1, 2, and 3 suffered core meltdowns, and in addition unit 4 lost heat removal capability for the spent fuel pool.

This event had a sobering impact on the nuclear power industry around the world. Many nuclear power programs were put on hold, and in some countries it was determined that they would abandon the nuclear power option.

At the end of 2014, there were 435 individual nuclear power reactors operating throughout the world. In some cases, there are multiple reactors in a single power station, so the number of power stations will be less than the number of reactors. Table 1.2 presents the number of reactors in operation and the total number of reactors, including those at some stage of construction. The MWe presented in Table 1.2 is the design net-generating

**TABLE 1.2**

Nuclear Power Plant Units by Nation

Nation	# PWR <sup>a</sup>	MWe	# BWR <sup>b</sup>	MWe	# Other <sup>c</sup>	MWe	Planned <sup>d</sup>	MWe	Total Units	Total MWe
Argentina					2	935	2	717	4	1,652
Armenia	1	375							1	375
Bangladesh							2	2,000	2	2,000
Belarus							2	2,400	2	2,400
Belgium	7	5,885							7	5,885
Brazil	2	1,901					1	1,275	3	3,176
Bulgaria	2	1,906							2	1,906
Canada					19	13,472			19	13,472
China	22	18,658					37	34,240	37	52,898
China (Taiwan)	2	1,780	4	3,104			2	2,600	8	7,484
Czech Republic	6	3,678							6	3,678
Finland	2	976	2	1,740			2	2,800	6	5,516
France	58	63,130					1	1,600	59	64,730
Germany	7	9,486	2	2,572					9	12,058
Hungary	4	1,889					2	2,400	6	4,289
India	1	917	2	300	18	4,091	8	5,257	29	10,565
Iran	1	915							1	915
Japan	24	19,286	24	22,991			3	3,002	51	45,279
Mexico			2	1,300					2	1,300
Netherlands	1	487							1	487
Pakistan	2	600			1	125	5	2,600	8	3,325
Romania					2	1,300	2	1,240	4	2,540
Russia	17	12,864			16	10,779	12	9,710	45	33,353

*(Continued)*

TABLE 1.2 (Continued)

## Nuclear Power Plant Units by Nation

Nation	# PWR <sup>a</sup>	MWe	# BWR <sup>b</sup>	MWe	# Other <sup>c</sup>	MWe	Planned <sup>d</sup>	MWe	Total Units	Total MWe
Slovakia	4	1,816					2	810	6	2,626
Slovenia	1	666							1	666
South Africa	2	1,800							2	1,800
South Korea	19	17,975			4	2,722	7	9,400	30	30,097
Spain	6	6,004	1	1,064					7	7,068
Sweden	3	2,799	7	6,504					10	9,303
Switzerland	3	1,700	2	1,538					5	3,238
Turkey							4	4,600	4	4,600
Ukraine	15	13,107					3	2,850	18	15,957
United Arab Emirates							4	5,600	4	5,600
United Kingdom	1	1,188			15	8,025			16	9,213
United States	65	66,161	34	34,896			8	9,490	107	110,547
Total	278	257,949	80	76,009	77	41,449	109	108,191	844	479,998

Source: American Nuclear Society, 17th Annual Reference Issue, *Nuclear News*, March 2015.

<sup>a</sup> Total operating PWRs.

<sup>b</sup> Total operating BWRs.

<sup>c</sup> Includes gas-cooled, heavy water, graphite-moderated light water, and liquid metal-cooled fast-breeder reactors.

<sup>d</sup> Includes reactors of all types planned or under construction.

capability of the plants. The electricity generated is dependent on the number of full power hours generated by the plants.

More than one-half of the nuclear reactors in the world are PWRs. The distribution of current reactors by type is listed in Table 1.3. There are six types of reactors currently used for electricity generation throughout the world (Table 1.3).

TABLE 1.3

## Nuclear Power Units by Reactor Type (Worldwide)

Reactor Type	Main Countries	Operational		Planned/Under Construction		Total		Fuel
		# Units	GWe	# Units	GWe	# Units	GWe	
Pressurized light-water reactors (PWR)	The United States, France, Japan, and Russia	276	257	88	93	354	350	Enriched UO <sub>2</sub>
Boiling light-water reactors (BWR and AWBR)	The United States, Japan, and Sweden	80	76	6	8	86	84	Enriched UO <sub>2</sub>
Gas-cooled reactors (Magnox and AGR)	The United Kingdom	15	8	1	.2	16	8	Natural U (metal), enriched UO <sub>2</sub>

(Continued)

**TABLE 1.3 (Continued)**

Nuclear Power Units by Reactor Type (Worldwide)

Reactor Type	Main Countries	# Units	GWe	# Units	GWe	# Units	GWe	Fuel
		Operational		Planned/Under Construction		Total		
Pressurized heavy-water reactors—CANDU (PHWR)	Canada	48	24	9	6	57	30	Natural UO <sub>2</sub>
Graphite-moderated light-water reactors (RMBK)	Russia	15	10	0	0	15	10	Enriched UO <sub>2</sub>
Liquid-metal-cooled fast-breeder reactors (LMFBR)	France and Russia	1	1	5	2	6	2	PuO <sub>2</sub> and UO <sub>2</sub>
		435	375	109	109	544	484	

## Reference

American Nuclear Society, 17th Annual Reference Issue, *Nuclear News*, March 2015.



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# 2

## *Pressurized Water Reactors*

Richard Schreiber

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## 2.1 Introduction

In the 1960s, the US Government, as well as other countries, promoted the development and application of nuclear energy for the production of electric power. The employment of nuclear navies throughout the world provided a knowledge base for the type of reactor using high-pressure “light” water as coolant and moderator. The fuel selected for domestic power stations was uranium dioxide in pellet form, slightly enriched in the isotope U-235, and protected from the coolant by stainless steel or a modified zirconium–tin alloy that came to be known as “Zircaloy.” Zircaloy-4 has been the tubular cladding material of choice today because of its corrosion resistance when pre-oxidized and its low absorptive “cross section” for neutrons. In the present century, pressurized water reactors (PWRs) are the most popular design, providing nearly two-thirds of the installed nuclear capacity throughout the world.

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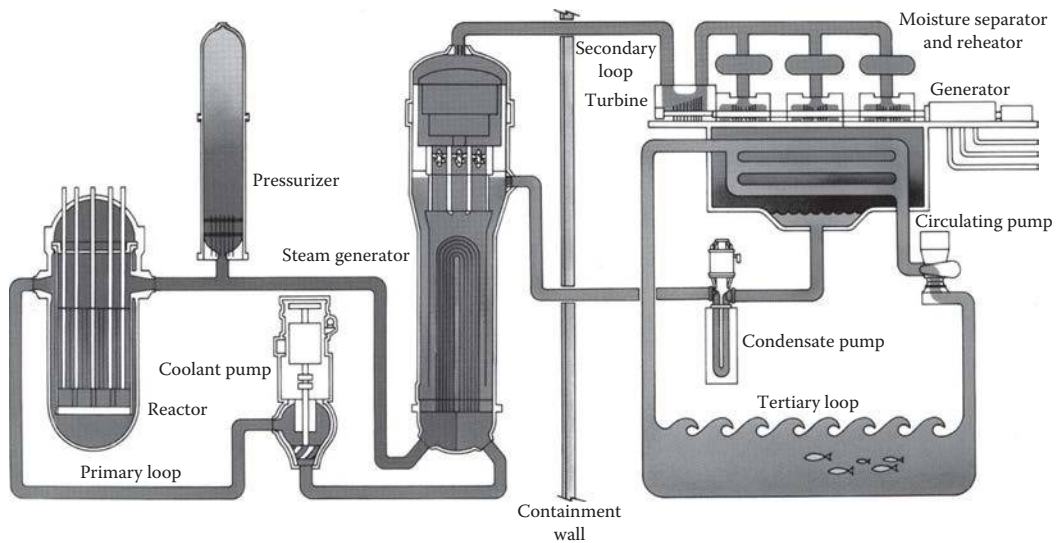
## 2.2 Overview

For general discussion purposes, a nuclear power plant can be considered to be made up of two major areas: a nuclear “island” and a turbine island composed of a turbine/generator (T-G). Only the former is being described in detail in this chapter. To a large extent, the design of the non-nuclear portion of a Rankine cycle power plant depends only on the steam conditions of temperature, pressure, steam “quality” (how little liquid is present with the vapor), and flow arriving at the turbine, regardless of the heat source. There are safety systems in the non-nuclear part of a nuclear plant that are unique, such as a diesel generator for emergency power. All essential nuclear systems are discussed below.

---

## 2.3 The Power Plant

For PWRs, the part of the coolant system (primary loop, Figure 2.1) that contains radioactivity is surrounded by a sturdy containment structure whose main purpose is to protect operating personnel and the public. Various auxiliary and safety systems attached to the primary are also located within the containment. This protected array of equipment we call the nuclear island is also called the “nuclear steam supply system” (NSSS). The NSSS and the balance-of-plant (including the T-G and all other systems) are composed of fluid, electrical, instrumentation, and control systems; electrical and mechanical components; and the buildings or structures housing them. There are also several shared fluid, electrical, instrumentation, and control systems, as well as other areas of interconnection or interface. The principal operating data for current Westinghouse NSSS models are listed in Table 2.1.



**FIGURE 2.1**  
Nuclear steam supply system (schematic).

**TABLE 2.1**  
Principal Data for Current Westinghouse NSSS Models

Model	212	312	412	414
<b>Number of Loops</b>	<b>2</b>	<b>3</b>	<b>4</b>	<b>4</b>
NSSS power (megawatt thermal)	1882	2785	3425	3819
Approximate electrical output (MWe)	600	900	1150	1280
Steam pressure, psia (bar)	920 (63)	960 (66)	1000 (69)	1100 (76)
Reactor vessel ID, in. (cm)	132 (335.3)	157 (398.8)	173 (439.4)	173 (439.4)
Steam generator model	F	F	F	H
Reactor coolant pump type	93A1	93A1	93A1	93A1
RCP motor horsepower	7000	7000	7000	9000
Hot leg ID, in. (cm)	29 (73.7)	29 (73.7)	29 (73.7)	29 (73.7)
Cold leg ID, in. (cm)	27.5 (69.9)	27.5 (69.9)	27.5 (69.9)	27.5 (69.9)
Number of fuel assemblies	121	157	193	193
Fuel length, ft (cm)	12 (365.8)	12 (365.8)	12 (365.8)	14 (426.7)
Fuel assembly array	16 × 16	17 × 17	17 × 17	17 × 17

## 2.4 Vendors

In the United States, the principal suppliers of the present generation of NSSS were units of Babcock & Wilcox (B&W), Combustion Engineering (C-E), General Electric (boiling water reactors [BWRs]) and Westinghouse. These and several other organizations supply the fuel assemblies. Other consortiums have been formed throughout the world. In Europe, a group named AREVA has been organized. Since March 1, 2006, all first-tier subsidiaries

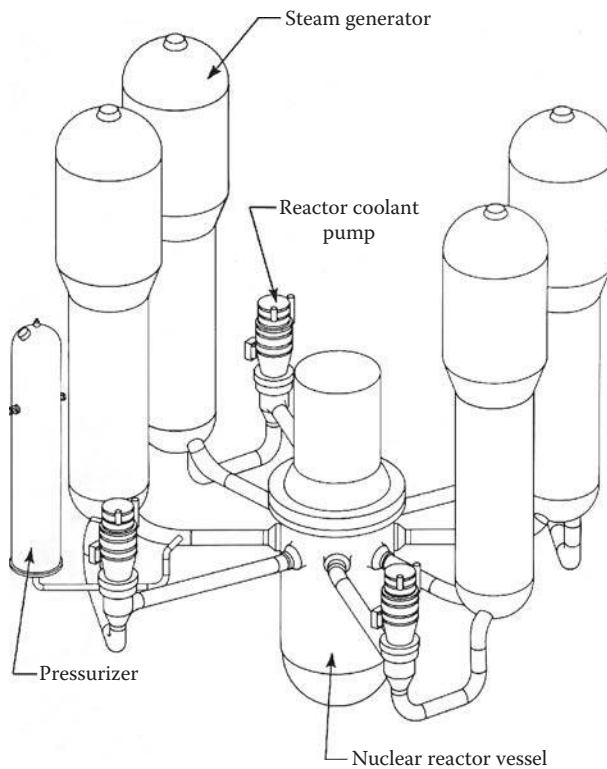
of the AREVA group have new names. The trade name of COGEMA is now AREVA NC, Framatome ANP is now AREVA NP, and Technicatome is AREVA TA. This initiative also applies to second-tier subsidiaries and sites in France or abroad where “COGEMA” or “Framatome ANP” is part of the name. Japanese suppliers include Mitsubishi Heavy Industries (MHI) for PWRs, as well as local and international manufacturers for reactor equipment and fuel. In South Korea, PWR vessel and equipment suppliers include Doosan Heavy Industries/Construction and Korea Power Engineering. Fuel suppliers are Korea Nuclear Fuel and international suppliers. In Germany, Siemens is the major player, but they also have absorbed Exxon Nuclear in the United States by way of Kraftwerk Union (Germany). Siemens has also turned over their nuclear assets to a joint venture with Framatome ANP of France. The new company is to be called AREVA NP. Many other companies and consortia worldwide supply the nuclear power industry. MHI has aligned with AREVA to form a joint venture ATMEA to build nuclear plants, but MHI has also joined with Westinghouse on some bids and as a sole bidder in others. AREVA has absorbed the former B&W nuclear unit in Lynchburg, Virginia.

In the 1960s, C-E began selling commercial nuclear power steam supply systems, having cut their teeth on naval systems, just as many other firms had done. C-E was generally credited with a superior design to its competitors, evidenced by the fact that the megawatt yield of its nuclear reactors was typically about 10% higher than that of comparable PWRs. The basis for this increase in efficiency was a computer-based system called the core operating limit supervisory system (COLSS), which leveraged almost 300 in-core neutron detectors and a patented algorithm to allow higher power densities. In 1990, C-E became a subsidiary of ASEA Brown Boveri (ABB), a Swiss–Swedish firm based in Zurich. In late December 1999, the British firm British Nuclear Fuels Limited (BNFL) agreed to purchase ABB’s worldwide nuclear businesses, including the nuclear facilities of C-E. In March 1999, BNFL had acquired the nuclear power businesses of Westinghouse Electric Company with the remaining parts of Westinghouse going to Morrison Knudson (MK) Corporation. In late 2006, Toshiba completed its acquisition of those nuclear units from BNFL, bringing C-E and Westinghouse design and manufacturing capabilities together. Westinghouse has also developed the ability to design and build BWRs and fuel. These rearrangements have taken place in the past 20 years while nuclear power dropped from the headlines. Expansion and development of new designs continues in the twenty-first century.

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## 2.5 General Description of PWR Nuclear Power Plants Presently in Use

The central component of the reactor coolant system (RCS) is a heavy-walled reactor vessel that houses the nuclear core and its mechanical control rods, as well as necessary support and alignment structures. It is shown schematically in Figure 2.1, in relation to other parts of the system in Figure 2.2, and as a cut-away showing the internal details in Figure 2.3. The vessel is cylindrical in shape with a hemispherical bottom head and a flanged and gasketed upper head for access. It is fabricated of carbon steel, but all wetted surfaces are clad with stainless steel to limit corrosion. The internal core support and alignment structures are removable to facilitate inspection and maintenance, as is the alignment structure for the top-mounted control rod drive mechanisms. Vessel inlet and outlet nozzles for the primary loops are located at a level well above the top of the fuel core.



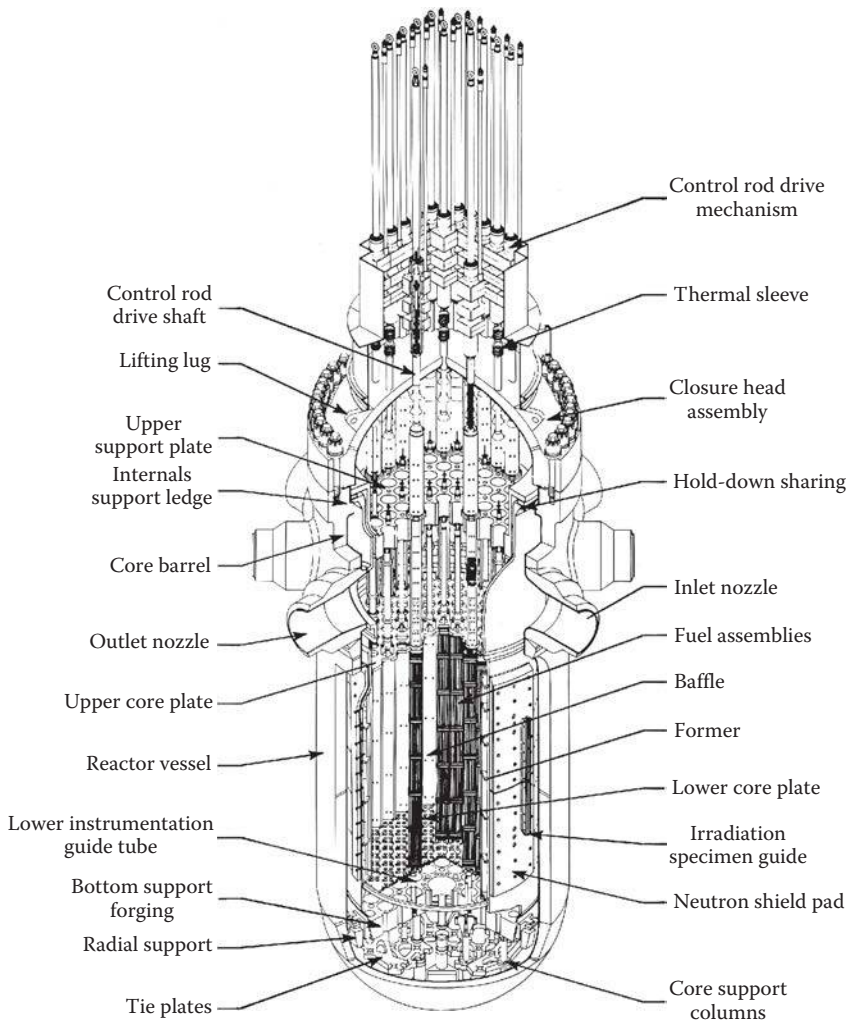
**FIGURE 2.2**  
Layout of nuclear island.

### 2.5.1 Fuel

The nuclear core comprises several fuel assemblies arranged in three regions to optimize fuel performance. All fuel assemblies are mechanically identical, but enrichment of the uranium dioxide fuel differs from assembly to assembly. In a typical initial core loading, three fuel enrichments are used. Fuel assemblies with the highest enrichments are placed in the core periphery, or outer region, and the groups of lower enrichment fuel assemblies are arranged in a selected pattern in the central region. In subsequent refuelings, one-third of the fuel (the highest “burnup”) is discharged and fresh fuel is loaded into the outer region of the core. The remaining fuel is rearranged in the central two-thirds of the core as to achieve optimal power distribution and fuel utilization. Figure 2.4 shows the details of the PWR fuel assembly. Figure 2.5 shows how they are distributed by enrichment within the core. Table 2.2 gives fuel rod design details. Further details regarding nuclear fuel are given in Chapter 12 this handbook.

### 2.5.2 Control

Rod cluster control (RCC) assemblies used for reactor control consist of absorber rods attached to a spider connector which, in turn, is connected to a drive shaft. The absorber (control) rods are loaded with a material that has a high affinity “cross section” for neutrons. Above the core, control rods move within guide tubes that maintain alignment of the control rods with empty thimbles of certain fuel assemblies at particular locations in

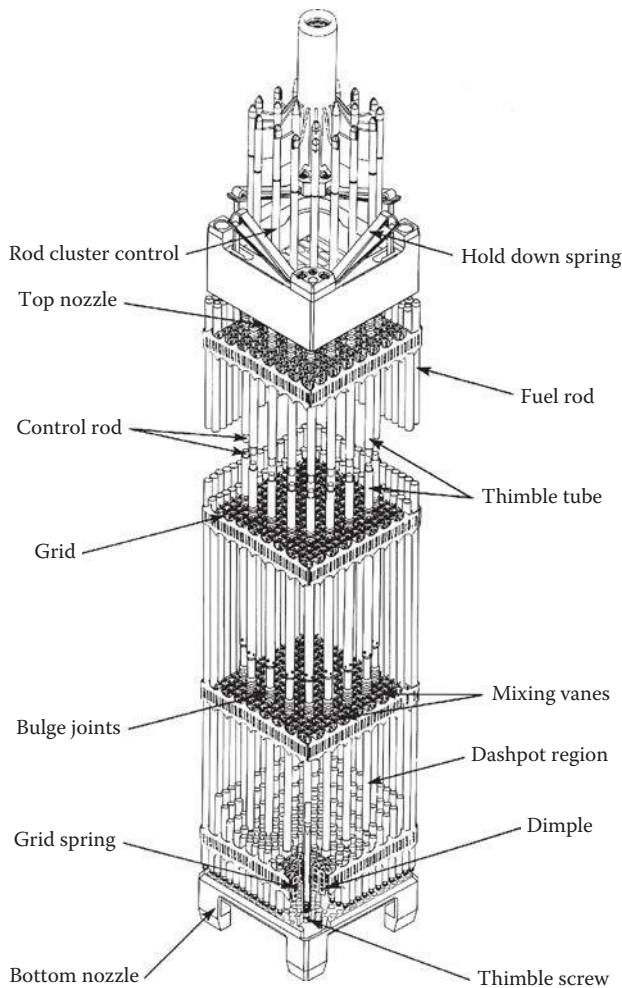


**FIGURE 2.3**  
Cut-away of reactor vessel.

the core. RCC assemblies are raised and lowered by a drive mechanism on the reactor vessel head. The drive mechanism allows the RCC assemblies to be released instantly, “trip,” when necessary for rapid reactor shutdown. Insertion of the assemblies during a trip is by gravity. Figure 2.6 shows the relationship of the fuel assembly and the RCC arrangement within the core. The intent is to equalize (“flatten”) the power distribution across the core as much as possible.

### 2.5.3 Burnable Poison

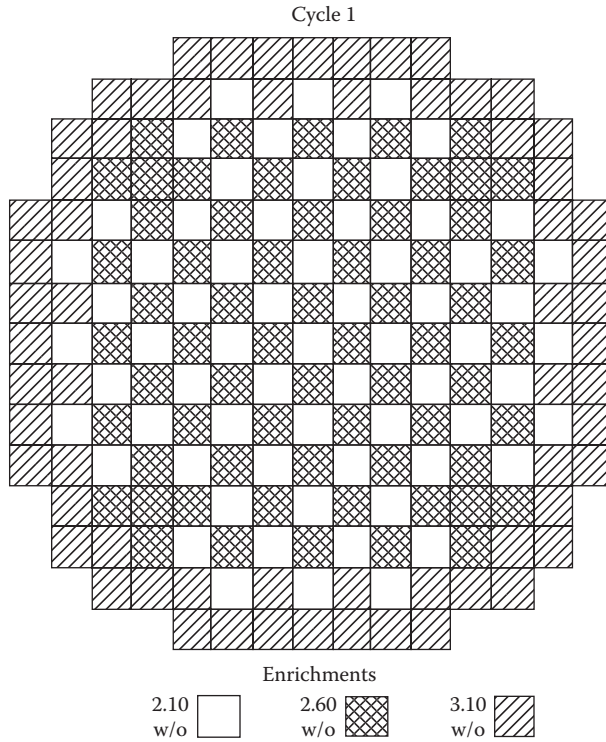
In addition to control rods, there is a distribution of absorber burnable poison (BP) rods that are mounted on RCC-like fixtures but are not connected to drive mechanisms. The BP rods remain in the core during operation but may be moved to new locations during



**FIGURE 2.4**  
Typical fuel assembly for the present generation of reactors.

shutdown. Figure 2.7 shows their distribution in a typical large core. Their intent is to suppress the large excess of nuclear reactivity during the early part of the cycle, using up the absorber during operation. They also allow a lower concentration of soluble boron poison during operation. There is a small burn-up penalty (Figure 2.8). The configuration of each BP assembly is similar in appearance to an RCC assembly with the exception of the handling fitting. Positions in the cluster not occupied by BP rods contain loose-fitting plugs that balance the coolant flow across the host fuel assembly. The plugs are also connected to the fixture. The fuel assemblies that contain neither control rods (including safety rods) nor BPs, nor neutron startup sources, contain "pluggers." Pluggers are all flow-balancing plugs mounted on a fixture for support and handling. Special handling tools are needed for each of these inserts into a fuel assembly because they all become "hot" in use but must be switched between assemblies. The long dangling rods are kept from splaying by the use of "combs" that keep them properly oriented for reinsertion. All of these manipulations are done deep underwater.





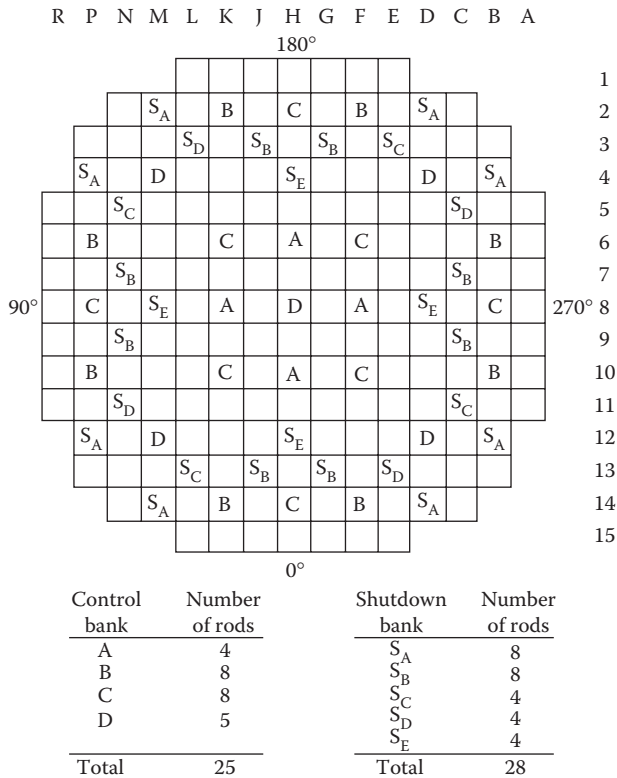
**FIGURE 2.5**  
Pattern of initial fuel load, three regions.

**TABLE 2.2**  
Fuel Rod Parameters (Four-Loop Plant)

Fuel rod length	12 ft. (365.8 cm)
Outside diameter	0.360 in. (0.914 cm)
Cladding thickness	0.0225 in. (0.0572 cm)
Cladding material	Zircaloy-4
Diametral gap	0.0062 in. (0.0157 cm)
Pellet diameter	0.3088 in. (0.7844 cm)
Lattice pitch	0.496 in. (1.260 cm)
Rods array in assembly	17 × 17
Rods in assembly	264
Total number of fuel rods in core	50,952

### 2.5.4 Coolant Pumps

Reactor coolant pumps (RCPs) (Figure 2.9) are vertical, single-stage, mixed flow pumps of the shaft-seal type. A heavy flywheel on the pump motor shaft provides long coast down times to preclude rapid decreases in core cooling flow during pump trips. Interlocks and automatic reactor trips ensure that forced cooling water flow is present whenever the reactor is at power. Additionally, two separate power supplies are available to the pump motor when the plant is at power.



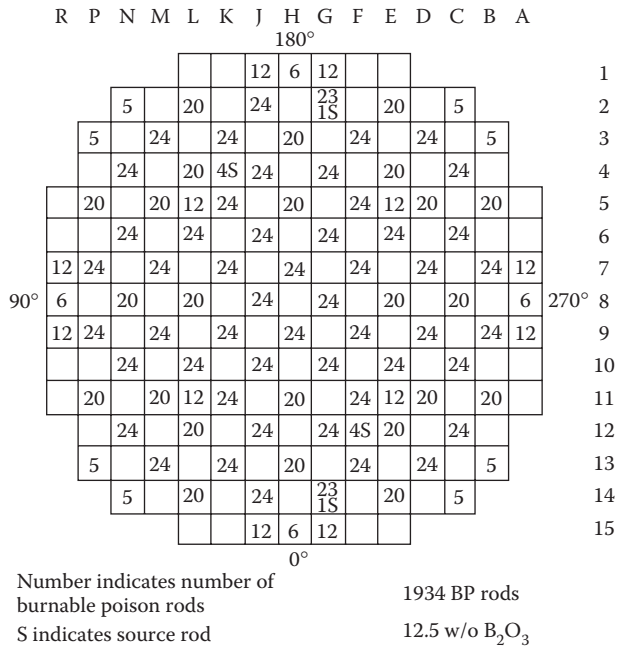
**FIGURE 2.6**  
Arrangement of control rod banks in the reactor core.

**2.5.5 Steam Generation**

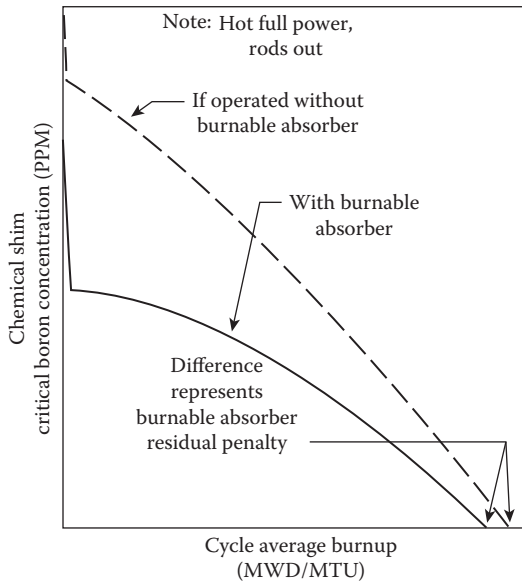
Steam generators are of a vertical U-tube design with an expanded upper section that houses integral moisture separation equipment to produce steam with a quality of at least 99.75% (Figure 2.10). Table 2.3 lists many design parameters. Preheated feedwater enters the top of the unit, mixes with effluent from the moisture separators, and then flows downward on the outside of the tube bundle. The feed is distributed across the bundle and then flows upward alongside the heated tubes. An alternate design used by another vendor (B&W) has a bundle of straight tubes. Water in the secondary loop is boiled in the lower section of the steam generator, dried to all steam in the middle section and superheated in the upper section, obviating the need for moisture separators before passing the dry steam to the turbines. Reactor coolant piping, the reactor internals, and all of the pressure-containing and heat transfer surfaces in contact with reactor water are stainless steel or stainless steel clad, except the steam generator tubes and fuel tubes, which are Inconel and Zircaloy, respectively.

**2.5.6 Pressurizer**

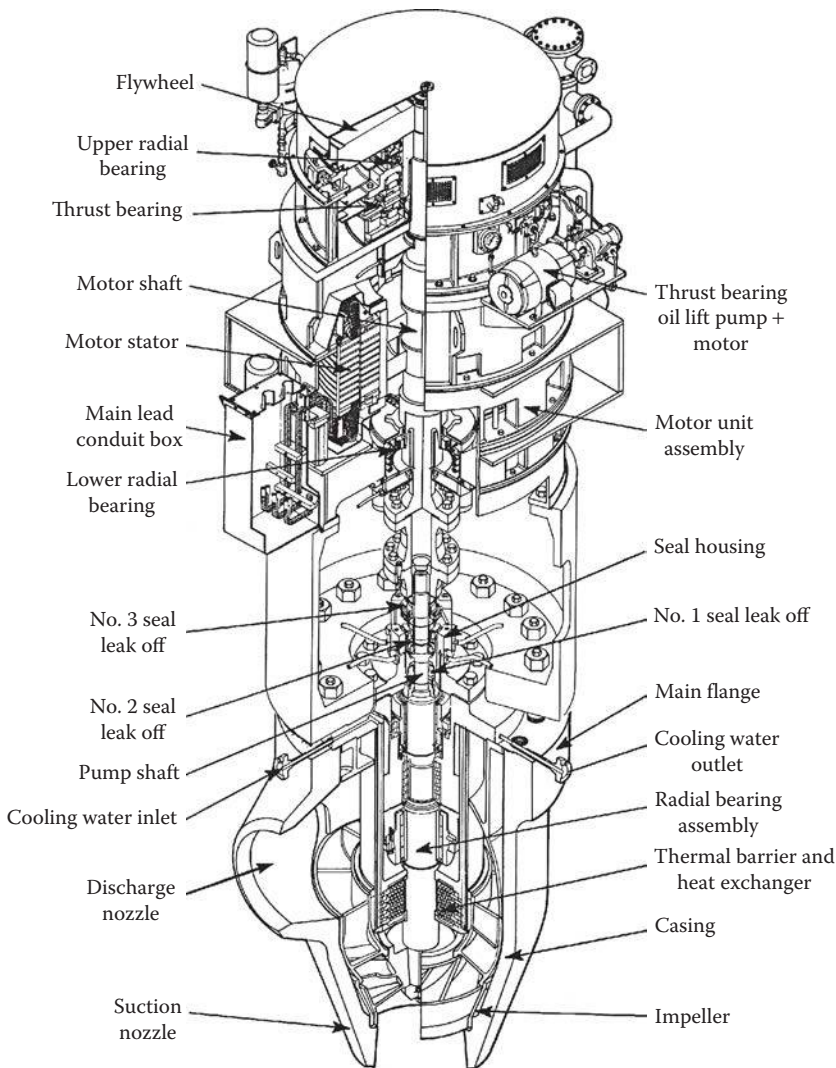
An electrically heated pressurizer connected to one of the reactor coolant hot legs maintains RCS pressure during normal operation, limits pressure variations during plant load transients, and keeps system pressure within design limits during abnormal conditions. A typical design is given in cut-away pictorial in Figure 2.11. For example, a transient that could decrease system pressure is counteracted by flashing water within the pressurizer which is kept at saturation



**FIGURE 2.7**  
 Arrangement of burnable poison rods, initial core loading.



**FIGURE 2.8**  
 Effect of burnable poison rods on soluble poison requirements.

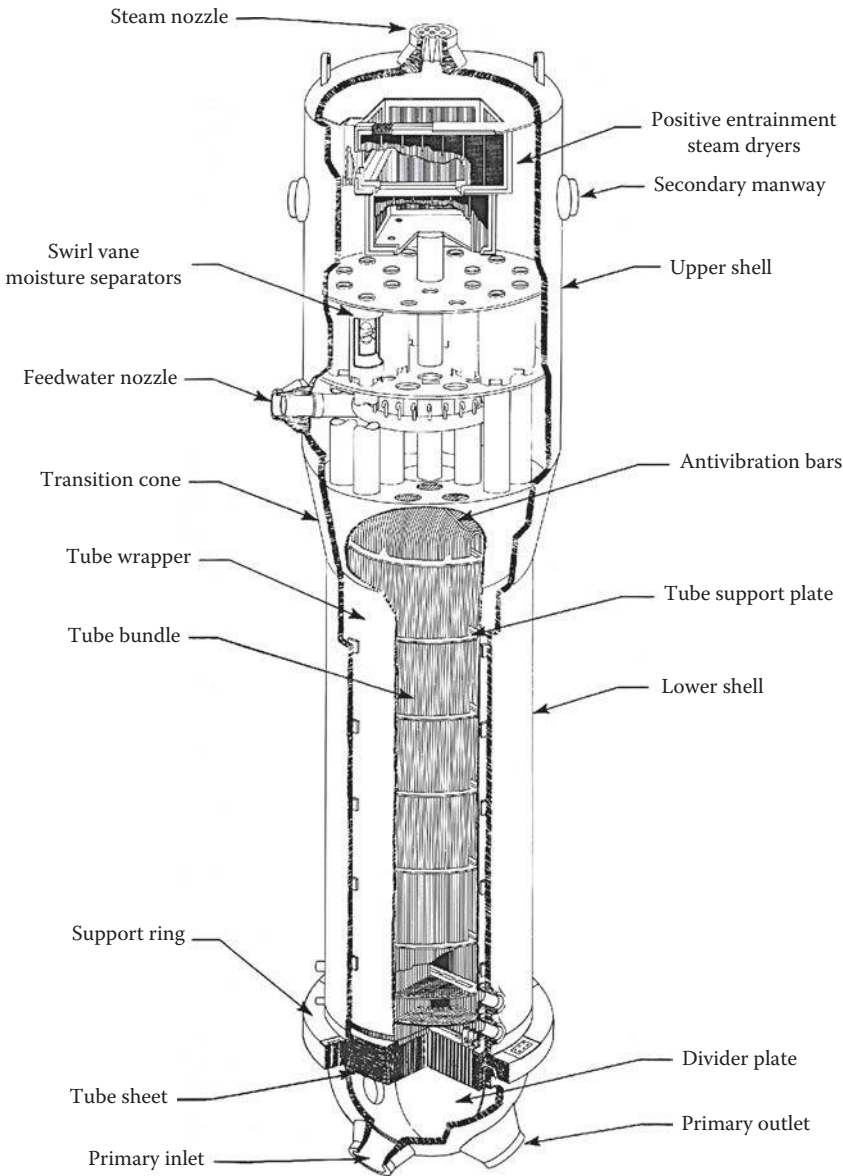


**FIGURE 2.9**  
Cut-away of reactor coolant pump.

temperature by the automatic heaters. An increasing pressure transient is limited by spraying cooler water from the primary loop into the pressurizer steam space to partially collapse the steam bubble, or by automatic operation of relief and safety valves.

## 2.6 Operations

Normal and emergency operation of the RCS requires several support functions to maintain water inventory, purify and treat primary coolant, remove residual heat after a plant shutdown, provide cooling water to pumps and motors, supply ventilation air, and provide



**FIGURE 2.10**  
Cut-away of steam generator.

emergency supplies of core cooling water. These functions are provided by auxiliary systems described later in this section.

Transient power behavior of a nuclear core is determined by a condition known as “reactivity.” For a core operating at a steady power level, the various factors that affect reactivity are balanced so that the net reactivity is zero. If the net reactivity is positive, power level will increase and, conversely, decrease if reactivity is negative. Power control of a PWR is based on balancing reactivity through the use of mechanical and

**TABLE 2.3**

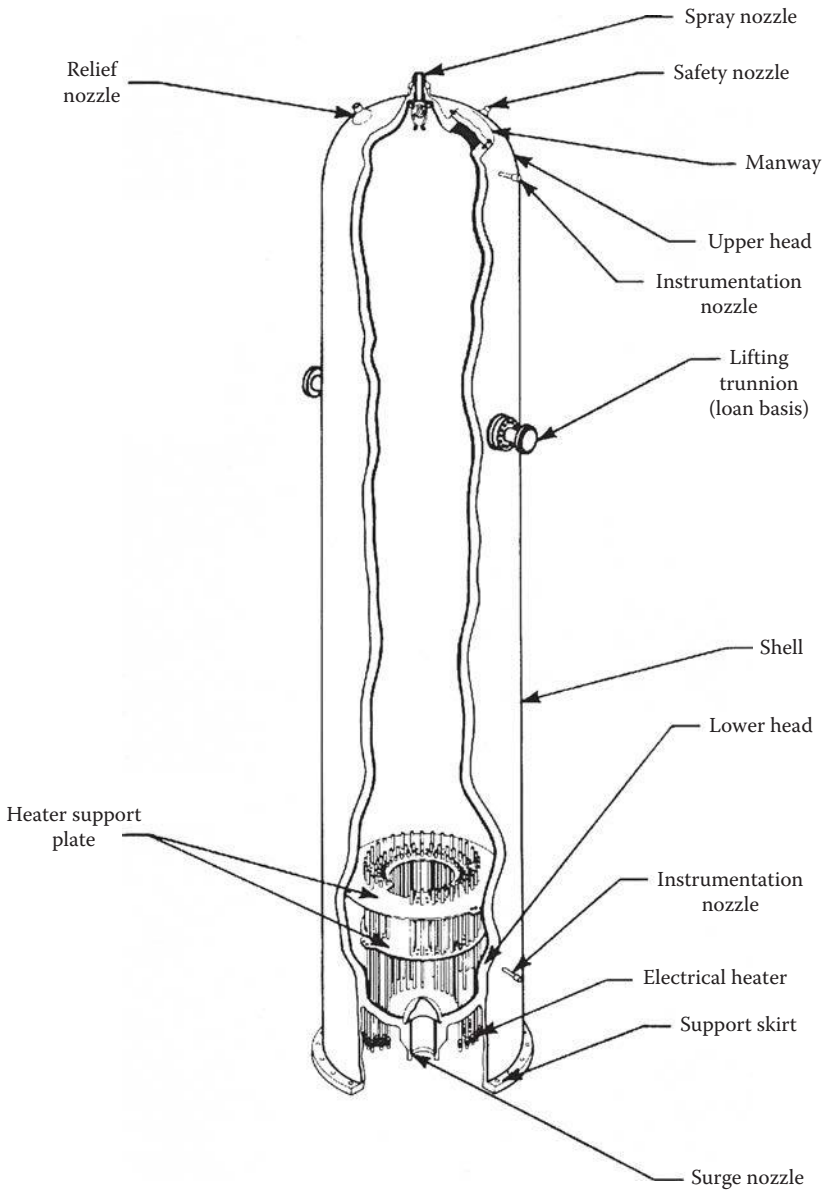
Steam Generator Principal Design Data

Number and Type	One Vertical, U-Tube Steam Generator with Integral Steam-Drum per Loop
Height overall	67 ft, 8 in. (20.6 m)
Upper shell OD	14 ft, 7-3/4 in. (4.5 m)
Lower shell OD	11 ft, 3 in. (3/4 m)
Operating pressure, tube side	2250 psia (155 bar)
Design pressure, tube side	2500 psia (172 bar)
Design temperature, tube side	650°F (343°C)
Full load pressure, shell side	
Two-loop plant	920 psia (63 bar)
Three-loop plant	964 psia (66 bar)
Four-loop plant	1000 psia (69 bar)
Steam flow per steam generator	3,813,000 lb/h (480 kg/s)
Maximum moisture at outlet (full load)	0.25%
Design pressure, shell side	1200 psia (82.7 bar)
Reactor coolant flow rate	35,075,000 lb/h (4419 kg/s)
Reactor coolant inlet temperature	621°F (327°C)
Reactor coolant outlet temperature	558°F (292°C)
Shell material	Mn-Mo steel
Channel head material	Carbon steel clad internally with stainless steel
Tube sheet material	Ni-Mo-Cr-V clad Inconel on primary face
Tube material	Thermally treated Inconel
Steam generator weights	
Dry weight, in place	346 tons (314,000 kg)
Normal operating weight, in place	422 tons (384,000 kg)
Flooded weight (cold)	560 tons (508,000 kg)

chemical neutron absorbers and appropriate allowance for physical phenomena that influence reactivity.

The principal natural phenomena that influence transient operation are the temperature coefficients of the moderator and fuel and the buildup or depletion of certain fission products. Reactivity balancing may occur through the effects of natural phenomena or the operation of the reactor control system using the RCCs or chemical "shim." A change in the temperature of moderator or fuel (e.g., due to an increase or decrease in steam demand) will add or remove reactivity (respectively) and cause the power level to change (increase or decrease, respectively) until the reactivity change is balanced out. RCC assemblies are used to follow fairly large load transients, such as load-follow operation, and for startup and shutdown.

The chemical shim system uses the soluble neutron absorber boron (in the form of boric acid), which is inserted in the reactor coolant during cold shutdown, partially removed at startup, and adjusted in concentration during core lifetime to compensate for such effects as fuel consumption and accumulation of fission products which tend to slow the nuclear chain reaction. The control system allows the plant to accept step



**FIGURE 2.11**  
Primary system pressurizer.

load increases of 10% and ramp load increases of 5% per minute over the load range of 15%–100% of full power, subject to “xenon” limitations near the end of core life. Equal step and ramp load reductions are possible over the range of 15%–100% of full power. Losses of reactor load up to 100% of rated power without a reactor trip can be accommodated by steam dump to the condenser in conjunction with the control system. Complete supervision of the nuclear and T-G islands is accomplished from a single plant control room.

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## 2.7 Detailed Description of Present Systems

This section describes the basic design and operating characteristics of a Westinghouse PWR plant. Differences from other manufacturers are mentioned if appropriate. Keep in mind that B&W and C-E made many of the domestic reactor vessels used by Westinghouse. The design is available in several ratings from approximately 600 megawatts electrical (MWe) to 1200 MWe. Exact ratings depend on several specific constraints (e.g., heat sink characteristics, customer system needs). Different ratings are attained through use of two, three, or four reactor coolant-piping loops, each loop comprising a steam generator, RCP, and interconnecting piping. The loops are each connected to a reactor vessel sized to contain nuclear cores composed of fuel elements of 12- or 14-ft length from 121 to 193 assemblies. The construction of the groups of fuel-bearing tubes ("rods") into assemblies will be described later. The objective of NSSS design is to satisfy the full range of utility requirements while maximizing the use of standard components. The principal parameters for the various power ratings are given in Table 2.1. The description given in this section is based on a four-loop plant with a 12-ft core having an electrical capacity of 1100 MWe. Refer to Figure 2.1 for the NSSS segment descriptions described below.

### 2.7.1 Primary Loop

The primary loop contains the heat source consisting of a nuclear fuel core positioned within a reactor vessel where the energy resulting from the controlled fission reaction is transformed into sensible heat in the coolant/moderator. The coolant is pumped to the steam generator where the heat is transferred to a secondary loop through several U-type tubes. The reactor coolant then returns back to the reactor vessel to continue the process. An electrically heated pressurizer connected to the loop maintains a pressure above the saturation pressure so that bulk boiling does not occur. See Figure 2.2 for the layout of the RCS. Figure 2.3 is a cut-away of the reactor vessel. Figure 2.10 gives details of the steam generator. Figure 2.9 shows the RCP, and Figure 2.11 is the pressurizer. In the RCS design of plants by C-E, two of the exit loops from the reactor vessel join to feed one steam generator. In a nominal "four-loop" plant, there are two such very large steam generators instead of four.

### 2.7.2 Secondary Loop

The secondary loop is the heat utilization circuit where dry steam produced in the steam generator flows to a T-G where it is expanded to convert thermal energy into mechanical energy and hence electrical energy. The expanded steam exhausts to a condenser where the latent heat of vaporization is transferred to the cooling system and is condensed. The condensate is pumped back to the steam generator to continue the cycle.

### 2.7.3 Tertiary Loop

The tertiary loop is the heat rejection loop where the latent heat of vaporization is rejected to the environment through the condenser cooling water. Depending on the specific site, this heat is released to a river, lake, ocean, or cooling tower system. The latter is becoming



the more common within the United States in part because of increasingly stringent environmental rules to reduce the thermal impact on natural water bodies.

#### 2.7.4 Confinement of Radioactivity

Use of a steam generator to separate the primary loop from the secondary loop largely confines the radioactive materials to a single building during normal power operation and eliminates the extensive turbine maintenance problems that would result from radioactively contaminated steam. Radioactivity sources are the activation products from the small amount of corrosion that is present in the primary loop over the 12–18-month reactor cycle, as well as from the occasional (<1 in 10,000) fuel rod that develops a crack and releases a small portion of its volatile fission products. Uranium dioxide fuel is very resistant to erosion by the coolant, so the rod does not dump its entire fission product inventory into the RCS.

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## 2.8 Component Design

Table 2.2 gives the essential parameters of the fuel rods and assemblies.

### 2.8.1 Fuel Assembly

A square array of fuel rods structurally bound together constitutes a fuel assembly. Control rod guide thimbles replace fuel rods at selected spaces in the array and are fastened to the top and bottom nozzles of the assembly. Spring clip grid assemblies are fastened to the guide thimbles along the height of the fuel assembly to provide support for the fuel rods. The fuel rods are contained and supported, and the rod-to-rod centerline spacing is maintained within this skeletal framework. Figure 2.4 is a cut-away of the assembly showing the structural details. The bottom nozzle of the fuel assembly controls the coolant flow distribution and also serves as the bottom structural element. The top nozzle functions as the fuel assembly upper structural element and forms a plenum space where the heated reactor coolant is mixed and directed toward the flow holes in the upper core plate. The spring clip grids provide support for the fuel rods in two perpendicular directions. Each rod is supported at six points in each cell of the grid. Four support points are fixed: two on one side of the grid strap and two similarly located on the adjacent side. Two more support points are provided by spring straps located opposite the fixed points. Each spring strap exerts a force on the fuel rod such that lateral fuel rod vibration is restrained, but small thermal expansions are allowed. Because the fuel rods are not physically bound to the support points, they are free to expand axially to accommodate thermal expansion and radiation-induced growth of the highly textured crystal structure of the Zircaloy cladding.

### 2.8.2 Grid Assemblies

Two types of grid assemblies are employed. One type features mixing vanes that project from the edges of the straps into the coolant stream to promote mixing of the coolant in the high heat region of the fuel assemblies. The other, a nonmixing type of grid (to minimize

flow pressure losses), is located at the bottom and top ends of the assembly where mixing for heat transfer purposes is not needed. The outside straps on all grids contain vanes which aid in guiding the grids and fuel assemblies past projecting surfaces during fuel handling or while loading and unloading the core. All fuel assemblies employ the same basic mechanical design.

### 2.8.3 Other Features of Assemblies

All assemblies can accept control rod clusters (the term "control rod cluster" is also referred to as "RCC" in the literature), but these are not used at every core location. Selected fuel assemblies have neutron sources or burnable absorber rods installed in the control rod guide thimbles. Fuel assemblies not containing RCCs, source assemblies, or burnable absorber rods are fitted with plugs in the upper nozzle to restrict the flow through the vacant control rod guide thimbles. This plug includes an end-flow mixing device to assure that these fuel assemblies have approximately the same coolant flow as those containing RCCs.

The fuel assembly design provides optimum core performance by minimizing neutron absorption in structural materials and maximizing heat-transfer capabilities. Mixing vane grids increase the heat-transfer capability of the fuel rods. High fuel utilization is achieved by minimizing the parasitic absorption of neutrons in the core. In the assembly design, the only structural materials in the fuel region are the spring clip grids, Zircaloy control rod guide thimbles, and Zircaloy fuel cladding. Zircaloy is used because it absorbs relatively few neutrons and has good mechanical and heat-transfer properties.

Because all fuel vendors bid on reloads for each others' reactor core designs, as well as their own, all design features of each core must be public knowledge. Each reload constitutes one-third of the core, so it must be compatible neutronically and mechanically to fuel already in the core to avoid power generation distortions within the core and equipment mismatches. This applies to all types of reactors, not just PWRs.

### 2.8.4 Control Rods

RCC assemblies are used for reactor startup or shutdown, to follow load changes, and to control small transient changes in reactivity. The control elements of an RCC assembly consist of cylindrical neutron absorber rods (control rods), having approximately the same dimensions as a fuel rod and connected at the top by a spider-like bracket to form rod clusters. Details of the RCC system are shown in Figures 2.6 and 2.7. The control rods, which are stainless steel tubes encapsulating a hafnium absorber material, extend the full length of the core when fully inserted. Full-length RCCs provide operational reactivity control and can shut the reactor down at all times, even with the most reactive RCC stuck out of the core. Each RCC is coupled to its drive shaft, which is actuated by a separate magnetically actuated drive mechanism mounted on the reactor vessel head. RCCs are arranged into groups and electrically interconnected so that the entire group moves together. Reactivity of the core is changed by raising or lowering a group in the core. Each control rod of an RCC moves vertically in its own tubular guide thimble (an empty tube with a reduced diameter near the base, a plug at the bottom and holes in the smaller diameter section to allow some coolant in or out). Located symmetrically within fuel assemblies, these thimbles replace fuel rods within the fuel assembly lattice. All fuel assemblies are built the same because each could be in a control rod position, or any other position in the core. When an assembly is not in RCC position, thimbles are plugged or

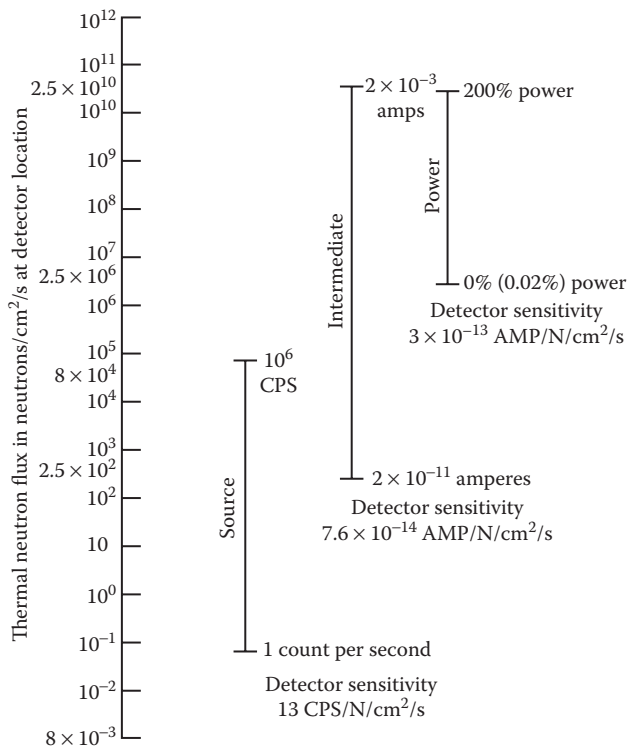
contain BP rods (absorber that is used up during the reactor cycle). The thimbles (1) act as guides for the control rods and (2) serve as “dashpots” (shock absorbers) for slowing control rod motion during reactor trip. In their fully withdrawn position, control rods do not leave the upper end of the guide thimbles. This assures that the rods are always properly aligned and reduces reactor coolant bypass through the thimbles. RCC design contributes to core performance improvement by providing a relatively homogeneous means of control. When control rods are withdrawn from the core, the resulting small water gaps do not cause significant power peaking.

### 2.8.5 Enrichment

Of the 193 fuel assemblies originally (no new reactors of this design have been built in many years) contained in a four-loop reactor first core, approximately 65 were of a low enrichment (2.10 weight percent U-235), 64 were an intermediate enrichment (2.60 w/o U-235), and 64 were a high enrichment (3.10 w/o U-235). Low and intermediate enrichment assemblies were arranged in a checkerboard pattern in the central portion of the core, whereas high enrichment assemblies were arranged about the periphery of the core (Figure 2.5). The first fuel cycle usually contained more excess reactivity than subsequent cycles due to the loading of all fresh (unburned) fuel. If soluble boron were the sole means of control, the concentration would be of the order of 1700 ppm, and the moderator temperature coefficient would be on the order of +7 pcm/°F (pcm stands for percent mille, i.e.,  $10^{-5}$ ) (a 15°F [9.5°C] “increase” in moderator temperature creates 0.001 reactivity “increase” under these high boron conditions). Because a large “positive” coefficient is undesirable, a reduction of the amount of control to be provided by chemical shim is accomplished by placing aluminum oxide–boron carbide burnable absorber material in the core. This material is depletable in the same fashion as U-235. Figure 2.7 shows the approximate BP locations in a four-loop core; Figure 2.8 shows the reduction in chemical shim brought about by this burnable absorber. As the fuel and BP deplete, the power shifts toward the center of the core, and this shift must be accounted for in design calculations. At the end of life, the power distribution is again quite uniform. Today, these older systems have achieved an optimal balance of BP and reload enrichment to achieve longer core life and much higher burnups than were initially believed possible.

### 2.8.6 Startup

Reactor startup neutron sources must be used to raise the neutron multiplication rate to levels detectable by the flux detectors outside the reactor, but still less than criticality. The fresh fuel configuration and the initially low core reactivity (from random fissions) by themselves would not permit a safely controlled startup. The source range as seen by the reactor instrumentation is shown in Figure 2.12. Neutron sources are of two types: (1) a primary source (which is active for initial reactor startup and startup early in the life of the first core) and (2) a secondary source (used for later startup of the reactor and which is activated during the operation of the reactor). The primary source is usually a spontaneously fissionable californium isotope. Secondary sources contain a mixture of antimony and beryllium (Sb–Be). The Sb becomes radioactive, emitting high-energy gamma particles that spall neutrons from the Be. The primary and secondary sources are similar to a control rod in mechanical construction. Both types of source rods are clad in stainless steel. The secondary source rods contain Sb–Be pellets which are not initially active. The primary source rods contain sealed capsules of source material at a specified axial position.



**FIGURE 2.12**  
Nuclear instrumentation system neutron detectors and ranges of operation.

Cladding encapsulation is completed by seal welding the end plugs. The specific core location of the sources is determined during final design of the core to assure adequate neutron flux at the source range detectors at all times.

Reactor coolant piping and fittings are made of stainless steel or are carbon steel-clad with stainless steel. Carbon steel is used for the pressurizer relief line, which connects the pressurizer safety and relief valves to the flanged nozzle on the pressurizer relief tank, and for the nitrogen supply, vent, and drain lines for the pressurizer relief tank. The pressurizer surge and spray lines, loop drains, and connections to other systems are of austenitic stainless steel. Except for the flanged pressurizer safety valve outlet nozzles, all joints and connections are welded. Thermal sleeves are installed where high thermal stresses could develop because of rapid changes in fluid temperature during transients. Valves, piping, and equipment which operate at elevated temperatures are normally covered with thermal insulation to reduce heat losses. Insulation covering the piping and components of the RCS are designed to facilitate its removal for periodic in-service inspections. Insulation used for the RCS is strictly specified to limit chlorides and other halogens. Reactor vessels are frequently insulated with reflective metal insulation systems.

### 2.8.7 Construction Materials

All valves in contact with reactor coolant are constructed primarily of austenitic stainless steel and employ special materials such as corrosion-resistant hard surfacing

and packing. Back seats and stem leak-offs reduce leakage to the containment to essentially zero. The pressurizer safety valves are spring-loaded and self-actuating, with back pressure compensation. The pressurizer also has power-operated relief valves which operate automatically to prevent overpressure. Remotely operated stop valves are provided to isolate these relief valves should excessive leakage occur. Automatic spray valves regulate the pressurizer spray to provide overpressure control. Locally adjustable throttling valves in parallel with the spray control valves deliver a small continuous flow through each spray line and the pressurizer. Local vents permit filling of the system. The piping is arranged so that any liquid discharged through a vent is collected in a container or drain without spilling. Reactor coolant loop isolation valves may be included in a plant design to facilitate some maintenance operations or operation with a loop out of service.

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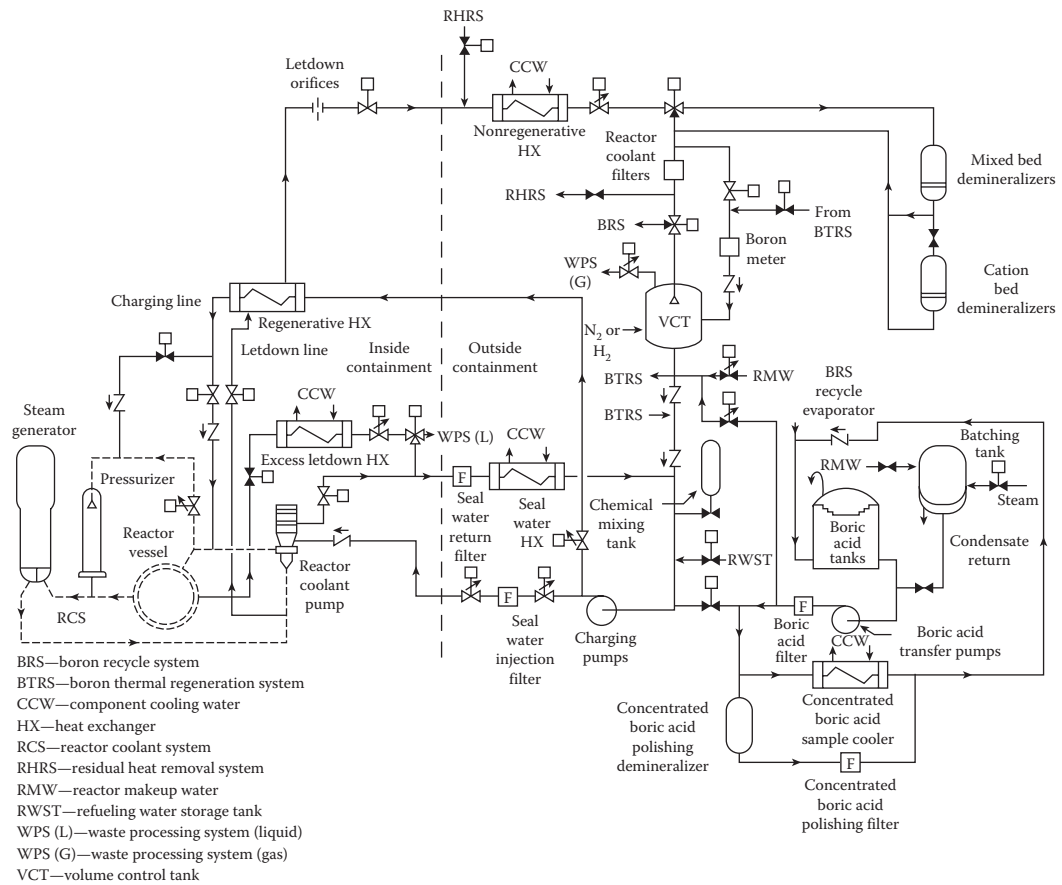
## 2.9 Auxiliary Systems

The chemical and volume control system (CVCS; Figure 2.13) is designed to carry out the functions listed below in support of the RCS:

- Maintain required water inventory in the RCS by maintaining programmed water level in the pressurizer.
- Reduce the concentration of corrosion and fission products in the reactor coolant.
- Provide a means for filling, draining, and pressure testing the RCS.
- In conjunction with the boron thermal regeneration system (BTRS), adjust the boric acid concentration of the reactor coolant for chemical shim control solution.
- Provide a means for control of RCS chemistry.
- Provide high-pressure seal water for the RCP seals.

### 2.9.1 Auxiliary Flows

During power operation, a continuous feed-and-bleed stream is maintained to and from the RCS. The feed rate to the RCS is automatically controlled by the pressurizer water level, while the bleed rate can be set by selecting the proper combination of letdown orifices to meet plant operational requirements. Letdown water leaves the RCS and flows through the shell side of the regenerative heat exchanger, where it gives up its heat to makeup water being returned to the RCS. The letdown water then flows through letdown orifices where its pressure is reduced, then through a nonregenerative heat exchanger, followed by a second pressure reduction by a low-pressure letdown valve. After passing through a mixed-bed demineralizer where ionic impurities are removed, the water flows through the reactor coolant filter and into the volume control tank (VCT) via a spray nozzle. An alternate path downstream of the mixed-bed demineralizers can be used to direct the letdown flow to the boron recycle system (BRS). The vapor space in the VCT contains hydrogen, which dissolves in the coolant and determines the hydrogen concentration in the RCS. Fission or other noncondensable (He, H<sub>2</sub>) gases present are removed from the system by



**FIGURE 2.13**  
 Chemical and volume control system flow diagram.

venting of the VCT continuously, intermittently, or prior to plant shutdown. Continuous purging of the VCT considerably reduces the activity level of the reactor coolant. The charging pumps take the coolant from the VCT or the BTRS and send it along two parallel paths: back to the RCS through the tube side of the regenerative heat exchanger and to the seals of the RCPs. The RCP seal injection flow enters the pump between the labyrinth seals and the number 1 seal. Here, the flow divides with some water flowing into the RCS and the remainder leaving the pumps as controlled seal leakage. From the pumps, the controlled leakage water goes to the seal water heat exchanger and then returns to the charging pump suction for another circuit. If the normal letdown and charging path through the regenerative heat exchanger is not operable, water injected into the RCS through the RCP seals is returned to the charging pump suction through the excess letdown heat exchanger and seal water heat exchanger. Surges in the RCS volume are normally accommodated by the pressurizer. However, the VCT is designed to allow for pressurizer level mismatches that may occur. If the VCT is full, a high water level signal diverts letdown flow to the BRS or waste processing system.

### 2.9.2 Water Sources

Low-level signals from the VCT initiate reactor makeup control or flow from the refueling water storage tank (RWST) as a backup. Makeup to the RCS can come from the sources listed below:

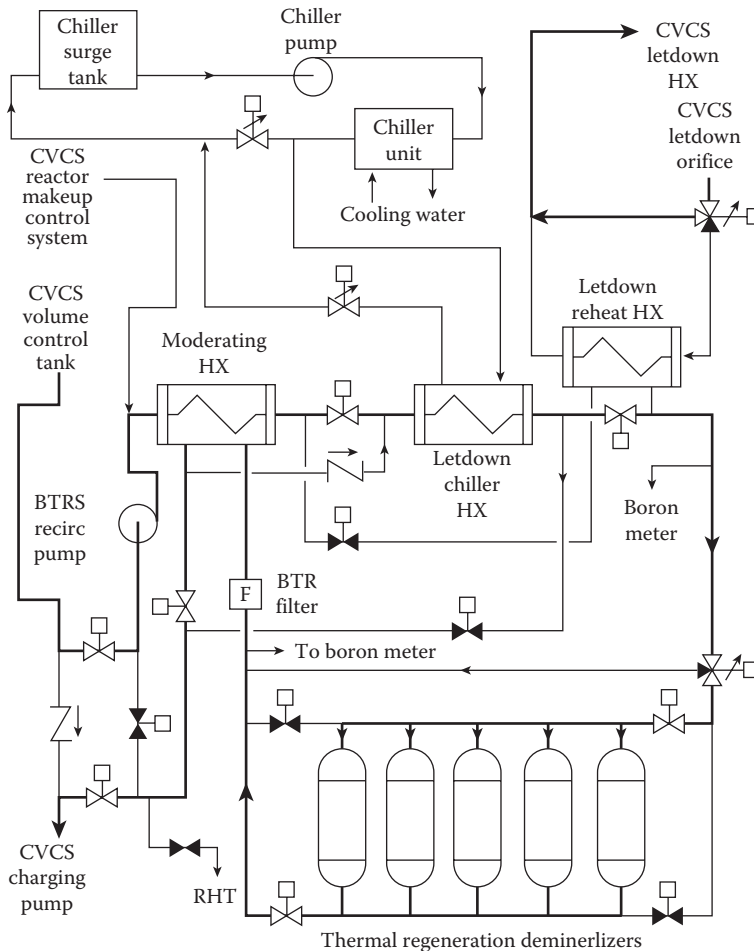
- Demineralized and deaerated water supply used to reduce the concentration of dissolved boric acid in chemical shim to accommodate core burnup
- Chemical mixing tank to add small quantities of hydrazine for oxygen scavenging or lithium hydroxide for pH control
- RWST for emergency makeup of borated water
- Boric acid from the boric acid tanks
- A blend of demineralized and deaerated water and concentrated boric acid to match the reactor coolant boron concentration for normal plant makeup

A continuing source of clean water and concentrated solution of boric acid must be provided to replenish these sources. An alternate flow path downstream of the VCT directs the charging pump suction flow to the BTRS to effect boron concentration changes during load follow operations and other plant operations. A boron concentration meter provides accurate measurement and monitoring of boron concentration in the reactor coolant. The readout provides control information which the plant operator can utilize in routine operations with makeup, let down, chemical treatment of reactor coolant, and the regulation of boron concentration. These data are also used in a boron follow program to assist the operator during various boron concentration changes, for example, load follow, fuel burnup, hot standby, and cold shutdown. Sample points for the measurement unit are located upstream and downstream of the boron thermal regeneration demineralizers. Provisions are also included for boric acid purification. Concentrated boric acid is passed through a demineralizer to remove water-borne impurities such as aluminum, calcium, and magnesium. A filter is also provided for removal of suspended solids.

### 2.9.3 BTRS

The primary function of the BTRS is to vary the RCS boron concentration during reactor power changes which include daily load follow operations. Boron concentration changes are accomplished automatically by the BTRS through an ion exchange (passive) method. The BTRS makes use of a temperature-dependent ion exchange process to store boron from and release boron to the RCS. Operation of the BRS and evaporators is not required during normal load follow operation. As a result, there is a significant reduction in the water to be processed during normal plant operations and a reduction in BRS requirements. The BRS provides a diverse and redundant method of making boron concentration changes during load follow operations. The BTRS can also assist in making RCS boron concentration changes associated with core burnup, shutdowns, and refuelings. BTRS schematics are shown in Figure 2.14.

The BTRS makes use of a temperature-dependent ion exchange process to store boron and release boron to the RCS without discharging water to the BRS for later evaporation. The BTRS, which operates in conjunction with the CVCS, consists mainly of several demineralizers, a chiller unit, a chiller pump, heat exchangers, a BTRS recirculation pump, a BTR



**FIGURE 2.14**  
Boron thermal regeneration system in dilution mode.

filter, valves, and associated piping. This equipment controls the temperature and flow rate of the fluid entering the BTRS demineralizers. An alternate charging pump suction path in the CVCS is provided which allows the flow from the VCT to pass through the BTRS when boron concentration changes are required. The fluid temperature is reduced to 50°F (10°C) to store boron on the ion exchange resin and thus dilute the reactor coolant. The fluid temperature is raised to 140°F (60°C) to release boron from the ion exchange resin and thus borate the reactor coolant. The rate of boration or dilution is controlled by varying the flow rate through the BTRS demineralizers. The flow path through the BTRS is different for dilution (storage) than for boration (release). During dilution, the letdown stream from the VCT enters the moderating heat exchanger, and from there it passes through the letdown chiller heat exchanger. These two heat exchangers cool the BTRS stream to 50°F (10°C) before it enters the demineralizers. The source of cold water for the letdown chiller heat exchanger is a closed-loop circuit consisting of the chiller unit, chiller pump, and chiller surge tank. The letdown reheater heat exchanger is valved out on the tube side during dilution. The temperature of the BTRS stream at the point of entry to the demineralizers is controlled automatically by the temperature control valve which controls the shell-side flow to the letdown chiller heat

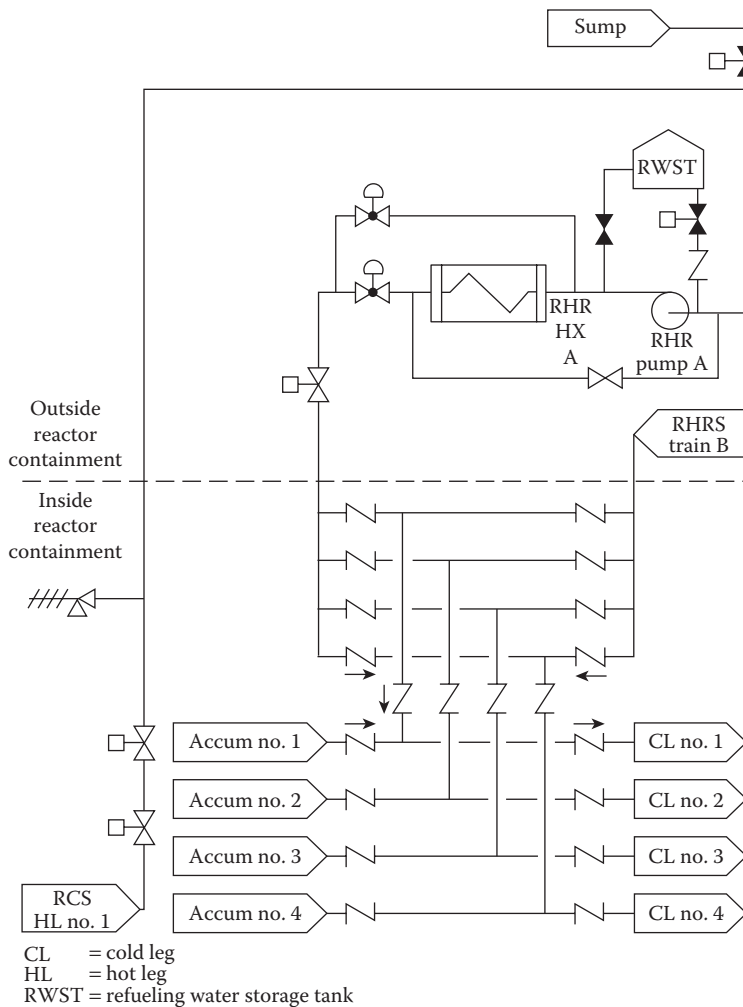


exchanger. The letdown stream passes through the demineralizers and enters the moderating heat exchanger shell side where it is heated by the incoming letdown stream. The flow is then returned to the VCT outlet line. Therefore, for dilution, a decrease in the boric acid concentration in the reactor coolant is accomplished by sending the flow from the VCT, at relatively low temperatures (50°F) (10°C), to the thermal regeneration demineralizers. The resin, which was depleted of boron at high temperature during a prior boration operation, is now capable of storing boric acid from the low temperature flow from the VCT. Reactor coolant with a lower boron concentration leaves the demineralizers and is directed to the CVCS. During boration, the charging pump suction stream enters the moderating heat exchanger tube side, bypasses the letdown chiller heat exchanger, and passes through the shell side of the letdown reheat heat exchanger. The moderating and letdown reheat heat exchangers heat the BTRS stream to 140°F (60°C) before the stream enters into the demineralizers. The temperature of the BTRS stream at the point of entry to the demineralizers is controlled automatically by the temperature control valve that controls the flow rate on the tube side of the letdown reheat heat exchangers. The BTRS stream then passes through the demineralizers, enters the shell side of the moderating heat exchanger, passes through the tube side of the letdown chiller heat exchanger, and enters the charging pump suction. The temperature of the BTRS stream entering the charging pump is controlled automatically by the temperature control valve that controls the shell-side flow rate on the letdown chiller heat exchangers. Thus, for boration, an increase in the boric acid concentration of the reactor coolant is accomplished by sending the flow from the VCT, at relatively high temperatures (140°F) (60°C), to the thermal regeneration demineralizers. The water flowing through the demineralizers releases boron that was stored on the resin at low temperature during a previous dilution operation. The boron-enriched reactor coolant is returned to the RCS via the CVCS. The BTRS recirculation pump provides the motive force to route flow from the VCT through the demineralizers during boration and dilution modes. When the BTRS is not being used for boration/dilution, it is isolated from CVCS and the recirculation pump can operate to change the temperature of the demineralizer beds for the next anticipated mode of operation. For example, after completion of a dilution mode, with demineralizers at 50°F (10°C), the BTRS recirculation pump is used to increase the temperature of the demineralizers to 140°F (60°C) for the subsequent boration mode.

#### 2.9.4 Residual Heat Removal System

The primary function of the residual heat removal system (RHRS) is to transfer heat energy from the core and RCS during plant cool down and refueling operations. The system is designed to reduce the RCS temperature to 140°F (60°C) within 20 h after reactor shutdown. Provisions are made for continued flow of the reactor coolant to the CVCS during shutdown. The RHRS may also be used to transfer refueling water between the refueling cavity and the RWST at the beginning and end of refueling operations. The residual heat removal pumps and heat exchangers are also utilized as part of the safety injection system for emergency core cooling in the event of a loss-of-coolant accident (LOCA). The accident mode of operation is more fully described in Section 2.10, Safety Injection System (SIS; part of the Engineered Safeguards System). A flow diagram of the RHRS is shown in Figure 2.15 and represents the configuration for a four-loop NSSS. The configuration for two-loop and three-loop plants differs only in the size of the components and number of branch injection lines to the RCS.

The RHRS consists of two independent, redundant mechanical subsystems, each of which receives electrical power from one of two separate and redundant electrical power trains. Each subsystem consists of one RHR pump; one RHR heat exchanger; and the required piping, valves, and instrumentation. The RHR pumps and heat exchangers are located in



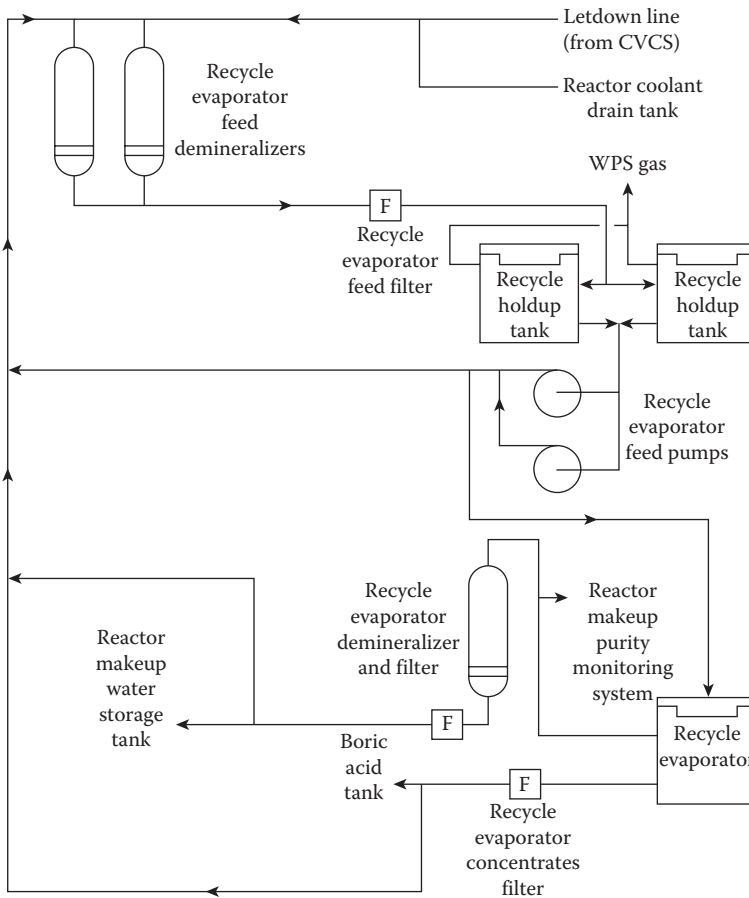
**FIGURE 2.15**  
Safety injection/residual heat removal system in RHR mode.

the auxiliary building as close as practical to the containment. The piping configurations of the two subsystems are identical, with no major piping cross connects between subsystems in the auxiliary building. Each subsystem has a suction line from an RCS hot leg with two normally closed motor-operated series isolation valves providing the isolation necessary to meet the RCS pressure boundary and containment isolation requirements. Each subsystem also has suction from the containment sump and RWST. These suction sources are required by the RHRs for its operation during mitigation of a LOCA as part of the SIS. The discharge line of each pump is routed to its respective RHR heat exchanger and thence into the containment. A bypass loop is provided around the RHR heat exchanger to permit control of RCS cool down while maintaining a constant total pump flow. A minimum flow for the RHR pump is provided by a recirculation line from the heat exchanger outlet to the pump suction, which remains open in all system operating modes. Inside the containment, each pump discharge is divided into four branch lines. Each branch line delivers to its respective cold leg through the appropriate accumulator discharge line. No cross connects

are present between the two subsystems prior to the discharge headers injecting into the accumulator discharge lines. The branch lines inside the containment are isolated from the RCS cold legs by check valves. During system operation, each RHR pump takes suction from one of the RCS hot legs by its separate suction line. The pumps then discharge flow through the RHR heat exchangers which transfer heat from the hot reactor coolant fluid to the component cooling water circulating through the RHR heat exchanger shell side. The cooled RCS flow is then returned to the RCS cold legs by the shared SIS piping.

**2.9.5 BRS**

The BRS (Figure 2.16) collects and processes deaerated reactor coolant effluents for reuse as makeup to the RCS as boric acid and reactor makeup water. The BRS provides a means of recycling borated reactor coolant so as to minimize activity releases. The BRS collects water from the RCS through the CVCS letdown line. The letdown is diverted to the BRS as a result of changes made to the RCS boron concentration by the CVCS reactor makeup control. In addition, the BRS collects the overflow of the RCS during heatup operations. These occur during plant shutdown and startup, refueling, and dilution resulting from the slow burnup of the core. Normally, all boron changes required for load changes are made by the BTRS with



**FIGURE 2.16**  
Boron recycle system.

no discharges to the BRS. However, as a backup to the BTRS, boron concentration changes for load follow can be made by using CVCS makeup and by discharging reactor coolant effluent to the BRS for processing. The BRS, by means of demineralization, gas stripping, and evaporation, reclaims the boric acid and the primary water. Letdown reactor coolant from the CVCS and deaerated liquid drains is passed through the recycle evaporator feed demineralizers where lithium and fission products are removed. The fluid then flows through a recycle evaporator feed filter and into the recycle holdup tanks. The borated liquid is then pumped to the recycle evaporator package by one of the recycle evaporator feed pumps. Here hydrogen and residual fission gases are removed in the stripping column before the liquid enters the evaporator shell. The evaporative process produces a batch of 4-weight-percent boric acid solution. Distillate from the evaporator is pumped to the evaporator condensate demineralizer and then to the reactor makeup water storage tank. Also located in this flow path is a recycle evaporator condensate filter. To provide that the reactor makeup water storage tank will not become contaminated, a scintillation counter is located in the line leading into the tank. A high radiation level signal at the monitor directs distillate back to the recycle evaporator feed demineralizer. Concentrates at 4-weight-percent boric acid are normally pumped to the boric acid tanks through the recycle evaporator concentrates filter. If for some reason the boric acid cannot be discharged to the boric acid tanks, it can be diverted back to the recycle holdup tanks. A small sidestream is routed to the reactor makeup purity monitoring system. This system is provided to monitor the condensate from the recycle evaporator to ensure that it conforms with chemical specifications before flowing to the reactor makeup water storage tank.

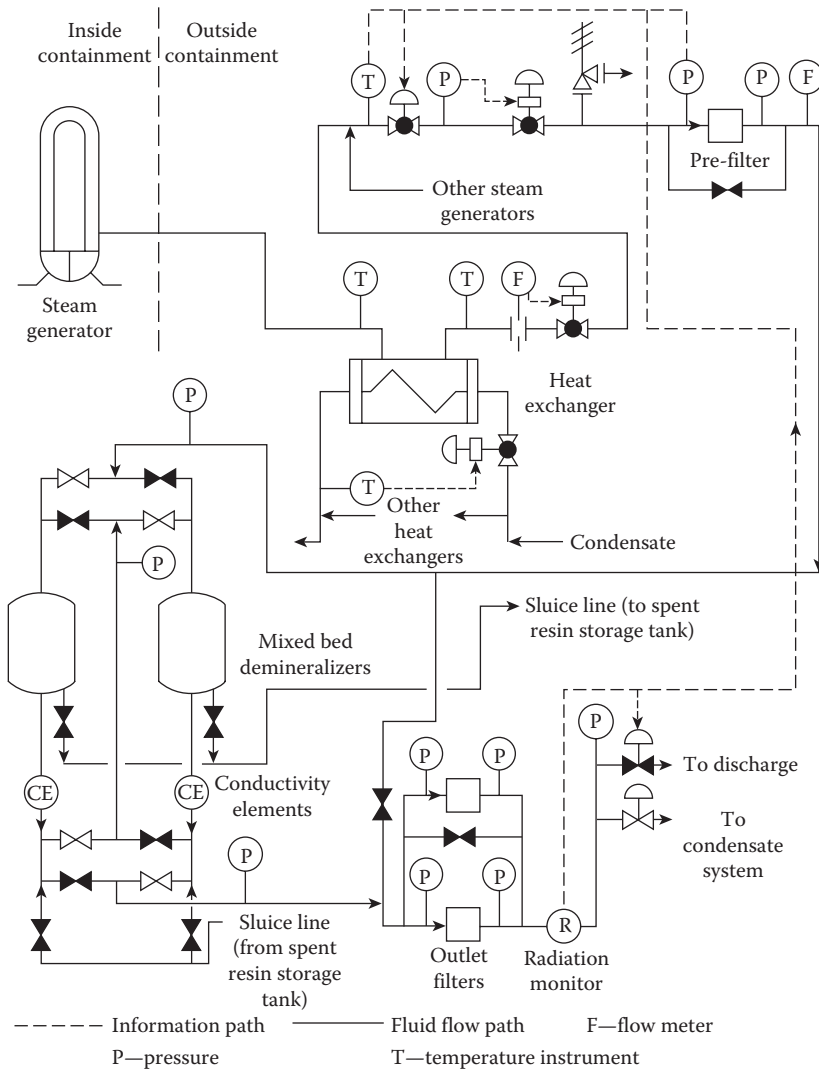
### 2.9.6 Steam Generator Blowdown Processing System

The steam generator blowdown processing system (SGBPS) (Figures 2.17 and 2.18) processes blowdown from steam generators to meet chemical specifications that are suitable to allow recycle into the main condenser and to meet radiochemical specifications that are suitable for discharge to the environment when discharge is required. During normal operation, blowdown from each steam generator enters a heat exchanger where the temperature is reduced by condensate water. The flows are then measured before being manifolded. The pressure is reduced, and the blowdown is directed through the prefilter and mixed-bed demineralizers in series. The fluid flows through a radiation monitor and is normally recycled to the main condenser but may be discharged to the environment through the discharge canal when required. Instrumentation is provided at strategic points to monitor the functional integrity and operating efficiency of the system. An increase in conductivity, indicated at the outlet of each demineralizer, signals resin bed exhaustion. When the upstream bed is exhausted, flow is directed around that bed. With the aid of the spent resin sluice pump, spent resin is transferred to the spent resin storage tank and new resin is charged to the demineralizer. This fresh demineralizer is then valved back online as the new downstream bed. Before resin disposal, the spent resin storage tank is fluidized with nitrogen gas or sluice water to loosen the resin. The tank is pressurized with nitrogen to discharge the resin to the solid waste processing system.

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## 2.10 Engineered Safeguards Systems

Engineered safeguards systems are integrated with the other main and auxiliary systems to protect the plant, its personnel, and the general public. After 2001, security systems were beefed up in the United States and elsewhere to meet the terrorist threat. These latter

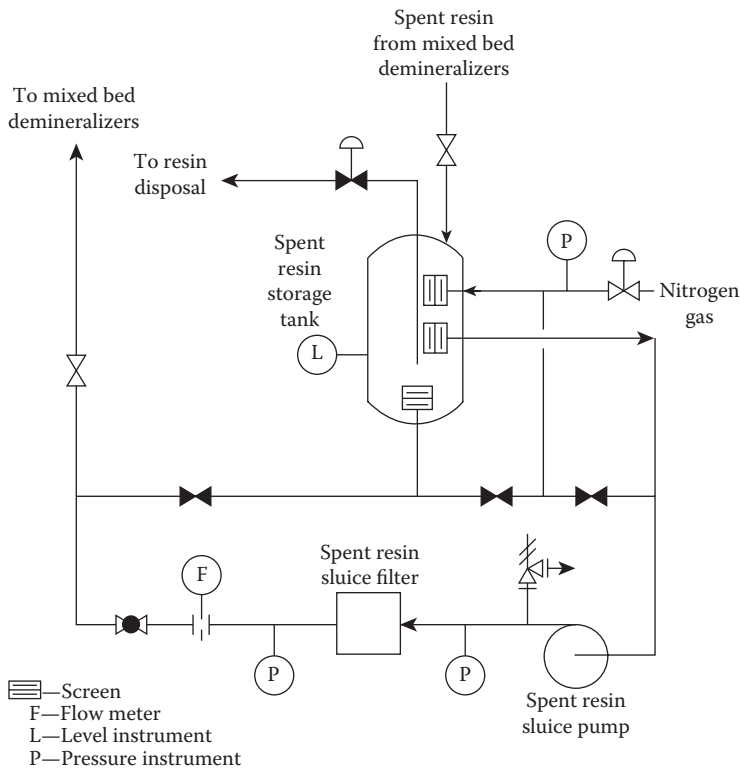


**FIGURE 2.17**  
Steam generator blowdown processing system.

systems, of course, are not discussed in the literature. Along with the plant safeguards systems are computers, alarms, interlocks, and sophisticated software to see that all systems work as they are supposed to in a wide variety of hypothesized upset and accident conditions; those that could be part of the everyday operation of the facility, as well as those that are extremely rare but are postulated to be potentially catastrophic if not contained or prevented from developing. What follows is a breakdown by component subsystems of the function and interaction of the various parts of the safeguards system.

### 2.10.1 SIS

The SIS has multiple functions. Its primary function is to provide emergency core cooling in the event of a LOCA resulting from a break in the RCS. The SIS also provides a safety



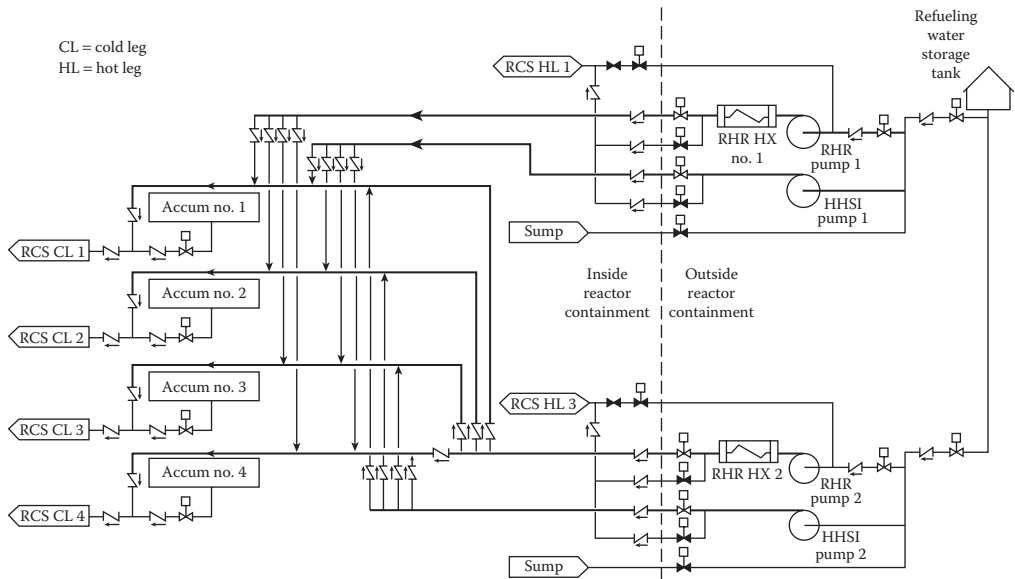
**FIGURE 2.18**  
SGBPS spent resin storage tank.

grade method for addition of negative reactivity via injection of borated water to meet shutdown requirements, or to compensate for the reactivity increase caused by cool down transients such as a steam line break. The SIS positive displacement hydrotest pumps also provide a backup source of RCP seal injection water, diverse to the CVCS. In the unlikely event of a LOCA, the SIS is designed to limit increases in fuel-clad temperatures, core geometry distortion, and metal-water reaction for all break-ups to and including the double-ended severance of a reactor coolant loop. For the more probable break sizes inside diameter (5 in. [12.7 cm]), the SIS is designed to minimize core damage by providing flow to the core that is sufficient to prevent the mass depletion-related uncovering of the core based on current analytical models. The system is designed to provide not only emergency core cooling, but also continued cooling during the long-term phase after the accident.

### 2.10.2 High-Pressure Injection

High-pressure safety injection water is provided by separate high-head pumps while lower pressure injection water is supplied by the RHRS pumps. Passive accumulator tanks are inside the containment to provide for fast injection of water after a LOCA. Figure 2.19 shows the injection phase of the SIS and is the configuration for a four-loop plant. The configurations for two- and three-loop plants differ only in the size of the components and the number of branch injection lines to the RCS.

The SIS consists of two independent and redundant mechanical subsystems, each of which receives electrical power from one of two separate and redundant class 1E electrical



**FIGURE 2.19**  
Safety injection/residual heat removal system, injection mode.

power trains, and each of which receives an actuation signal from one of two separate and redundant actuation trains. The major components are two high-head safety injection (HHSI) pumps, two RHR pumps, and four accumulators. There is also a single positive displacement pump which provides emergency injection flow if the centrifugal charging pumps are not available. It is not directly associated with subsystem A or B and can be powered from either electrical train. The piping configurations and valving associated with each of the two subsystems are identical. There are no major piping interconnections between the two subsystems in the auxiliary building with the exception of the containment, the individual loop injection branch lines from each of the four main supply headers from the RWST. Inside, the pumps are directed into a common pipe header and injected into the respective accumulator discharge lines to the RCS cold legs. Each pump delivers to all of the cold legs. Additionally, each subsystem delivers to the hot leg of one loop.

Each of the two mechanical subsystems contains one HHSI pump and one RHR pump. With the exception of drawing from a common suction header from the RWST, these pumps share no piping and are not connected outside the containment. Lowhead safety injection is provided by the RHRS. The HHSI pump is a dedicated component of the SIS and does not provide any function required for normal plant operation. This permits the HHSI pump and associated piping to be retained in a ready configuration which requires only the pump to start to initiate safety injection. Each HHSI pump is also provided with a full flow test loop which permits periodic verification of hydraulic performance. Design of the HHSI pump is such that testing or inadvertent start will not result in injection to the cold leg while the RCS is at normal operating temperature and pressure.

### 2.10.3 System Safeguards

An inadvertent safety injection signal will not result in lifting the pressurizer power-operated relief or safety valves. A continuous minimum flow path which does not require

isolation during the injection phase is provided so that an inadvertent pump start will not result in the deadheading of the HHSI pump. Each HHSI pump discharge header feeds each RCS cold leg through individual branch lines. Redundancy of the two independent subsystems is incorporated to provide for reliability of operation and continued core cooling, even in the event of a failure of a single component in the fluid system or related control and power supply systems to respond actively in accordance with its design function. The HHSI and the RHR pumps can take direct suction from the containment sump, thereby increasing the system reliability for long-term post-accident recirculation. The HHSI and RHR pumps and all motor-operated valves, except the normally open accumulator isolation valves, are located outside the containment to provide for ease of access for maintenance. The accumulator tanks are passive components normally requiring no maintenance and are located inside the containment to provide rapid injection of water following a LOCA.

Operation of the SIS following a LOCA is described in the following paragraphs in three distinct phases:

- Injection phase
- Cold-leg recirculation phase
- Cold-leg and hot-leg recirculation phases

In addition, this system can be operated with a nonfaulted RCS to provide an alternate source of borated makeup water injection. This function is described separately in the following paragraphs.

#### 2.10.4 SIS Components

The principal mechanical components of the SIS that function immediately following a LOCA are the accumulators, the HHSI pumps, the RHR pumps, the RWST, and the associated piping and valves. Because the SIS components have no active function during normal power operation, they are maintained in a configuration aligned for safety injection immediately upon pump start; only check valves are required to change position. For large pipe ruptures, the RCS would be rapidly depressurized and voided of coolant. A high flow rate of emergency coolant would therefore be required to quickly recover the exposed fuel rods to limit possible core damage. This high flow is provided by the accumulators, followed by the HHSI and RHR pumps, all discharging into the cold legs of the RCS. The HHSI and RHR pumps are aligned to take suction directly from the RWST. For smaller breaks, depressurization of the RCS will occur at a slower rate. The HHSI and RHR pumps will be started upon receipt of the actuation signal and will run on miniflow until the RCS pressure falls below the respective shutoff heads of the pumps. Similarly, the accumulator discharge will begin automatically as the RCS pressure decreases below the accumulator tank pressure. The active safety injection phase is actuated by any one of the following:

- Low pressurizer pressure reported by two out of four signals
- Excessive cool down protection signals
  - Low compensated steam line pressure as reported by two out of four signals in any steam line. This protects against secondary breaks during power generation.



- Steam pressure high negative rate as reported by two out of four signals in any one steam line plus low pressurizer pressure in two out of four channels. This protects against secondary breaks during planned cool down from power generation.
- High containment pressure as reported by two out of four signals
- Manual actuation
- The receipt of the safety injection signal initiates the following emergency core cooling actions:
  - The onsite emergency diesel generators are started.
  - HHSI pumps are started.
  - RHR pumps are started.

### 2.10.5 Cold-Leg Recirculation Mode

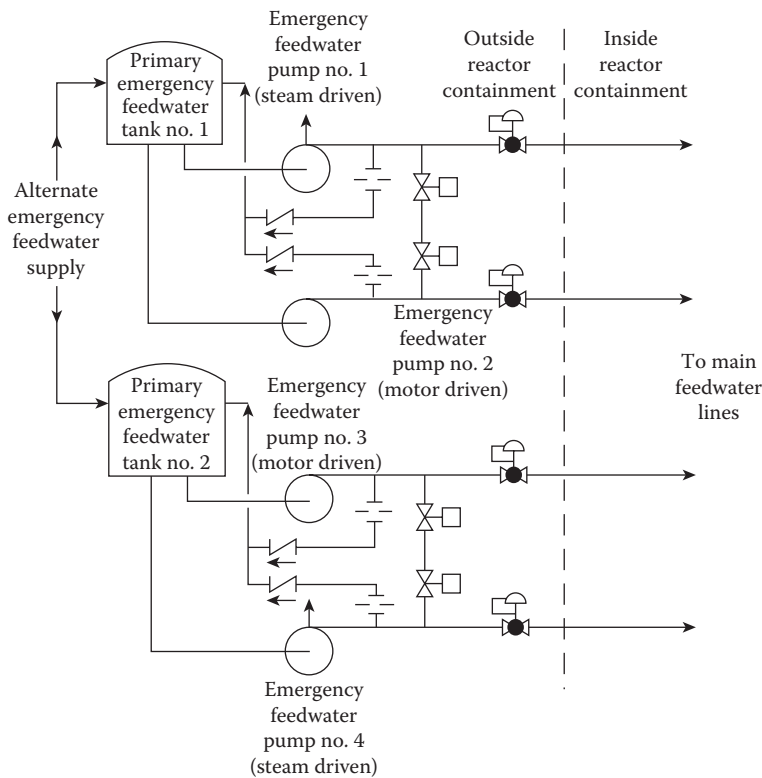
The changeover from the injection mode to the cold-leg recirculation mode is initiated automatically when the RWST level channels indicate an RWST level less than a low-level setpoint in conjunction with the initiation of the engineered safeguards actuation signal. Protection logic is provided to automatically open the recirculation valve in each of the SIS subsystems when the above signals are generated. To prevent backflow to the RWST of contaminated water from the containment sump, the HHSI pump mini-flow isolation valves are automatically closed at this time. This automatic action would align each of the SIS subsystems for cold-leg recirculation. In this mode, the HHSI and RHR pumps for each subsystem take suction directly from one of the two containment sump recirculation lines. All pumps will continue to operate and will deliver water to the RCS cold legs without interruption throughout the automatic recirculation switchover operation. As in the injection phase, no operator action is required to initiate realignment of either HHSI or the RHR pump. Furthermore, each pump draws directly from the containment sump and operates independently of the operation of any other pump.

Approximately 24 h after the switchover to cold-leg recirculation, simultaneous flow to the RCS cold legs and hot legs is established for the purpose of long-term core cooling. The alignment of the HHSI pumps and the RHR pumps remains essentially the same as during the cold-leg recirculation phase, except that a substantial portion of the flow is directed to the RCS hot legs. The purpose of the change in flow alignment is to provide for termination of boiling and to prevent buildup of boron from interfering with core cooling following a large cold-leg break. Switchover to this phase is performed manually from the control board.

The HHSI pumps also serve as safety grade backup to the CVCS charging pumps for RCS boration. Manual starting of the HHSI pumps will result in the injection of borated water as RCS pressure is decreased below the shutoff head of the pump. A manual throttling valve, operable from the main control board, is provided to permit control of the injection flow rate as the RCS pressure is reduced.

### 2.10.6 Emergency Feedwater for Secondary Loop

The primary function of the emergency feedwater system (EFWS; Figure 2.20) is to supply feedwater to the steam generators following accident or transient conditions when the



**FIGURE 2.20**  
Emergency feedwater system, flow diagram.

main feedwater system is not available. The EFWS thereby maintains the capability of the steam generators to remove plant-stored heat and core decay heat by converting the emergency feedwater to steam which is then discharged to the condenser or to the atmosphere. Although the EFWS is also capable of supplying feedwater to the steam generator during normal plant operations of startup, shutdown, and hot standby, it is normally not used for this service. Instead, a small startup pump(s) is located in the main feedwater system and used to supply the reduced amount of feedwater required during these normal plant operations.

The EFWS consists of two identical subsystems, each of which receives electrical power from one of two separate Safety Class 1E electrical power trains. Each subsystem consists of a primary emergency feedwater tank; one motor-driven emergency feedwater pump; one turbine-driven emergency feedwater pump; and the required piping, valves, instruments, and controls necessary for system operation. The motor-driven and turbine-driven pumps are located in the emergency feedwater pump building. The use of motor-driven and turbine-driven pumps satisfies the requirement that the pumps be powered by diverse power sources. In operation, the emergency feedwater pumps take suction from the primary emergency feedwater tanks and discharge the water into the main feedwater piping between the steam generator feed nozzle and the last check valve in the main feedwater line. Each pump is provided with an orificed recirculation line leading back to the primary emergency feedwater tank. This line provides recirculation flow any time the pumps are operating. The steam supply line for

each turbine-driven pump is connected to the main steam line from a steam generator. This line is fitted with a steam admission valve which is a pneumatically operated valve arranged to fail-open on loss of air or electrical power. A primary emergency feedwater supply tank, to which the suction of the emergency feedwater pumps are normally aligned, is provided in each subsystem. The tanks are safety grade and seismically qualified. Each tank contains a quantity of condensate quality water sufficient to allow the plant to be maintained in hot standby for 13 h then allow a 5 h cool down of the plant to 350°F (176.7°C). The maximum permissible water temperature in the primary emergency feedwater supply tank is 120°F (48.9°C).

An alternate emergency feedwater source (condensate storage tanks) should also be provided. The alternate water source should contain sufficient water to allow the plant to be maintained in hot standby conditions, with one RCP operating, for 2 days (48 h) beginning 18 h after reactor trip. The maximum water temperature in the alternate water supply should be 120°F (48.9°C). Normally open, fail-open, air-operated, flow-modulating valves are located in each pump discharge line. These valves will normally be full open when the system is activated. The valves are provided to allow operator control of the emergency feedwater flow rates to the steam generators so that, in the long term, steam generator water levels can be restored and maintained in the narrow control range. For conservatism, it is assumed that no operator action can be taken for 30 min, and for this period it is assumed that these valves will be full open.

The EFWS can be used to supply feedwater to the steam generators during a plant startup when only small amounts of feedwater are required; however, the system is not normally used for this purpose. Other equipment in the main feedwater system design is provided to supply this reduced amount of feedwater during the plant startup and heatup.

The EFWS is not operated during normal plant operations but remains in a state of readiness to provide emergency feedwater to the steam generators in the event of transient or accident conditions. In the event of such occurrences, the emergency feedwater pumps are automatically started as follows:

- Low–low level in 2/4 level channels in any one steam generator: motor-driven pumps
- Low–low level in 2/4 level channels in any two steam generators: turbine-driven pumps
- Safety injection: motor-driven pumps

All valves in the system discharge path are open, so the automatic startup of the pumps will result in the immediate delivery of emergency feedwater into the steam generators. The system is designed to supply at least the minimum required flow, within 1 min of the actuation signal, to at least two effective steam generators (or one effective steam generator in the case of a two-loop plant), and to continue this delivery for an indefinite period without operator action. When operator action can be taken (after an assumed 30 min delay), the emergency feedwater flow is adjusted by positioning the flow-modulating valves to restore and maintain the steam generator water levels within the narrow control range.

With the reactor tripped, and with the EFWS supplying water to the steam generators at a rate equivalent to the rate at which steam is being removed to dissipate core decay heat and the heat input of one RCP (assumed to be operating), the plant is in a stable hot standby condition. The plant can be maintained in this condition for a period limited only by the amount of water in the primary and alternate emergency feedwater supplies. If the initiating event can be resolved, plant power operations can be resumed. Normal

feedwater flow to the steam generators by the main feedwater system is resumed and the emergency feedwater pumps are manually stopped. If the initiating event cannot be resolved, a plant cool down must be performed.

In this case, the EFWS continues to supply feedwater to the steam generators throughout the cool down until the primary system hot-leg temperature is reduced to 350°F (176.7°C). At this temperature, the RHRS is activated and the EFWS is secured. The RHR continues the cool down to cold shutdown conditions.

The EFWS can be used to supply feedwater to the steam generators for normal plant cool downs. To accomplish this, the pumps are manually started and the flow rates manually controlled by positioning the flow modulating valves. When the reactor coolant hot-leg temperature is reduced to 350°F (176.7°C) and the RHRS has been activated, the EFWS is secured. The EFWS is not normally used for plant cool down during shutdown because other design provisions contained in the main feedwater system should be used to supply the steam generators in this case.

### 2.10.7 Component Cooling Water System

The component cooling water system (CCWS) provides a continuous supply of cooling water to plant components which handle potentially radioactive fluids. In doing so, it forms an intermediate barrier between these potentially radioactive systems and the service water system, thereby reducing the possibility of discharging radioactivity to the environment. Component cooling water is supplied to NSSS as required by the following operations:

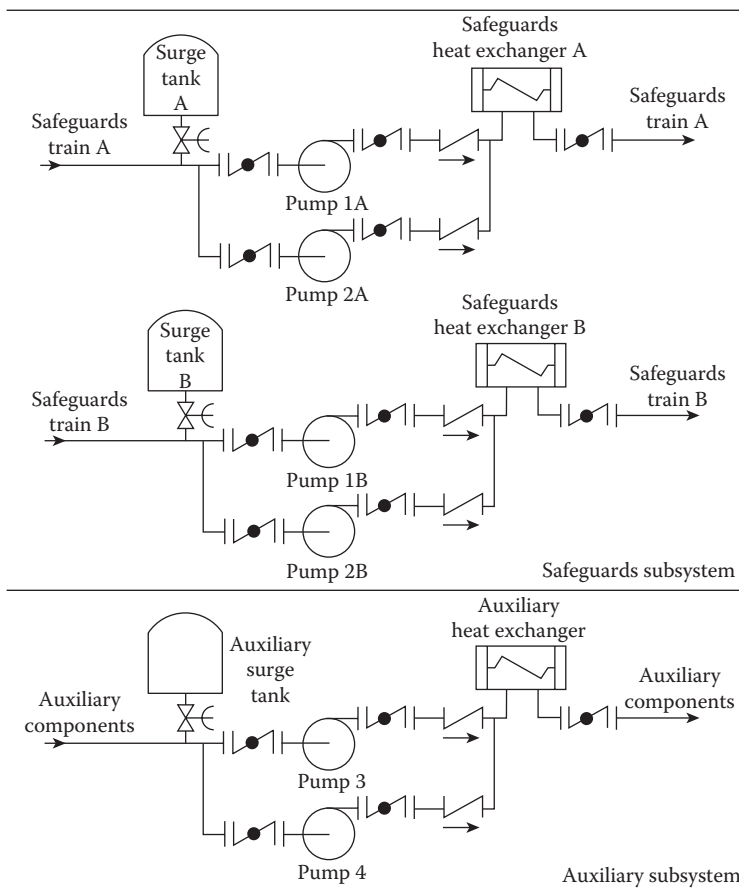
- Removal of heat from various components during normal power generation
- Removal of residual and sensible heat from the RCS through the RHRS during plant cool down
- Cooling of safeguards equipment following an accident

The CCWS is designed to provide cooling for the following components (The acronyms in the following list are for the systems of which these components are a part, e.g., CSS is containment spray system.):

- RCPs (RCS)
- Residual heat exchangers (RHRS)
- Residual heat removal pumps (RHRS)
- Letdown heat exchanger (CVCS)
- Excess letdown heat exchanger (CVCS)
- Seal water heat exchanger (CVCS)
- Centrifugal charging pumps (CVCS)
- Control rod-driven mechanism coil cooler (RCS)
- Refrigerated dryer
- Recycle evaporator package (BRS)
- Chiller unit (BTRS)
- Waste evaporator package(s) (WPS)
- Waste gas compressor package (WPS)

- Hydrogen monitor
- Reactor coolant drain tank heat exchanger (WPS)
- Spent fuel pit heat exchangers (SFPCS)
- Sample heat exchangers (SS)
- Gross failed fuel detector
- Containment spray pump heat exchangers (CSS)
- Containment fan coolers
- Service air compressor
- Instrument air compressor
- Positive displacement pump (CVCS)

The CCWS consists of a safeguards subsystem and an auxiliary subsystem (Figure 2.21). The CCWS safeguards subsystem is considered an engineered safeguards system because it is required to remove decay heat during post-accident and to provide cooling water to safeguard equipment. The subsystem is designed to retain total physical separation consisting



**FIGURE 2.21**  
Component cooling water system, flow diagram.

of two separate, independent safeguards trains with no cross connection between the two trains. Components of each safeguards train (pumps and heat exchanger) are housed in separate safeguards equipment cubicles. The plant separation facilitates compliance to standards in the area of fire protection, flooding prevention, and plant sabotage.

The safeguards subsystem is divided into two separate and redundant safeguards trains and provides cooling water to safeguards equipment such as the residual heat removal heat exchangers and pump, the RCP thermal barriers, the containment fan coolers, the containment spray pump heat exchangers, the spent fuel pit heat exchangers, and the positive displacement pump. Each safeguards train is sized to supply 100% post-accident component cooling flow requirements. Each train consists of two 100%-capacity component cooling pumps; one 100% component cooling heat exchanger; one component cooling surge tank; a chemical addition tank; cooling lines to the various safeguards components; and associated piping, valves, and instrumentation. This arrangement permits single failure criteria to be met with up to one pump per train removed from service for maintenance. The component cooling water flows from the pump, through the shell side of the component cooling heat exchangers, through the components being cooled, and back to the pump. The surge tank is connected to the suction side of the component cooling pumps.

The component cooling auxiliary subsystem provides cooling water to nonsafeguards components in the CVCS, BRS, waste processing systems, and other auxiliary subsystems. The auxiliary subsystem provides 100% component cooling for normal operation. It consists of two 100% component cooling pumps; one 100% component cooling heat exchanger; one component cooling surge tank; a chemical addition tank; and associated piping, valves, and instrumentation. The subsystem's flow path is similar to that of the safeguards subsystem. The surge tanks of safeguards and auxiliary subsystems provide like functions. They provide a surge volume to accommodate thermal expansion and contraction of system volume during transients and collect water that may leak into the system from components being cooled. The tanks also contain sufficient water volume to provide component cooling water until a design basis passive failure can be isolated.

Water chemistry control of the CCWS is accomplished by chemical additions to the chemical addition tank. A safety grade makeup source is provided by the demineralized water system or reactor makeup system (emergency makeup only) and delivered to the surge tank.

During normal full-power operation, one component cooling safeguards train and the component cooling auxiliary subsystem (each operating with one pump and one heat exchanger) are required to accommodate the heat removal loads. The safeguards train services the containment fan coolers, the RCP thermal barriers, and the spent fuel pit heat exchangers; the auxiliary train supplies component coolant to other components required during normal operation.

During plant normal cool down, the standby train of the safeguards subsystem is placed in operation at approximately 4 h after reactor shutdown. The safeguards subsystem provides component cooling water flow to the operating residual heat exchangers and pumps. The component cooling water inlet temperature to various components during normal cool down is permitted to increase to 120°F (49°C), but it must return to 105°F (40°C) after 4 h. Pump failure will not affect the time required for cool down because a standby pump in each train of the safeguards subsystem is provided. A spare pump is also available in the auxiliary subsystem. Failure of a safeguards train will not affect the plant's capability to remove decay heat; but cool down time will be extended. The CCWS pumps, heat exchangers of the safeguards subsystem and its associated piping, valves and instrumentations are seismically qualified as required for an engineered safeguards system.

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## 2.11 Containment Systems

Multiple barriers designed into the power plant provide containment of radioactive products at three fundamental levels:

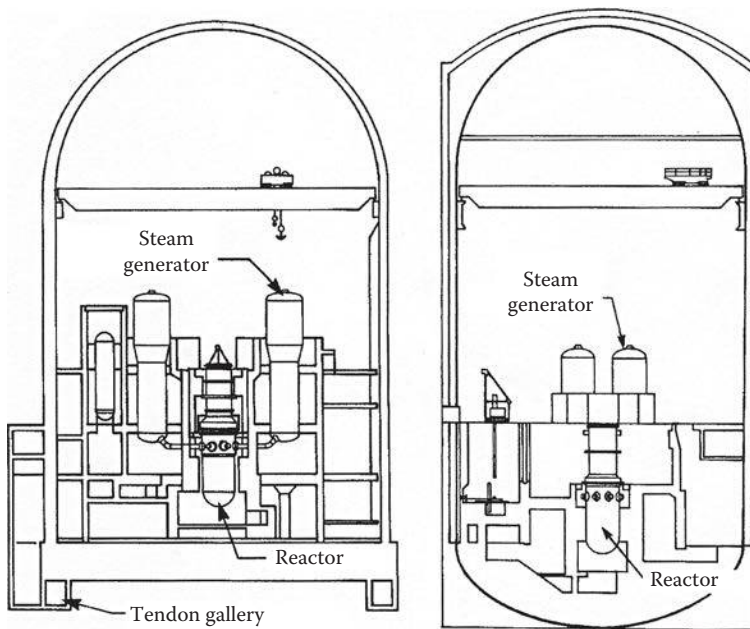
- Zircaloy tubes of the fuel rods
- RCS pressure boundaries
- Reactor containment vessel (building)

Diversity of concept, defense-in-depth, and redundancy of systems as it relates to protection and containment are discussed in other sections of this handbook. This subsection discusses the design of the reactor containment building (hereafter referred as “the containment”) that will contain and control any release of radioactivity to the environment under normal, upset, or emergency conditions. Also included are containment systems that will protect the integrity of the containment by reducing steam pressure and controlling hydrogen to avoid an explosive mixture. Ventilation systems provided with filters will reduce radioactivity in the containment atmosphere to permit safe access into the containment. The structure provides biological shielding for normal and accident situations.

Several types of containment structures have been designed and proposed to utilities. Those designs in prevalent use incorporate steel vaults or concrete vessels lined with steel plate. Steel vessels can be cylindrical or spherical in shape. Reinforced concrete vessels, which may in some cases be post-tensioned, are cylindrical with hemispherical domes. The type of structure chosen by a utility is dependent on plant layout, site characteristics, and relative costs of alternatives for a particular project. Typical containments are shown in Figure 2.22. A description is given of the prestressed concrete containment having a cylindrical shell, a hemispherical dome, and a flat base slab (with a pit). The inside face of the concrete shell, dome, and floor is steel plate lined to ensure a high degree of leak tightness. Other designs while varying in detail will meet the same functional requirements. The cylindrical shell is prestressed by a post-tensioning system consisting of horizontal and vertical tendons. The dome has a two-way post-tensioning system. There are three buttresses equally spaced around the containment. Hoop tendons are anchored at buttresses 240° apart, bypassing the intermediate buttress. Each successive hoop is progressively offset 120° from the one beneath it. (Another possible tendon arrangement includes U-tendons and hoops; a third system includes helicoidal tendons in opposite patterns.)

The foundation base slab is a concrete structure conventionally reinforced with high-strength reinforcing steel. A continuous access gallery is provided beneath the base slab for the installation and inspection of vertical tendons. The base liner, installed on top of the structural slab, is covered with concrete for post-tension. The containment completely encloses the entire reactor and RCS and ensures that an acceptable upper limit for leakage of radioactive materials to the environment would not be exceeded even if gross failure of the RCS were to occur. The approximate dimensions of the containment are 124 ft (37.8 m) inside diameter, 205 ft (62.5 m) inside height, 3–1/2 ft (1.07 m) wall thickness, and 2–1/2 ft (0.76 m) dome thickness. The internal net free volume approximates 2,000,000 ft<sup>3</sup> (56,600 m<sup>3</sup>).

The containment is designed for all credible conditions of loading, including normal loads, loads during LOCA, test loads, and loads due to adverse environmental conditions.



**FIGURE 2.22**  
Different types of containments, prestressed concrete on left and cylindrical steel on right.

The two critical loading conditions are those caused by the design basis accident (DBA) resulting from failure of the RCS and those caused by an earthquake.

Loading considerations are applied as detailed below.

### 2.11.1 DBA

The minimum design pressure and temperature of the containment are equal to the peak pressure and temperature occurring as the result of any rupture of the RCS up to and including the double-ended severance of a reactor coolant pipe. The supports for the RCS are designed to withstand the blowdown forces associated with the sudden severance of the reactor coolant piping so that its coincidental rupture with that of the steam or the feedwater systems is not considered credible. Transients resulting from the DBA serve as the basis for a containment design pressure of 60 psig (4.46 bar). Transients resulting from other accidents could be controlling for specific compartments of the containment.

### 2.11.2 Thermal Loads

The variation of temperature with time and the expansion of the liner plate are considered in designing for the thermal stresses associated with the DBA.

### 2.11.3 Dead Loads

Dead loads consist of the weight of the concrete wall, dome, base slab, internal concrete and permanent equipment, machinery, components, and the like.



#### 2.11.4 Live Loads

Live loads consist of all loads except dead, accident, seismic, flood, and wind and include snow loads on the domed roof of the containment. Live loads are assumed for the design of internal slabs consistent with the intended use of the slabs.

#### 2.11.5 Earthquake Loads

Earthquake loading is predicated upon a design earthquake with a ground acceleration equal to the operating basis earthquake (OBE) for the site selected. In addition, a maximum hypothetical earthquake having a ground acceleration equal to the design safe shutdown earthquake (SSE) for the site selected is used to check the design and ensure no loss of function. A vertical component having a magnitude equal to the horizontal component is applied simultaneously. A three-dimensional dynamic analysis is used to arrive at equivalent static loads for design. Soil-structure interactions are included in the dynamic model.

#### 2.11.6 Wind Forces

Wind loading results from site conditions, including consideration of hurricane winds. This wind loading is considered for the design of all structures. However, wind loads are not applied simultaneously with seismic loads.

#### 2.11.7 Hydrostatic Loads

Uplift forces created by hydrostatic pressure are included in the design of all structures.

#### 2.11.8 External Pressure Loads

External pressure loading with a differential of 2-1/2 psi (0.17 bar) from outside to inside are considered. The external design pressure is also adequate to permit the reactor building to be cooled from an initial maximum operating condition of 120°F (49°C) to an internal temperature during shutdown of 50°F (10°C) (winter).

#### 2.11.9 Prestressing Loads

Prestressing loads allow for slip at anchorage, elastic shortening of concrete, creep of concrete, shrinkage of concrete, relaxation of steel stress, and frictional losses in the tendon ducts. Load factors are in accordance with the ACI/ASME Code.

#### 2.11.10 Containment Design Criteria

The safety of the structure under extraordinary circumstances and the performance of the containment at various loading stages are the primary considerations in establishing the structural design criteria. The two basic criteria are

- Integrity of the containment liner is guaranteed under all credible loading conditions.
- Structure has a low-strain elastic response such that its behavior is predictable under all design loadings.

The strength of the containment at working stress and overall yielding is compared with the allowable values under the various loading combinations to ensure safety. To ensure proper performance, the analysis and design of the containment is carried out with consideration for strength, the nature and the amount of cracking, the magnitude of deformation, and the extent of corrosion. The structure is designed to meet performance and strength requirements under the following conditions:

- Before prestressing
- At transfer of prestress
- Under sustained prestress
- At design loads
- At factored loads

The base slab acts primarily in bending rather than membrane stress and, therefore, it is not prestressed.

#### 2.11.11 Design Method

The containment shell is analyzed for individual and various combinations of loading cases of dead load, live load, prestress, temperature, and pressure. The design output includes direct stresses, shear stresses, principal stresses, and displacements of each nodal point. Stress plots which show total stresses resulting from appropriate combinations of loading cases are made and areas of high stress identified. If necessary, the modulus of elasticity is corrected to account for the nonlinear stress–strain relationship at high stresses. Stresses are then recomputed if a sufficient number of areas requiring attention exist.

#### 2.11.12 Containment Liner Criteria

To meet the specified leak rate under accident conditions, the containment liner satisfies the following criteria:

- Containment liner is protected against damage by missiles
- Containment liner strains are limited to allowable values that have been shown to result in leak tight vessels or pressure piping
- Containment liner is prevented from developing significant distortion
- All discontinuities and openings are anchored to accommodate the forces exerted by the restrained containment liner, and careful attention is paid to details at corners and connections to minimize the effects of discontinuities. The following fatigue loads are considered in the design of the containment liner:
  - Thermal cycling due to annual outdoor temperature variations. Daily temperature variations do not penetrate a significant distance into the concrete shell to appreciably change the average temperature of the shell relative to the containment liner. The number of cycles for this loading will be 40 cycles for the plant life of 40 years.
  - Thermal cycling due to containment interior temperature varying during the startup and shutdown of the reactor system. The number of cycles for this loading is assumed to be 500.

- Thermal cycling due to DBA is assumed to be one cycle. Thermal load cycles in the piping systems are somewhat isolated from the containment liner penetrations by the concentric sleeves between the pipe and the liner. All penetrations are verified for a conservative number of cycles to be expected during the plant life. The containment liner as well as any carbon steel surface exposed to the containment atmospheres is coated with a coating system qualified to resist accident conditions without peeling, scaling, or blistering.

### 2.11.13 Equipment and Personnel Access Hatches

An equipment hatch 21 ft (6.4 m) in diameter is provided for access to the containment. It is fabricated from welded steel and furnished with a double gasketed flange and bolted dished door. Equipment up to and including the size of the steam generators can be transferred into and out of the containment through this hatch. Two personnel locks are provided. One of these is for emergency egress only. Each personnel lock is a double door, welded steel assembly. Quick-acting valves connect the personnel lock with the interior and exterior of the containment for the purpose of equalizing pressure in the two systems when entering or leaving the containment. The two doors in each personnel lock are interlocked to ensure that one door is completely closed before the opposite door can be opened.

Remote indicating lights and annunciators situated in the control room indicate the door operational status. Provision is made to permit bypassing the door interlocking system to allow doors to be left open during plant cold shutdown. Each door hinge is designed to be capable of independent, three-dimensional adjustment to assist proper seating. An emergency lighting and communication system operating from an external emergency supply is provided in the lock interior.

### 2.11.14 Special Penetrations

*Fuel transfer penetration:* A fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the reactor containment building and the spent fuel pool in the fuel handling building. The penetration consists of a 20-in. (51 cm) stainless steel pipe installed inside a casing pipe. The inner pipe acts as the transfer tube and is fitted with a double gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pit. The casing pipe is provided with expansion joints and is connected to the containment liner. This arrangement prevents leakage in the event of an accident.

*Containment supply and exhaust purge ducts:* The ventilation system purge duct is equipped with two tight seating valves to be used for isolation purposes. The valves are remotely operated for containment purging.

### 2.11.15 Containment Isolation System

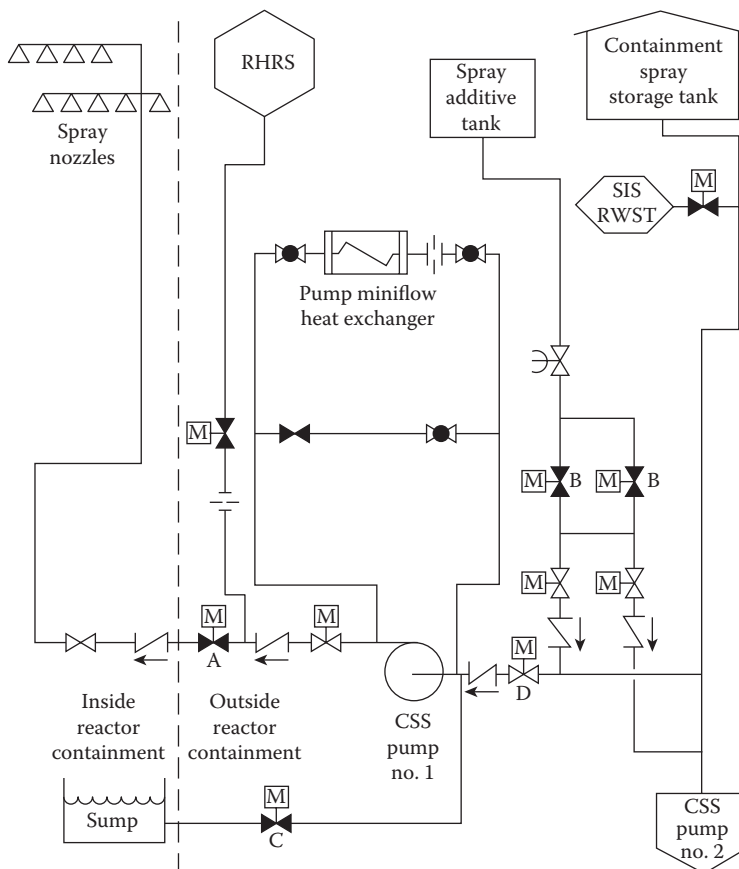
The containment isolation system (CIS) provides the means of isolating the various fluid systems passing through the containment walls as required to prevent the release of radioactivity to the outside environment. Design bases of the CIS consider several factors. Subsequent to an accident which may release radionuclides within the containment, there must be a barrier in all pipes or ducts that penetrate the containment. Leakage from the containment through these pipes or ducts which penetrate the containment is minimized by a double barrier. This double barrier ensures that failure of a single active

component along a leakage path will not result in loss of ability to isolate the containment. A barrier may be a valve, a blind flange (two barriers by use of a double gasket), or a closed piping system or vessel. The barriers must be missile protected, Seismic Class 1 and designed for a pressure equal to or greater than the design pressure of the containment. The isolation barriers are located as close to the containment penetration as practicable.

### 2.11.16 Containment Spray System

The containment spray system (CSS) (Figure 2.23) is an engineered safeguards designed to limit the peak pressure in the reactor containment building to a pressure less than the containment design pressure, in the event of a LOCA or a steam break accident inside the containment. The system also acts to remove airborne fission products (principally iodine) from the containment atmosphere should they be present due to a fuel cladding break.

The CSS achieves the above objectives by spraying a sodium hydroxide solution of borated water throughout a large volume portion of the containment atmosphere. The system consists of a containment spray storage tank (CSST), a spray additive tank (SAT), two



**FIGURE 2.23**  
Containment spray system (one of two trains) flow diagram.

pumps, two heat exchangers, and a set of spray ring headers located in the upper dome of the containment and having nozzles designed to provide adequate containment spray coverage. A pump mini-flow line, containing a heat exchanger and flow limiting orifice, and a test loop with a throttling valve are connected in parallel across each pump. The test line is open only when testing the pump at various flows. The pumps are normally aligned to take suction from the CSST and the SAT, which drain down together to provide the correct spray pH. Each one of these subsystems is also independently capable of delivering the necessary flow to limit containment pressure in conjunction with the reactor containment fan cooler (RCFC) system.

#### 2.11.17 Initial Injection Mode

Containment spray is initiated automatically in response to a containment high pressure signal. This signal starts the spray pumps, opens the spray header isolation valves "A," and opens the SAT isolation valves "B." Suction is provided to the spray pump from the CSST, with the SAT providing sodium hydroxide solution. The SAT drains into each spray pump's suction header from the CSST; this mixing results in a solution pH that is conducive to the entrainment of iodine (airborne fission product). The spray pumps deliver this solution to the ring headers, and the solution is dispersed throughout the containment atmosphere by the spray nozzles. As this water comes in contact with the steam in the containment atmosphere, the steam is condensed and falls to the containment floor as drops of a borated sodium hydroxide solution. Also, as the spray is introduced to the containment atmosphere, the solution entrains airborne iodine particles that it contacts; these are then carried to the floor with the spray droplets. The storage tanks are sized to provide approximately 30 min of spray with spray additive, and when the CSST is drained to the "Lo-Lo" setpoint, the spray pumps are manually switched to the recirculation mode of spray operation. An additional function which is performed by the CSS during its normal operation is the pH adjustment of the containment sump. The sodium hydroxide (spray additive) delivered through the injection phase is sufficient to raise the long-term sump pH to a level adequate for inhibiting chloride stress corrosion/cracking of stainless steel components, and for keeping the iodine entrained in the water.

The recirculation mode of the CSS will be initiated manually by the operator. To accomplish this, the sump isolation valves "C" are opened, and the spray pump suction isolation valves "D" are closed. The pumps will then deliver spray from the sump to the spray headers. Spray additive is not injected during this mode.

#### 2.11.18 RCFC System

The RCFC system is designed to remove heat from the containment building during normal operation and in the event of a LOCA. The RCFC is an engineered safeguards system. In a containment, the system includes four fan units that operate in parallel. A minimum of two units must function to satisfy design requirements for normal operation and post-accident operation. During normal operation, air is drawn from the upper part of the containment, through the return air ductwork, and continues through the normal flow inlet damper into the roughing filter plenum. This bypasses the HEPA filters and other components upstream of the roughing filter, which is mainly for the protection of the fan from foreign objects and minimizing the dust in the containment. The air continues its flow through the cooling coils and discharged by the fan into the ventilation system distribution ductwork.

During the post-LOCA operating mode, the air flow is routed from the return air ductwork, through the emergency flow inlet damper, and into the filtration package plenum. Here, the air flows through moisture separators, HEPA filters, and discharges through the emergency flow outlet dampers into the roughing filter plenum and then into the cooling coils. When a LOCA is sensed, the fan motors are automatically switched to low speed to provide proper flow of the steam–air mixture. A gravity-actuated backdraft damper is installed in the ventilation system discharge ductwork of each fan. These dampers serve to isolate units from the ventilation system when the fan is not in use and to protect each unit from damage due to reverse flow during a LOCA pressure transient. The cooling coils remove heat from the air with the fan providing the required air flow rates. Cooling water is supplied by the essential service water system. Drain troughs and piping are provided to remove condensate humidity from the cooling coils. The drain piping is routed to the containment sump.

### 2.11.19 Hydrogen Control in Containment

Under accident conditions, coolant radioactivity and the Zirconium water reaction can result in the release of hydrogen gas to the containment atmosphere. Eventually the gas could accumulate to an explosive hydrogen/oxygen concentration. To control this potential risk the reactor containment is provided with

- A hydrogen detection system that can detect and report in a continuous manner the hydrogen content in the containment atmosphere. Alarms at proper setpoints are provided.
- A containment ventilation system that circulates the containment atmosphere to remove pockets of high hydrogen concentration after an accident.
- Hydrogen recombiners that burn the free hydrogen. The hydrogen/oxygen combination results in water.

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## 2.12 Instrumentation

This is an extremely detailed subject involving not only detection systems for radiation (or its surrogates: electronic currents, radioactive isotopes, etc.) but also control theory, computer representations of data and their interpretation, and automatic systems that take over when events happen too fast for human intervention.

The instrumentation and control systems can be viewed as the “central nervous system” of the plant and consist of the following major systems:

- *Nuclear instrumentation system*—Provides continuous indications of the reactor core power level from shutdown to full power.
- *In-core instrumentation system*—Senses the distribution of the nuclear flux within the core.
- *Digital rod position indication system*—Detects the position of the control rods in the reactor core.

- *Process instrumentation system*—Senses the state of the plant, when used together with the nuclear instrumentation, in-core instrumentation, and digital rod position indication system.
- *Nuclear steam supply control system*—Implements the operator's control decisions and automatically changes the plant to and maintains it at selected operating states.
- *Reactor protection system*—Protects the reactor core and the NSSS by monitoring operating parameters and initiating safeguard actions on the detection of abnormal conditions.
- *The control room*—Provides information to the operator to enable him/her to comprehend the plant's state and to make and implement operating decisions. Increasingly, ergonomic considerations have come to dominate the design of the layout: immediately in front of the operator are the controls and read-outs for normal operation. Up a little higher on the panel are instruments that show the effect of control manipulations, together with setpoint limits on key variables. Above that display are the alarms and signals of upset and accident conditions. All key instrumentation for each panel is within sight of the operator. Panels are arranged from the reactor and its auxiliaries, to the secondary systems, to the tertiary plant systems, and finally the electrical systems and their adjuncts. Arrangements are logical for normal power operation, shutdown and maintenance configurations, and emergency operations.
- *Plant computer system*—Provides computational, data processing, and data presentation services for the plant. Flow maps and instrumentation diagrams may be called-up and data-logged to allow sequence analysis after events. Computers are especially valuable in the training center to model the reactor and all other systems, and to simulate upset and accident conditions that could not be conducted on the actual plant. This tool is also important in the development of emergency procedures.

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## 2.13 Fuel Handling

Systems for fuel handling can vary widely. In reactor sites on which multiple plants are located, a single storage pool for spent fuel may be most convenient. Where long-term storage is necessary, a large pool and dry storage pads for shielded containers may be used for spent fuel. After the radioactivity of the fuel has died down for several years, it may be stored dry in vertical concrete vaults that allow air circulation while preventing people and small animals from being exposed.

### 2.13.1 Spent Fuel Handling

Spent fuel is handled underwater from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site, or placed in dry storage. The water provides an effective, economical, and transparent radiation shield as well as a reliable medium for decay heat removal. Boric acid is added to the water to ensure subcritical conditions during refueling. The fuel handling facilities are generally divided into two areas:

1. The refueling cavity and fuel transfer canal, which are flooded only during the refueling shutdown

2. The fuel storage pit, which is kept full of water and is always accessible to operating personnel

These two areas are connected by the fuel transfer tube through which an underwater conveyor or fuel transfer system carries the new fuel into the reactor containment and spent fuel into the fuel handling building.

In the refueling cavity, fuel is removed from the reactor vessel by a refueling machine, transferred through the water, and placed in the fuel transfer system. In the fuel handling building, the fuel is removed from the transfer carriage and placed in storage racks in the fuel storage pit by using a manually operated spent fuel assembly handling tool suspended from an overhead bridge crane. After a sufficient decay period, the spent fuel can be removed from the fuel racks and loaded into a shipping cask for removal from pool into dry storage, or shipped from the site. The shipping cask is inspected for internal or external contamination, dirt, and so on before being lowered onto a special pad at the end of the pool near the truck door of the fuel handling building. The cask lid is removed and the spent fuel assemblies are placed therein, the lid replaced, the cask raised and drained, the lid fastened down, and the cask is loaded onto the truck.

### 2.13.2 New Fuel Handling

New fuel assemblies normally arrive at the site shortly before refueling is to commence. Following depressurization of the container and site receipt inspection to check for any possible shipping distortion or damage, new assemblies are attached to the “new” (short-handled) fuel handling tool and are lowered into the fuel storage pit by means of the new fuel elevator and are placed underwater in the storage racks. During the refueling operation, the new assemblies are transported one at a time from their storage locations in the pit to the fuel transfer system by means of the fuel handling machine and the “spent” (long-handled) fuel handling tool.

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## 2.14 Waste Handling

Waste processing systems, which can vary greatly from plant to plant, process liquid, gaseous, and solid plant effluents during power operation and plant shutdowns. The systems consist of the liquid waste processing system, the waste gas processing system, and the solid waste processing systems. The liquid waste processing system is designed to collect, process, monitor, and recycle for reuse the liquid waste effluents generated during various plant operations. The waste gas processing system stores waste gases for fission product decay and eventual release. Wastes which cannot be recycled and must be disposed of safely are volume reduced and packaged for disposal by the solid waste systems.

### 2.14.1 Liquid Waste Processing

The liquid waste processing system is provided for use in the processing and handling of radioactive wastes generated during various modes of plant operation. The system is designed to receive, segregate, process, monitor, and recycle for reuse all primary



system waste effluents. The system is designed so that tritium-containing water can be segregated from nontritium-containing water and includes a separate laundry waste treatment system, thereby allowing for easy tritium control. In addition, provisions are made to handle spent regenerant chemicals from condensate polishers and waste from decontamination of spent fuel shipping casks. The system consists of several waste holdup and collection tanks, corresponding pumps, waste evaporators, demineralizers, filters, monitors, and associated piping. Liquid wastes from floor drains, laundry, hot showers, laboratory rinses, equipment drains, and the like are collected in their respective holdup tanks. After collection, some wastes are processed for disposal; most are processed through the waste evaporators or the reverse-osmosis package (separation of pure water from dissolved salts). Sample analysis is performed and the wastes are either recycled for reuse or disposed of safely.

The system design incorporates features specifically aimed at minimizing the environmental impact of plant operation. In addition, the design provides adequate processing capacity to accommodate unforeseen occurrences of high liquid waste leakage. Sufficient capacity, redundancy, and flexibility provide a wide range of operability.

### **2.14.2 Gaseous Waste Processing**

The gaseous waste processing system receives noble fission gases which have been stripped from the reactor coolant through the use of hydrogen gas as a carrier. The stripping of the fission gases by hydrogen reduces the fission gas concentration in the reactor coolant to a low residual level. This minimizes the release of radioactive gases during maintenance operations on the RCS or through unavoidable equipment leaks in the RCS. The system consists of a waste gas dryer, charcoal adsorption beds, guard beds, a charcoal fines filter, a surge tank, and recycle line compressors. The hydrogen carrier gas from the VCT (part of the CVCS) first enters the refrigerated waste gas dryer which cools the hydrogen purge stream and condenses and removes the water vapor. The dried gas then flows to the guard beds which protect the charcoal adsorption beds from water contamination. The flow is then routed through the charcoal adsorption tanks where the noble fission gases contained in the waste stream are absorbed. Xenon-133 is delayed for 60 days, after which time the concentration of xenon in the exiting hydrogen stream is negligible. The system is designed to also delay krypton-85 for 3 days. The hydrogen carrier gas experiences no delay; it passes through the charcoal beds to the plant vent. The recycle line compressors provide the system with the capability to process gases from other areas of the plant by routing the gases to the VCT for processing. In addition, the compressors may be used for hydrogen gas recycle where its use is economically justified.

### **2.14.3 Solid Waste Processing**

The solid waste processing systems include the radwaste volume reduction/solidification system and the radwaste incinerator. The systems reduce in volume and solidify low-level radioactive plant wastes to prepare them for safe storage and/or disposal. The radwaste volume reduction/solidification system employs a vacuum-cooled crystallization process to effect volume reduction, coupled with high speed, higher shear mixing of the waste with cement to achieve solidification. For combustible plant wastes, the radwaste incinerator utilizes a controlled air incineration process.

#### 2.14.4 Radwaste Volume Reduction

Volume reduction of concentrated evaporator bottoms, which may include boric acid wastes, laundry wastes, chemical wastes, and other floor drain wastes, is accomplished in the radwaste volume reduction system. The major components of the system are the crystallizer chamber and recirculation system, condenser, and vacuum pump system. The crystallizer chamber consists of a conical tank and an inner circular baffle to separate solid crystals from a clear recycle stream. Combustible wastes such as clothing, filter cartridges, and wood are volume-reduced in the radwaste incinerator. Solidification of volume-reduced wastes and other low-level radioactive wastes, such as spent resins and contaminated tools, is performed in the cement solidification system. The major components of the cement solidification system include the high shear radwaste mixer, waste dispensing system, flush water recycle steam, cement storage and feed system, and the container handling system.

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### 2.15 Advanced Passive Reactor

The year 2006 was a seminal one for the design of a new generation of PWRs. The Westinghouse design of the AP1000 was one of the first certified. It was initially approved by the Nuclear Regulatory Commission (NRC) for a lower power model, the AP600, when it seemed that revitalization of the nuclear industry would center around smaller units that were expected to come online sooner than the larger-capacity plants. They would also appeal to smaller utilities who would not want too large a share of their total capacity dominated by any one plant. In the same year that new designs were being announced, many mergers were taking place that changed the vendor array available to the world's utilities. These changes were discussed earlier in this section. Discussion of these other new designs by other organizations is contained in the handbook sections after this one. At the same time as new vendor organizations were developing, several utilities formed consortia to operate the existing and future nuclear plants. These consortia are not under the control of state public utility commissions (PUCs). Consequently, individual utilities under PUC control do not risk having their investment in nuclear power arbitrarily declared "imprudent investment," thus disallowing the costs to be recovered in the rate base. The consortia are under the control of the NRC, as well as the Federal Energy Regulatory Commission (FERC) because electric power is often sold across state lines. Increasing numbers of interties between systems and "wheeling" of power across the country have greatly increased the reliability of the nation's electrical distribution system. Other innovations on the horizon include cryogenic (low resistance losses) and direct current (DC) long-distance transmission lines (possibly underground) that produce a negligible radiation field.

#### 2.15.1 New PWR Designs

For the convenience of the reader, the AP600 and AP1000 are compared with the largest conventional reactor design that preceded them. The comparison of key parameters is given in Table 2.4. Following that is a detailed discussion of changes in the design from that described above in earlier subsections. Much has been done to improve neutron

**TABLE 2.4**

Reactor Design Comparison Table

Thermal and Hydraulic Design Parameters	AP1000	AP600	Typical XL Plant
Reactor core heat output (MWt)	3,400	1,933	3,800
Reactor core heat output (10 <sup>6</sup> Btu/h)	11,601	6596	12,969
Heat generated in fuel (%)	97.4	97.4	97.4
System pressure nominal (psia)	2,250	2,250	2,250
System pressure, minimum steady-state (psia)	2,190	2,200	2,204
Minimum departure from nuclear boiling (DNBR)			
Typical flow channel	>1.25 <sup>a</sup> , >1.22 <sup>a</sup>	>1.23	>1.26
Thimble (cold wall) flow channel	>1.25 <sup>a</sup> , >1.21 <sup>a</sup>	>1.22	>1.24
Departure from nucleate boiling (DNB) correlation <sup>b</sup>	WRB-2M <sup>b</sup>	WRB-2	WRB-1 <sup>c</sup>
<i>Coolant Flow<sup>d</sup></i>			
Total vessel thermal design flow rate (10 <sup>6</sup> lbm/h)	113.5	72.9	145.0
Effective flow rate for heat transfer (10 <sup>6</sup> lbm/h)	106.8	66.3	132.7
Effective flow area for heat transfer (ft <sup>2</sup> )	41.5	38.5	51.1
Average velocity along fuel rods (ft/s)	15.9	10.6	16.6
Average mass velocity (10 <sup>6</sup> lbm/h-ft <sup>2</sup> )	2.41	1.72	2.60
<i>Coolant Temperature<sup>e</sup></i>			
Nominal inlet (°F)	535.0	532.8	561.2
Average rise in vessel (°F)	77.2	69.6	63.6
Average rise in core (°F)	81.4	75.8	68.7
Average in core (°F)	578.1	572.6	597.8
Average in vessel (°F)	573.6	567.6	593.0
<i>Heat Transfer</i>			
Active heat transfer surface area (ft <sup>2</sup> )	56,700	44,884	69,700
Average heat flux (BTU/h-ft <sup>2</sup> )	199,300	143,000	181,200
Maximum heat flux for normal operation (BTU/h-ft <sup>2</sup> ) <sup>f</sup>	518,200	372,226	489,200
Average linear power (kW/ft) <sup>g</sup>	5.72	4.11	5.20
Peak linear power for normal operation (kW/ft) <sup>g,s</sup>	14.9	10.7	14.0
Peak linear power (kW/ft) <sup>t,h</sup> (resulting from overpower transients/operator errors, assuming a maximum overpower of 118%)	□22.45	22.5	□22.45
Heat flux hot channel factor (F <sub>Q</sub> )	2.60	2.60	2.70
Peak fuel centerline temperature (°F) (for prevention of centerline melt)	4,700	4,700	4,700
Fuel assembly design	17 × 17 XL Robust fuel	17 × 17	17 × 17 XL Robust fuel/no IFM
Number of fuel assemblies	157	145	193
Uranium dioxide rods per assembly	264	264	264
Rod Pitch (in.)	0.496	0.496	0.496
Overall dimensions (in.)	8.426 × 8.426	8.426 × 8.426	8.426 × 8.426
Fuel weight as uranium dioxide (lb)	211,588	167,360	261,000
Clad weight (lb)	43,105	35,555	63,200
Number of grids per assembly			
Top and bottom (Ni–Cr–Fe Alloy 718)	2 <sup>i</sup>	2 <sup>i</sup>	2
Intermediate	8 ZIRLO™	7 Zircaloy-4 or 7 ZIRLO™	8 ZIRLO™

(Continued)

TABLE 2.4 (Continued)

Reactor Design Comparison Table

Thermal and Hydraulic Design Parameters	AP1000	AP600	Typical XL Plant
Intermediate flow mixing	4 ZIRLO™	4 Zircaloy-4 or 5 ZIRLO™	0
Loading technique, first cycle	Three region nonuniform	Three region nonuniform	Three region nonuniform
<i>Fuel Rods</i>			
Number	41,488	38,280	50,952
Outside diameter (in.)	0.374	0.374	0.374
Diametral gap (non-IFBA) (in.)	0.0065	0.0065	0.0065
Clad thickness (in.)	0.0225	0.0225	0.0225
Clad material	ZIRLO™	Zircaloy-4 or ZIRLO™	Zircaloy-4/ ZIRLO™
<i>Fuel Pellets</i>			
Material	UO <sub>2</sub> sintered	UO <sub>2</sub> sintered	UO <sub>2</sub> sintered
Density (% of theoretical)	95.5	95	95
Diameter (in.)	0.3225	0.3225	0.3225
Length (in.)	0.387	0.387	0.387
<i>Neutron Absorber</i>			
RCCA	24 Ag–In–Cd rodlets	24 Ag–In–Cd rodlets	24 Hafnium or Ag–In–Cd
GRCA	20,304 SS rodlets 4 Ag–In–Cd rodlets	20,304 SS rodlets 4 Ag–In–Cd rodlets	
Cladding material	Type 304 SS, cold-worked	Type 304 SS, cold-worked	Type 304 SS, cold-worked
Clad thickness (Ag–In–Cd)	0.0185	0.0185	0.0185
Number of clusters	53 RCCAs 16 GRCA	45 RCCAs 16 GRCA	57 RCCAs 0 GRCA
<i>Core Structure</i>			
Core barrel, ID/OD (in.)	133.75/137.75	133.75/137.75	148.0/152.5
Thermal shield	None	None	Neutron panel
Baffle thickness (in.)	Core shroud	Radial reflector	0.875
<i>Structure Characteristics</i>			
Core diameter, equivalent (in.)	119.7	115.0	132.7
Core height, cold, active fuel (in.)	168.0	144.0	168.0
<i>Fuel Enrichment First Cycle (Weight Percent)</i>			
Region 1	2.35	1.90	Typical
Region 2	3.40	2.80	3.8–4.4
Region 3	4.45	3.70	(5.0 max)

<sup>a</sup> 1.25 applies to core and axial offset limits; 1.22 and 1.21 apply to all other RTDP transients.

<sup>b</sup> See Section 4.4.2.2.1 of reference AP-1000 design control document for the use of the W-3, WRB-2, and WRB-2M correlations.

<sup>c</sup> WRB-2M will be used in future reloads.

<sup>d</sup> Flow rates and temperatures are based on 10% steam generator tube plugging for the AP600 and AP1000 designs.

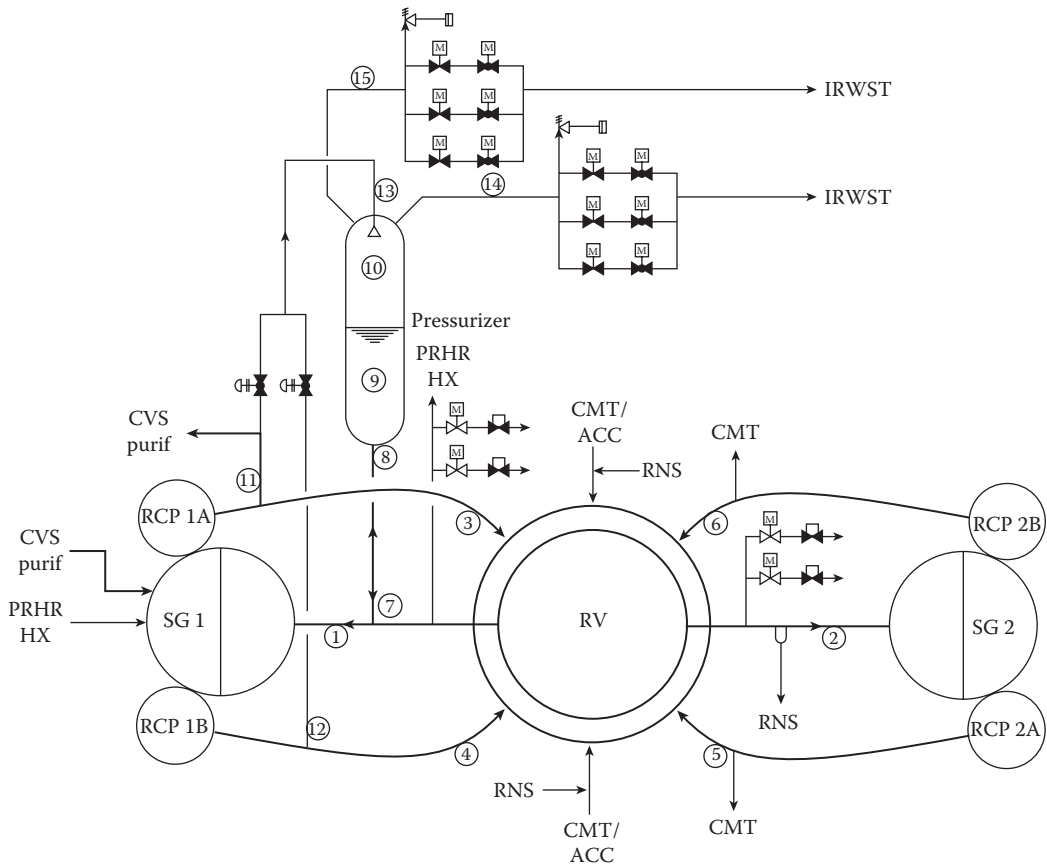
<sup>e</sup> Coolant temperatures based on thermal design flow (for AP600 and AP1000).

<sup>f</sup> Based on  $F_Q$  of 2.60 for AP600 and AP1000.

<sup>g</sup> Based on densified active fuel length.

<sup>h</sup> See Section 4.3.2.2.6 of reference AP1000 design control document.

<sup>i</sup> The top grid may be fabricated of either nickel–chromium–iron Alloy 718 or ZIRLO™.



**FIGURE 2.24**  
AP1000 reactor coolant system schematic flow diagram.

economy, system and component reliability. Also, equipment design and layout have been reworked to reduce cost and simplify plant operation. Finally, dramatic changes have been made in safeguards systems to provide for passive measures that would operate after a severe accident, assuming no operator action is possible in the containment vessel for a considerable period of time.

Figure 2.24 is a schematic of the RCS. Table 2.5 provides the principal pressures, temperatures, and flow rates of the system at the locations noted in Figure 2.24 under normal steady-state, full-power operating conditions. These parameters are based on the best-estimate flow at the pump discharge. Table 2.6 contains a summary of nominal system design and operating parameters under normal steady-state, full-power operating conditions. These parameters are based on the best-estimate conditions at nominal full power. The RCS volume under these conditions is also provided. The RCS consists of two heat-transfer circuits, each with a steam generator, two RCPs, and a single hot leg and two cold legs for circulating reactor coolant. In addition, the system includes the pressurizer, interconnecting piping, valves, and instrumentation for operational control and safeguards actuation. All RCS equipment is located in the reactor containment. During operation, the RCPs circulate pressurized water through the reactor vessel than the steam generators. Water serves

TABLE 2.5

Principal System Pressures, Temperatures, and Flow Rates (Nominal Steady-State, Full Power Operating Conditions) Location Numbers as Identified on Figure 2.24

Location	Description	Field	Pressure (psig)	Nominal Temp (°F)	Flow <sup>a</sup> (gpm)
1	Hot leg 1	Reactor coolant	2248	610	177,645
2	Hot leg 2	Reactor coolant	2248	610	177,645
3	Cold leg 1A	Reactor coolant	2310	537.2	78,750
4	Cold leg 1B	Reactor coolant	2310	537.2	78,750
5	Cold leg 2A	Reactor coolant	2310	537.2	78,750
6	Cold leg 2B	Reactor coolant	2310	537.2	78,750
7	Surge line inlet	Reactor coolant	2248	610	–
8	Pressurizer inlet	Reactor coolant	2241	653.0	–
9	Pressurizer liquid	Reactor coolant	2235	653.0	–
10	Pressurizer steam	Steam	2335	653.0	–
11	Pressurizer spray 1A	Reactor coolant	2310	537.2	1–2
12	Pressurizer spray 1B	Reactor coolant	2310	537.2	1–2
13	Common spray Line	Reactor coolant	2310	537.2	2–4
14	ADS valve inlet	Steam	2235	653.0	–
15	ADS valve inlet	Steam	2235	653.0	–

<sup>a</sup> At the conditions specified.

as a coolant, moderator, and solvent for boric acid (chemical shim control) and is heated as it passes through the core. It is transported to the steam generators where the heat is transferred to the steam system. Then it is returned to the reactor vessel by the pumps to repeat the process like all previous designs. Spring-loaded safety valves are installed above and connected to the pressurizer to provide overpressure protection for the RCPs. These valves discharge into the containment atmosphere. Three stages of RCS automatic depressurization valves are also connected to the pressurizer. These valves discharge steam and water through spargers to the in-containment RWST (IRWST) of the passive core cooling system (PXS). This is an entirely new system. On older designs, the RWST was located outside of the containment. Most (initially all) of the steam and water discharged to the spargers is condensed and cooled by mixing with the water in the tank. The fourth-stage automatic depressurization valves are connected by two redundant paths to each reactor coolant loop hot leg and discharge directly to the containment atmosphere. The RCS is also served by several auxiliary systems, including the CVCS, PXS, the normal residual heat removal system (RNS), the steam generator system (SGS), the primary sampling system (PSS), the liquid radwaste system (WLS), and the CCWS. With the exception of the PXS, these connections are similar to previous designs. The safeguards system in previous designs was called the SIS.

The design of the AP1000 reactor vessel closely matches the existing vessel designs of Westinghouse three-loop plants. New features have been incorporated without departing from the proven features of existing vessel designs. The vessel has inlet and outlet nozzles positioned in two horizontal planes between the upper head flange and the top of the core. The nozzles are located in this configuration to provide an acceptable cross-flow velocity in the vessel outlet region and to facilitate optimum layout of the RCS

**TABLE 2.6**

Nominal System Design and Operating Parameters for the AP1000

<i>General</i>	
Plant design objective (years)	60
NSSS power (MWt)	3415
Reactor coolant pressure (psia)	2250
Reactor coolant liquid volume at power conditions (including 1000 ft <sup>3</sup> pressurizer liquid) (ft <sup>3</sup> )	9600
<i>Loops</i>	
Number of cold legs	4
Number of hot legs	2
Hot leg ID (in.)	31
Cold leg ID (in.)	22
<i>Reactor Coolant Pumps</i>	
Type of reactor coolant pumps	Canned-motor
Number of reactor coolant pumps	4
Nameplate motor rating (hp)	7000
Effective pump power to coolant (MWt)	15
<i>Pressurizer</i>	
Number of units	1
Total volume (ft <sup>3</sup> )	2100
Water volume (ft <sup>3</sup> )	1000
Spray capacity (gpm)	500
Inside diameter (in.)	90
Height (in.)	607
<i>Steam Generator</i>	
Steam generator power (MWt/unit)	1707.5
Type	Vertical U-tube Feeding-type
Number of units	2
Surface area (ft <sup>2</sup> /unit)	123,540
Shell design pressure (psia)	1200
Zero load temperature (°F)	557
Feedwater temperature (°F)	440
Exit steam pressure (psia)	836
Steam flow (lb/hr per steam generator)	7.49×10 <sup>6</sup>
Total steam flow (lb/hr)	14.97×10 <sup>6</sup>

equipment. The inlet and outlet nozzles are offset, with the inlet positioned above the outlet, to allow mid-loop operation for removal of a main coolant pump without discharge of the core.

RCS and steam system overpressure protection during power operation are provided by the pressurizer safety valves and the steam generator safety valves, in conjunction with the action of the reactor protection system. Combinations of these systems provide compliance with the overpressure protection requirements of the NRC for PWR

systems. Low temperature overpressure protection is provided by a relief valve in the suction line of the RNS system. The sizing and use of the relief valve for low temperature overpressure protection is also consistent with government guidelines.

The new design by AREVA-NP is called the evolutionary power reactor (EPR). It is going through the approval process. General features include:

- Airplane crash resistance
- Core melt catcher
- Quadruple redundancy with independent trains for each safeguards system
- Lowered probability of core damage
- Double containment shell
- Accident consequences limited by an optimal combination of passive and active safety systems
- Maximum power output per site
- Lower natural uranium consumption per MWh
- Reduced long-lived high-level waste generation during operation
- Capability to use mixed uranium and plutonium oxide fuel from recycled used fuel
- Minimal quantities of high-activity metal structure
- Less water usage than present operating reactors thanks to improved thermal efficiency
- Thermal power: 4250 or 4500 MW
- Electrical power approximately 1600 MW or 36% efficiency
- Four primary loops like most current generation PWRs
- 241 Fuel assemblies operating to >60 GWd/tonne of oxide fuel
- Steam produced at a pressure of 78 bar
- Design seismic limit: 0.25g
- Service life of the plant expected: 60 years

### 2.15.2 Chemical Control of the Coolant System

Returning to the Westinghouse AP reactor design, the primary coolant system water chemistry is selected to minimize corrosion. Routinely scheduled analyses of the coolant chemical composition are performed to verify that the reactor coolant chemistry meets the specifications. Other additions, such as those to reduce activity transport and deposition, may be added to the system. The CVCS provides a means for adding chemicals to the RCS. The chemicals perform the following functions:

- Control the pH of the coolant during prestartup testing and subsequent operation
- Scavenge oxygen from the coolant during heatup
- Control radiolysis reactions involving hydrogen, oxygen, and nitrogen during power operations following startup

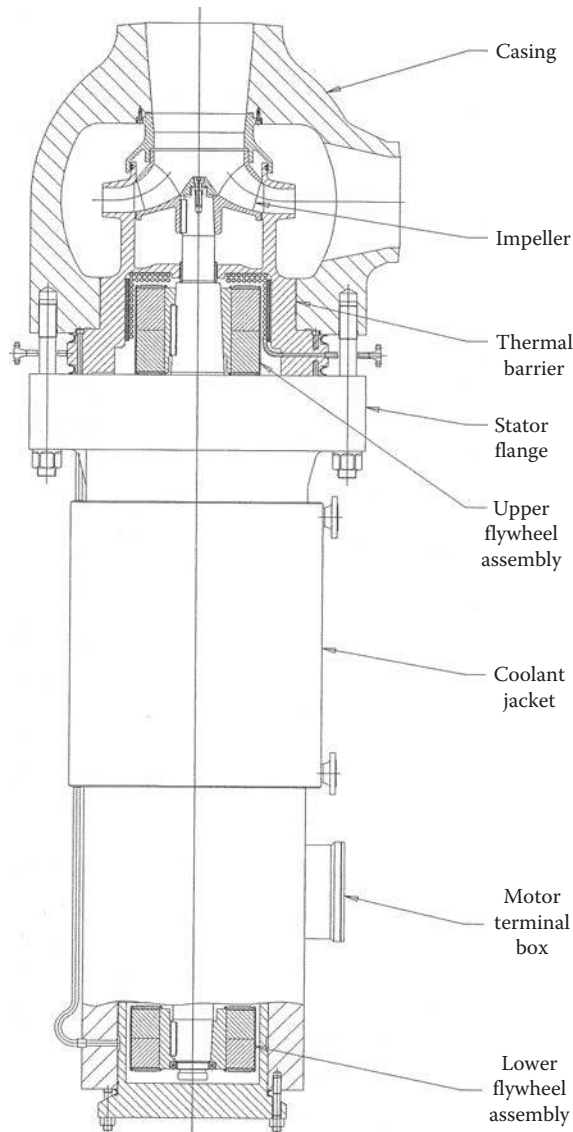
The pH control chemical is lithium hydroxide, enriched in the lithium-7 isotope to 99.9%. This chemical is chosen for its compatibility with the materials and water



chemistry of borated water/stainless steel/zirconium/nickel–chromium–iron systems. In addition, lithium-7 is produced in solution from the neutron irradiation of the dissolved boron in the coolant. The lithium-7 hydroxide is introduced into the RCS via the charging flow. The concentration of lithium-7 hydroxide in the RCS is maintained in the range specified for pH control. The other major isotope of lithium, Li-6, is a strong neutron absorber. The concentration of isotope Li-6 in naturally occurring lithium is much higher than 0.1% and consequently would interfere with the chemical shim (boron-10) system. During reactor startup from the cold condition, hydrazine is used as an oxygen-scavenging agent. The hydrazine solution is introduced into the RCS in the same manner as described for the pH control agent. The reactor coolant is treated with dissolved hydrogen to control the net decomposition of water by radiolysis in the core region. The hydrogen reacts with oxygen introduced into the RCS by the radiolysis effect on water molecules. Hydrogen makeup is supplied to the RCS by direct injection of high-pressure gaseous hydrogen, which can be adjusted to provide the correct equilibrium hydrogen concentration. Boron in the form of boric acid is added to the RCS for long-term reactivity control of the core. Suspended solid (corrosion product particulates) and other impurity concentrations are maintained below specified limits by controlling the chemical quality of makeup water and chemical additives and by purification of the reactor coolant through the CVCS.

### 2.15.3 RCP

The RCP is a single stage, hermetically sealed, high-inertia, centrifugal canned-motor pump. It pumps large volumes of reactor coolant at high pressures and temperature. Figure 2.25 shows the RCP. Table 2.7 gives the RCP design parameters. An RCP is directly connected to each of two outlet nozzles on the steam generator channel head. The two pumps on a steam generator turn in the same direction. A canned motor pump contains the motor and all rotating components inside a pressure vessel. The pressure vessel consists of the pump casing, thermal barrier, stator shell, and stator cap, which are designed for full RCS pressure. The stator and rotor are encased in corrosion-resistant cans that prevent contact of the rotor bars and stator windings by the reactor coolant. Because the shaft for the impeller and rotor is contained within the pressure boundary, seals are not required to restrict leakage out of the pump into containment. A gasket and canopy seal-type connection between the pump casing, the stator flange, and the thermal barrier is provided. This design provides definitive leak protection for the pump closure. To access the internals of the pump and motor, the canopy seal weld is severed. When the pump is reassembled a canopy seal is re-welded. Canned-motor RCPs have a long history of safe, reliable performance in military and commercial nuclear plant service. The RCP driving motor is a vertical, water-cooled, squirrel-cage induction motor with a canned rotor and a canned stator. It is designed for removal from the casing for inspection, maintenance, and replacement (if required). The stator can protect the stator (windings and insulation) from the controlled portion of the reactor coolant circulating inside the motor and bearing cavity. The can on the rotor isolates the copper rotor bars from the system and minimizes the potential for the copper to plate out in other areas. The motor is cooled by component cooling water circulating through a cooling jacket on the outside of the motor housing and through a thermal barrier between the pump casing and the rest of the motor internals. Inside the cooling jacket are coils filled with circulating rotor cavity coolant. This rotor cavity coolant is a controlled volume of reactor coolant that circulates inside the rotor cavity. After the rotor cavity coolant is cooled in the cooling jacket, it enters the lower end



**FIGURE 2.25**  
AP1000 reactor coolant pump.

of the rotor and passes axially between the rotor and stator cans to remove heat from the rotor and stator. Each pump motor is driven by a variable speed drive, which is used for pump startup and operation when the reactor trip breakers are open. When the reactor trip breakers are closed, the variable frequency drives are bypassed and the pumps run at constant speed. A flywheel, consisting of two separate assemblies, provides rotating inertia that increases the coast down time for the pump. Each flywheel assembly is a composite of a uranium alloy flywheel casting or forging contained within a welded nickel–chromium–iron alloy enclosure. The upper flywheel assembly is located between the motor and pump impeller. The lower assembly is located within the canned motor below the thrust bearing.

**TABLE 2.7**

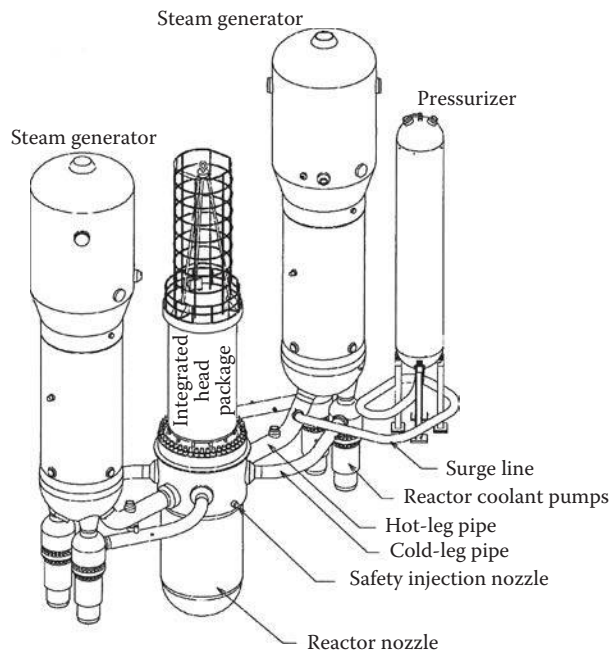
## Reactor Coolant Pump Design Parameters

Unit design pressure (psia)	2500
Unit design temperature (°F)	650
Unit overall length, (ft-in)	21–11.5
Component cooling water flow (gpm)	600
Maximum continuous component cooling water inlet temperature (°F)	95
Total weight motor and casing, dry (lb) nominal	184,500
<i>Pump</i>	
Design flow (gpm)	78,750
Developed head (ft)	365
Pump discharge nozzle, inside diameter (in.)	22
Pump suction nozzle, inside diameter (in.)	26
Speed (synchronous) (rpm)	1800
<i>Motor</i>	
Type	Squirrel cage induction
Voltage (V)	6900
Phase	3
Frequency (Hz)	60
Insulation class	Class H or N
<i>Current (amp)</i>	
Starting	Variable
Nominal input, cold reactor coolant	Variable
Motor/pump rotor minimum required moment of inertia (lb-ft <sup>2</sup> )	16,500

Surrounding the flywheel assemblies are the heavy walls of the motor end closure, casing, thermal barrier flange, stator shell, or main flange.

#### 2.15.4 Steam Generator

The steam generator channel head, tubesheet, and tubes are a portion of the RCP boundary. The tubes transfer heat to the steam system while retaining radioactive contaminants in the primary system. The steam generator removes heat from the RCS during power operation and anticipated transients and under natural circulation conditions. The steam generator heat transfer function and associated secondary water and steam systems are not required to provide a safety-related safe shutdown of the plant. The steam generator secondary shell functions as containment boundary during operation and during shutdown when access opening closures are in place. The AP1000 steam generator is a vertical-shell U-tube evaporator with integral moisture separating equipment. Figures 2.26 and 2.27 show the steam generator, indicating several of its design features. The design of the Model Delta-125 steam generator, except for the configuration of the channel head, is similar to an upgraded Model Delta-75 steam generator. The Delta-75 steam generator has been placed in operation as a replacement steam generator in older design plants. Steam generator design features are described in the following paragraphs. The steam generator design requirements are listed in Table 2.8 and the design parameters are listed in Table 2.9.

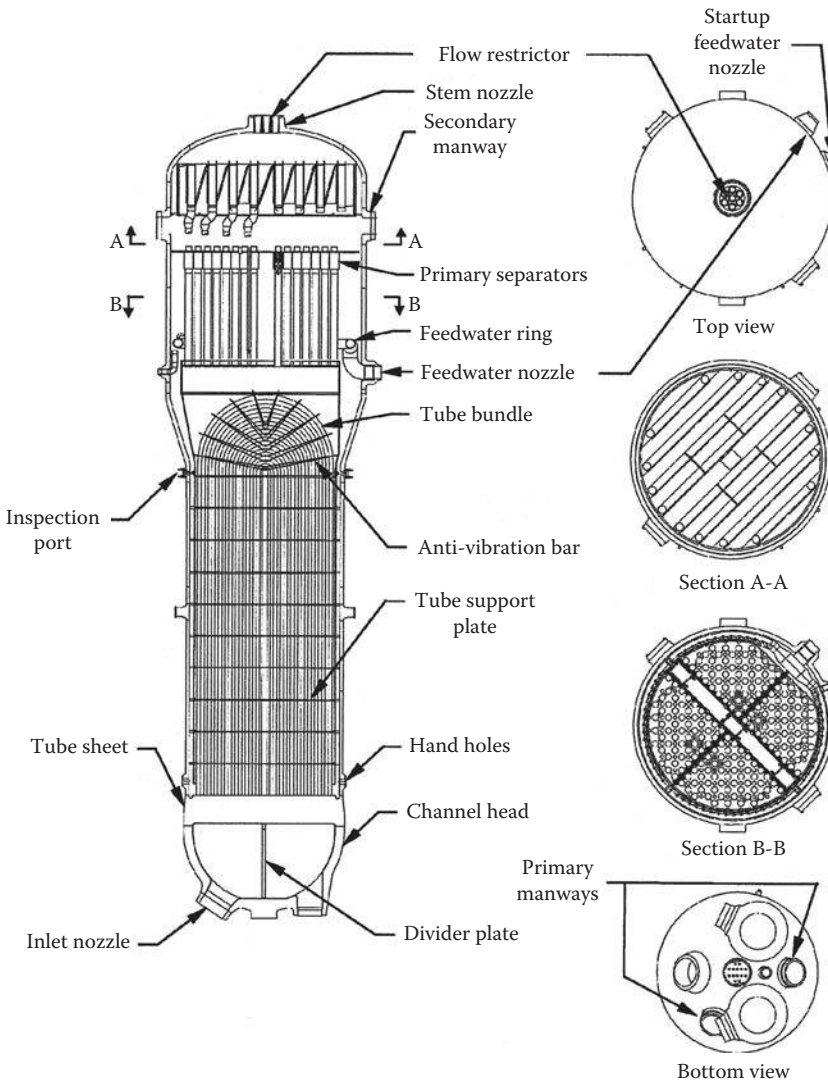


**FIGURE 2.26**  
AP1000 primary system showing two rather than four steam generators.

On the primary side, the reactor coolant flow enters the primary chamber via the hot-leg nozzle. The lower portion of the primary chamber is elliptical and merges into a cylindrical portion, which mates to the tubesheet. This arrangement provides enhanced access to all tubes, including those at the periphery of the bundle, with robotics equipment. This feature enhances the ability to inspect, replace, and repair portions of the AP1000 unit compared with the more spherical primary chamber of earlier designs. The head is divided into inlet and outlet chambers by a vertical divider plate extending from the apex of the head to the tubesheet. The reactor coolant flow enters the inverted U-tubes, transferring heat to the secondary side during its traverse, and returns to the cold-leg side of the primary chamber. The flow exits the steam generator via two cold-leg nozzles to which the canned-motor RCPs are directly attached. A high-integrity, nickel–chromium–iron (Alloy 690) weld is made to the nickel–chromium–iron alloy-buttered ends of these nozzles.

A passive residual heat removal (PRHR) nozzle attaches to the bottom of the channel head of the loop-1 steam generator on the cold-leg portion of the head. This nozzle provides recirculated flow from the PRHR heat exchanger to cool the primary side under emergency conditions. A separate nozzle on one of the steam generator channel heads is connected to a line from the CVCS. The nozzle provides for purification flow and makeup flow from the CVCS to the RCS.

The AP1000 steam generator channel head has provisions to drain the head. To minimize deposits of radioactive corrosion products on the channel head surfaces and to enhance the decontamination of these surfaces, the channel head cladding is machined or electropolished for a smooth surface. The primary manways provide enhanced primary chamber access compared with previous model steam generators. Should steam generator replacement using a channel head cut be required, the arrangement of the AP1000 steam generator



**FIGURE 2.27**  
AP1000 steam generator details.

channel head facilitates steam generator replacement in two ways. It is completely unobstructed around its circumference for mounting cutting equipment and is long enough to permit post-weld heat treatment with minimal effect of tubesheet acting as a heat sink. The tubes are fabricated of nickel–chromium–iron Alloy 690. The tubes undergo thermal treatment following tube-forming operations. The tubes are tack-expanded, welded, and hydraulically expanded over the full depth of the tubesheet. Westinghouse has used this practice in F-type steam generators. It was selected because of its capability to minimize secondary water access to the tube-to-tubesheet crevice. Residual stresses smaller than from other expansion methods result from this process and are minimized by tight control of the pre-expansion clearance between the tube and tubesheet hole. Support of the tubes

**TABLE 2.8**

## Steam Generator Design Requirements

Type	Vertical U-Tube Feeding-Type
Design pressure, reactor side (psia)	2500
Design Pressure, steam side (psia)	1200
Design pressure, primary to secondary (psi)	1600
Design temperature, reactor coolant side (°F)	650
Design temperature, steam side (°F)	600
S/G Power (MWt/unit)	1707.5
Total heat transfer surface area (ft <sup>2</sup> )	123,538
Steam nozzle outlet pressure (psia)	836
Steam flow (lb/h per S/G)	$7.49 \times 10^6$
Total steam flow (lb/h)	$14.97 \times 10^6$
Maximum moisture carryover (weight percent) maximum	0.25
No load temperature (°F)	557
Feedwater temperature (°F)	440
Number of tubes per unit	10,025
Tube outer diameter (in.)	0.688
Total wall thickness (in.)	0.040
Tube pitch (in.)	0.980 (triangular)

**TABLE 2.9**

## Steam Generator Nominal Design Parameters

Tube pitch (in.)	0.980 (triangular)
Overall length (in.)	884.26 <sup>a</sup>
Upper shell I.D. (in.)	210
Lower shell I.D. (in.)	165
Tubesheet thickness (in.)	31.13 <sup>b</sup>
Primary water volume (ft <sup>3</sup> )	2077
Water volume in tubes (ft <sup>3</sup> )	1489
Water volume in plenums (ft <sup>3</sup> )	588
Secondary water volume (ft <sup>3</sup> )	3646
Secondary steam volume (ft <sup>3</sup> )	5222
Secondary water mass (lbm)	175,758
Design fouling factor (h-°F-ft <sup>2</sup> /BTU)	$1.1 \times 10^{-4}$

<sup>a</sup> Measured from steam nozzle to the flat, exterior portion of the channel head.

<sup>b</sup> Base metal thickness.

is provided by ferritic stainless steel tube support plates. The holes in the tube support plates are broached with a hole geometry to promote flow along the tube and to provide an appropriate interface between the tube support plate and the tube. Anti-vibration bars installed in the U-bend portion of the tube bundle minimize the potential for excessive vibration. Steam is generated on the shell side, flows upward, and exits through the outlet nozzle at the top of the vessel. Feedwater enters the steam generator at an elevation above the top of the U-tubes through a feedwater nozzle. The feedwater enters a feeding via a

welded thermal sleeve connection and leaves it through nozzles attached to the top of the feeding. The nozzles are fabricated of an alloy that is very resistant to erosion and corrosion with the expected secondary water chemistry and flow rate through the nozzles. After exiting the nozzles, the feedwater flow mixes with “saturated” (at the boiling point) water removed by the moisture separators. The flow then enters the downcomer annulus between the wrapper and the shell.

Fluid instabilities and water hammer phenomena are important considerations in the design of steam generators. Water-level instabilities can occur from density wave instabilities which could affect steam generator performance. Density wave instability is avoided in the AP1000 steam generator by including appropriate pressure losses in the downcomer and the risers that lead to negative damping factors. Steam generator bubble collapse water hammer has occurred in certain early PWR steam generator designs having feedrings equipped with bottom discharge holes. Prevention and mitigation of feedline-related water hammer has been accomplished through an improved design and operation of the feedwater delivery system. The AP1000 steam generator and feedwater system incorporate features designed to eliminate the conditions linked to the occurrence of steam generator water hammer. The steam generator features include introducing feedwater into the steam generator at an elevation above the top of the tube bundle and below the normal water level by a top discharge feeding. The top discharge of the feeding helps to reduce the potential for vapor formation in the feeding. This minimizes the potential for conditions that can result in water hammer in the feedwater piping. The feedwater system features designed to prevent and mitigate water hammer include a short, horizontal, or downward sloping feedwater pipe at steam generator inlet. These features minimize the potential for trapping pockets of steam which could lead to water hammer events. Stratification and striping are reduced by an upturning elbow inside the steam generator which raises the feeding relative to the feedwater nozzle. The elevated feeding reduces the potential for stratified flow by allowing the cooler, denser feedwater to fill the nozzle/elbow arrangement before rising into the feeding. The potential for water hammer, stratification, and striping is additionally reduced by the use of a separate startup feedwater nozzle. The startup feedwater nozzle is located at an elevation that is just below the main feedwater nozzle and is rotated circumferentially away from the main feedwater nozzle. A startup feedwater spray system independent of the main feedwater feeding is used to introduce startup feedwater into the steam generator. The layout of the startup feedwater piping includes the same features as the main feedwater line to minimize the potential for water hammer.

At the bottom of the wrapper, the water is directed toward the center of the tube bundle by the lowest tube support plate. This recirculation arrangement serves to minimize the low-velocity zones having the potential for sludge deposition. As the water passes the tube bundle, it is converted to a steam–water mixture. Subsequently, the steam–water mixture from the tube bundle rises into the steam drum section, where centrifugal moisture separators remove most of the entrained water from the steam. The steam continues to the secondary separators, or dryers, for further moisture removal, increasing its quality to a designed minimum of 99.75% (0.25% by weight maximum liquid). Water separated from the steam combines with entering feedwater and recirculates through the steam generator. A sludge collector located amidst the inner primary separator risers provides a preferred region for sludge to settle away from the tubesheet and tube support plates. The dry, saturated steam exits the steam generator through the outlet nozzle, which has a steam-flow restrictor.

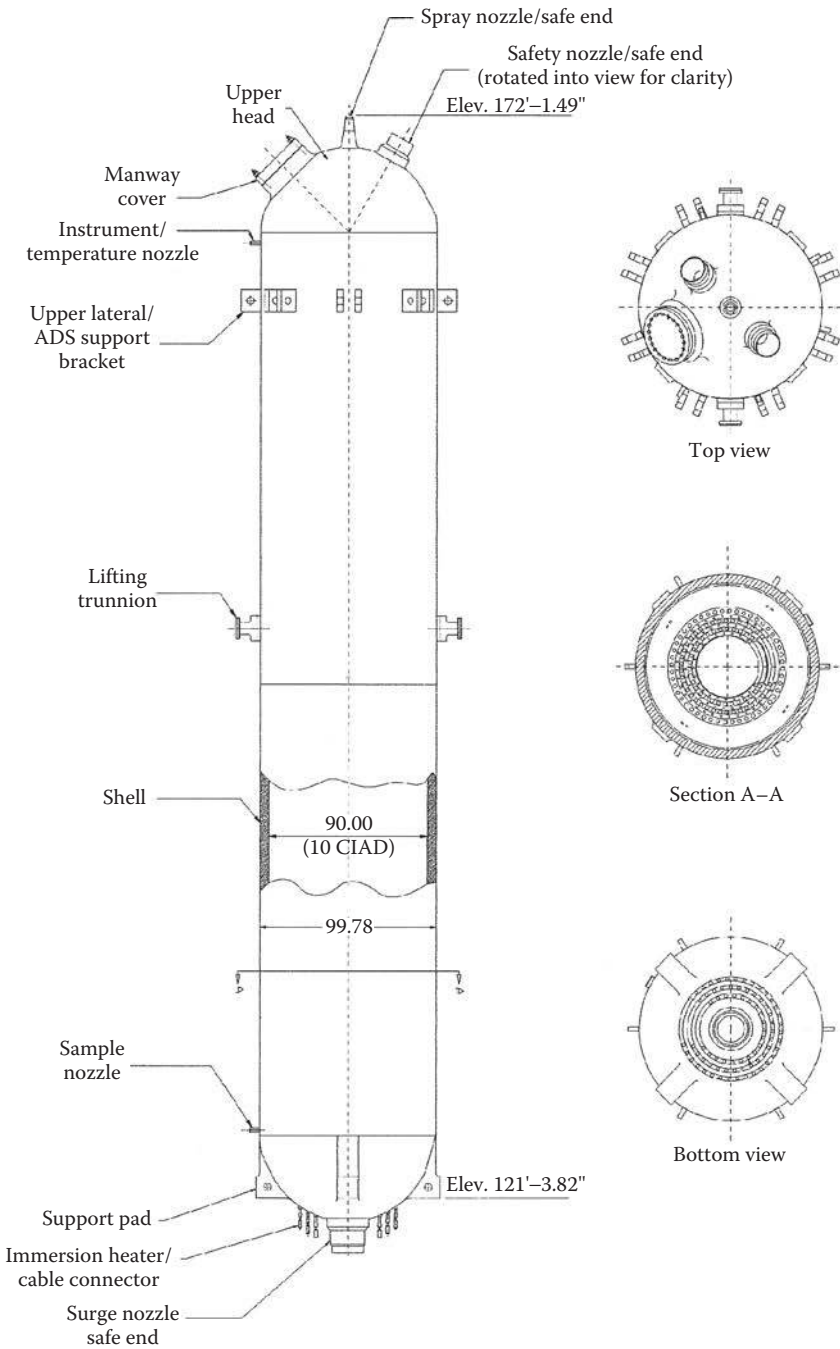
### 2.15.5 Reactor Coolant Pressurizer

The pressurizer maintains the system pressure during operation and limits pressure transients. During the reduction or increase of plant load, the pressurizer accommodates volume changes in the reactor coolant. It is directly connected to the hot leg of the RCS. The pressurizer is a vertical, cylindrical vessel having hemispherical top and bottom heads constructed of low alloy steel. Internal surfaces exposed to the reactor coolant are clad austenitic stainless steel. The general configuration of the pressurizer is shown in Figure 2.28. The design data for the pressurizer are given in Tables 2.10 and 2.11. Nickel–chromium–iron alloys are not used for heater wells or instrument nozzles for materials compatibility reasons. The spray line nozzles and the automatic depressurization and safety valve connections are located in the top head of the pressurizer vessel. Spray flow is modulated by automatically controlled air-operated valves. The spray valves can also be operated manually from the control room. In the bottom head at the connection of the surge line to the surge nozzle, a thermal sleeve protects the nozzle from thermal transients. A retaining screen above the surge nozzle prevents passage of foreign matter from the pressurizer to the RCS. Baffles in the lower section of the pressurizer prevent an in-surge of cold water from flowing directly to the steam/water interface. The baffles also assist in mixing the incoming water with the water in the pressurizer. The retaining screen and baffles also act as a diffuser. The baffles also support the heaters to limit vibration. Electric direct-immersion heaters are installed in vertically oriented heater wells located in the pressurizer bottom head. The heater wells are welded to the bottom head and form part of the pressure boundary. Heaters can be removed for maintenance or replacement. Heaters are grouped into a control group and backup groups. Heaters in the control group are proportional heaters which are supplied with continuously variable power to match heating needs. Heaters in the backup group are off or at full power. The power supply to the heaters is a 480-V 60 Hz three-phase circuit. Each heater is connected to one leg of a delta-connected circuit and is rated at 480 V with one-phase current. A manway in the upper head provides access to the internal space of the pressurizer in order to inspect or maintain the spray nozzle. The manway closure is a gasketed cover held in place with threaded fasteners. Brackets on the upper shell attach the structure (a ring girder) of the pressurizer safety and relief valve (PSARV) module. The PSARV module includes the safety valves and the first three stages of ADS valves. The support brackets on the pressurizer represent the primary vertical load path to the building structure. Sway struts between the ring girder and pressurizer compartment walls also provide lateral support to the upper portion of the pressurizer. Four steel columns attach to pads on the lower head to provide vertical support for the vessel. The columns are based at elevation 107'-2". Lateral support for the lower portion of the vessel is provided by sway struts between the columns and compartment walls.

### 2.15.6 ADS

ADS valves are part of the RCS and interface with the PXS. Twenty valves are divided into four depressurization stages. These stages connect to the RCS at three locations. The ADS first, second, and third stage valves are included as part of the PSARV module and are connected to nozzles on top of the pressurizer. Fourth-stage valves connect to the hot leg of each reactor coolant loop. Opening of the ADS valves is required for the PXS to function as required to provide emergency core cooling following postulated accident conditions. First-stage valves may also be used as required following an accident to remove





**FIGURE 2.28**  
AP1000 primary system pressurizer.

**TABLE 2.10****Pressurizer Design Data Including Heater Group Parameters**

<i>Pressurizer Design Data</i>	
Design pressure (psig)	2485
Design temperature (°F)	680
Surge line nozzle nominal diameter (in.)	18
Spray line nozzle nominal diameter (in.)	4
Safety valve nozzle nominal diameter (in.)	14
Internal volume (ft <sup>3</sup> )	2100
<i>Pressurizer Heater Group Parameters</i>	
Voltage ( $V_{ac}$ )	480
Frequency (Hz)	60
Power capacity (kW)	
Control group	370
Backup group A	245
Backup group B	245
Backup group C	370
Backup group D	370

**TABLE 2.11****Pressurizer Safety Valves—Design Parameters**

Number	2
Minimum required relieving capacity per valve (lbm/h)	750,000 at 3% accumulation
Set pressure (psig)	2485 ± 25 psi
Design temperature (°F)	680
Fluid	Saturated steam
Backpressure	
Nominal (psig)	3–5
Expected maximum during discharge (psig)	500
Environmental conditions	
Ambient temperature (°F)	50–120
Relative humidity (%)	0–100
<i>Residual Heat Removal Relief Valve—Design Parameters</i>	
Number	1
Nominal relieving capacity per valve, ASME flow rate (gpm)	850
Nominal set pressure (psig)	500
Full-open pressure, with accumulation (psig)	550
Design temperature (°F)	400
Fluid	Reactor coolant
Backpressure	
Nominal (psig)	3–5
Expected maximum during discharge (psig)	200
Environmental conditions	
Ambient temperature (°F)	50–120
Relative humidity (%)	0–100

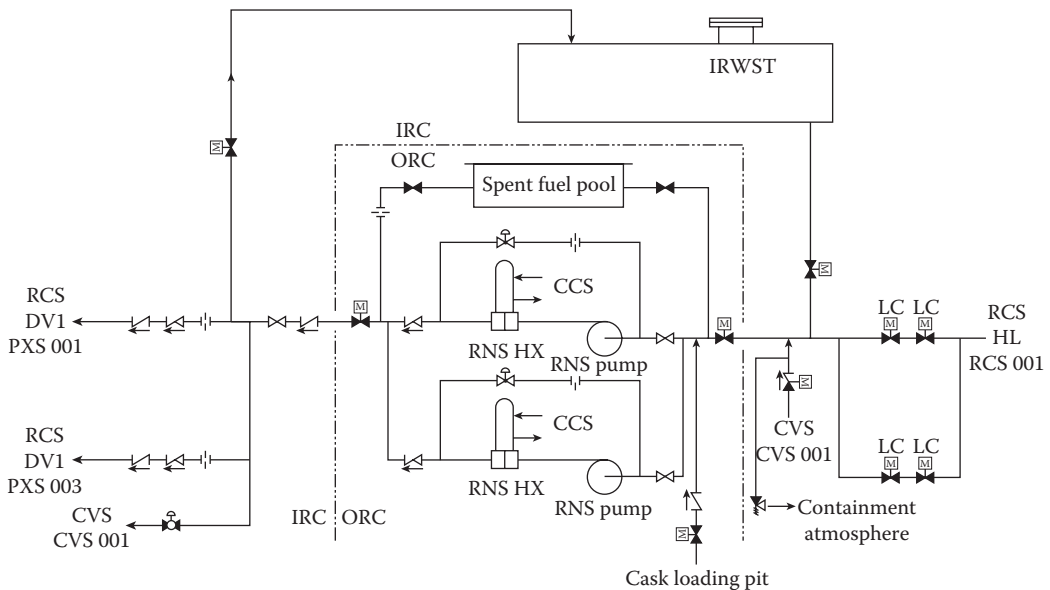
noncondensable gases from the steam space of the pressurizer. The flashing coolant is directed to the IRWST by means of spargers. The tank itself is vented to containment with the buildup of very low internal pressure.

### 2.15.7 RNS

The RNS performs the following major functions:

- *RCS shutdown heat removal*—Remove heat from the core and the RCS during shutdown operations
- *Shutdown purification*—Provide RCS and refueling cavity purification flow to the CVCS during refueling operations
- *IRWST cooling*—Provide cooling for the IRWST
- *RCS makeup*—Provide low-pressure makeup to the RCS
- *Post-accident recovery*—Remove heat from the core and the RCS following successful mitigation of an accident by the PXS
- *Low-temperature overpressure protection*—Provide low-temperature overpressure protection for the RCS during refueling, startup, and shutdown operations
- *Long-term, post-accident containment inventory makeup flowpath*—Provide long-term, post-accident makeup flowpath to the containment inventory
- *Spent fuel pool cooling*—Provide backup for cooling the spent fuel pool

The layout of the RNS is shown in Figure 2.29 and the component data are listed in Table 2.12.



**FIGURE 2.29**  
AP1000 normal residual heat removal system.

**TABLE 2.12**

## Heat Removal System Component Data

<i>Nominal RHR Pumps (per pump)</i>		
Minimum flow required for shutdown cooling (gpm)		1425
Minimum flow required for low pressure makeup (gpm)		1100
Design flow (gpm)		1500
Design head (ft)		360
<i>Normal RHR Heat Exchangers</i>		
Minimum UA required for shutdown cooling (BTU/h-°F)		2.2×10 <sup>6</sup>
Design heat removal capacity (BTU/h) <sup>a</sup>		23×10 <sup>6</sup>
	<b>Tube Side</b>	<b>Shell Side</b>
Design flow (lb/h)	750,000	1,405,000
Inlet temperature (°F)	125	87.5
Outlet temperature (°F)	94	104
Fluid	Reactor coolant	CCS
<i>Design Bases for Normal Residual Heat Removal System Operation</i>		
RNS initiation (hours after reactor shutdown)		4
RCS initial pressure (psig)		450
RCS initial temperature (°F)		350
CCS design temperature (°F)		95
Cool down time (hours after shutdown)		96
RCS temperature at end of cool down (°F)		125

<sup>a</sup> Design heat removal capacity is based on decay heat at 96 hours after reactor shutdown.

## 2.16 PXS

The primary function of the PXS is to provide emergency core cooling following postulated design basis events. To accomplish this primary function, the PXS is designed to perform the functions detailed below.

- *Emergency core decay heat removal*—Provide core decay heat removal during transients, accidents, or whenever the normal heat removal paths are lost. This heat removal function is available at RCS conditions including shutdowns. During refueling operations, when the IRWST is drained into the refueling cavity, other passive means of core decay heat removal are utilized.
- *RCS emergency makeup and boration*—Provide RCS makeup and boration during transients or accidents when the normal RCS makeup supply from the CVCS is unavailable or is insufficient.
- *Safety injection*—Provide safety injection to the RCS to provide adequate core cooling for the complete range of LOCA, up to and including the double-ended rupture of the largest primary loop RCS piping.

- *Containment pH control*—Provide for chemical addition to the containment during post-accident conditions to establish floodup chemistry conditions that support radionuclide retention with high radioactivity in containment, and to prevent corrosion of containment equipment during long-term floodup conditions.

The PXS is designed to operate without the use of active equipment such as pumps and AC power sources. The PXS depends on reliable passive components and processes such as gravity injection and expansion of compressed gases. The PXS requires a one-time alignment of valves upon actuation of the specific components.

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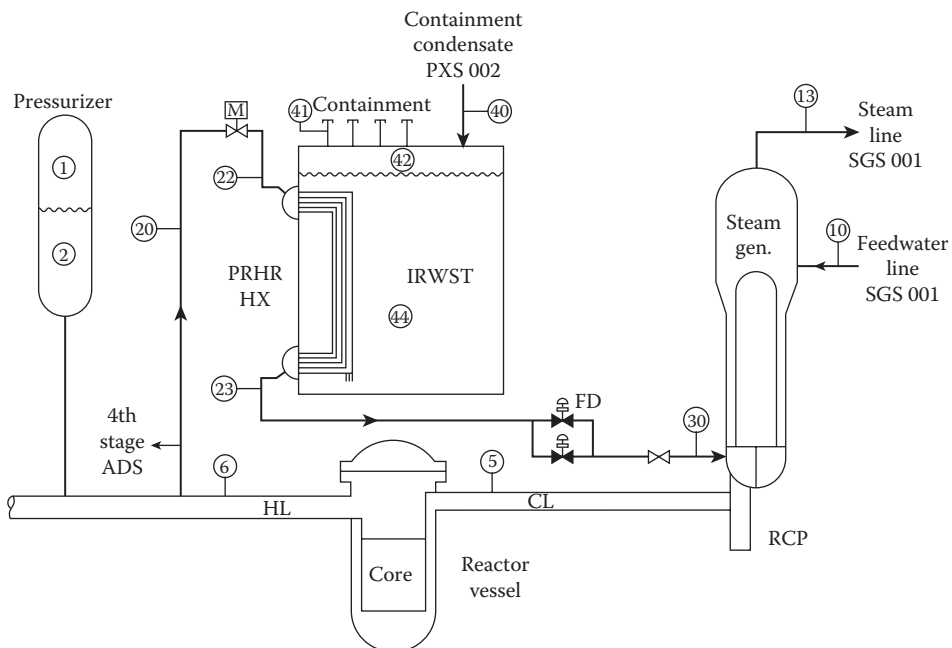
## 2.17 Detection and Ignition of Hydrogen

Another important component of the safety-related systems is the hydrogen detection and ignition system. Reaction of steam with the Zircaloy-4 or ZIRLO™ fuel cladding can release hydrogen into the containment if there is a pipe rupture or other venting of RCS coolant outside the pressure boundary. An explosive mixture can result if the hydrogen concentration with dry air reaches about 20%. The limit is higher if a large amount of steam is also present. The objective of the hydrogen control system is to detect it at levels <20% and neutralize it by reacting it with air in catalytic heaters located at 64 places around the inside of containment such that natural circulation will assure adequate mixing and prevent the formation of an explosive mixture anywhere.

Zircaloy-4 is an alloy of zirconium with about 1.5% tin and lesser amounts of iron, nickel, chromium, and about 200 ppm of carbon. It is an excellent window for neutrons, forms a tough, adherent oxide coating to prevent corrosion, and its mechanical properties provide adequate strength and ductility. It does not interact with the uranium dioxide fuel and barely reacts with fission products. Large ingots of the alloy are cut down into “trexes” a few inches thick and several inches in diameter. A hole, an inch or so in diameter, is carefully centered in the trex. From there it is worked by a three-rod process that scuffs the material into a high tensile hydrostatic stress at the center. A mandrel pushes from behind and smoothens the inside surface as the material gets longer, thinner, and smaller in diameter through many steps until it becomes tubing. Another process is called the Pilger process; a pair of mandrels squeeze the material in one spot, then the trex/tube is turned slightly and the process is repeated. A spiral path is followed until the tube reaches the desired dimensions for that step. The zirconium cold works readily and must be annealed between steps to maintain ductility for the next step in the process. Westinghouse has patented what they believe is an improvement, called ZIRLO™. The newer alloy includes 0.5–2.0 weight percent niobium as well as the other ingredients of Zircaloy-4. The alloy is also preferably subjected to intermediate recrystallization anneals at about 1200–1300°F, and to a beta quench two steps prior to final size.

## 2.18 IRWST

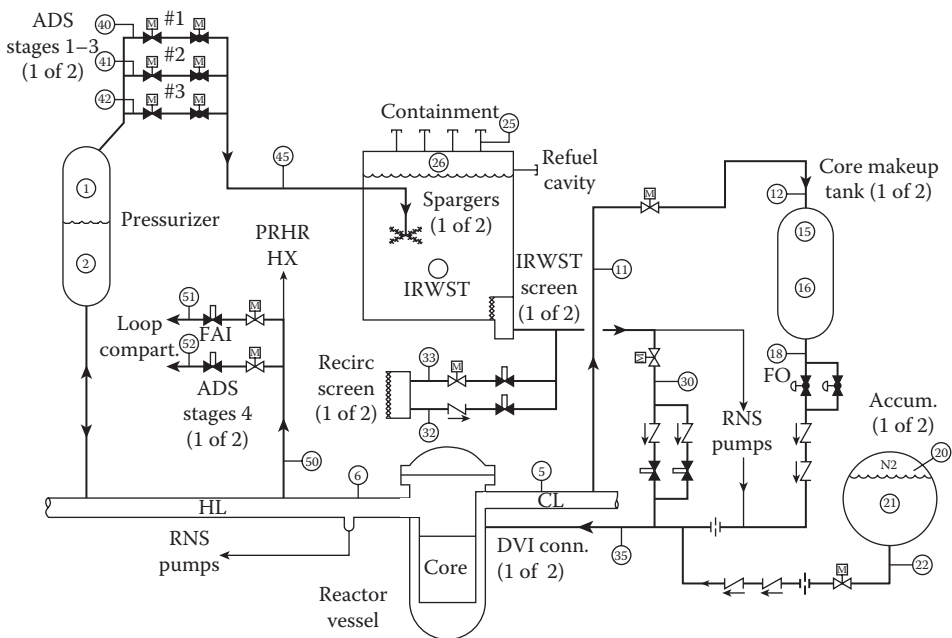
Another vital piece of equipment inside the containment, but not inside the RCS pressure boundary, is the IRWST (Figure 2.30). In older plant designs, the tank was located outside the shield building and was considered vulnerable. Locating the IRWST inside the containment allows it to participate in additional important functions. The IRWST is a large, stainless steel-lined tank located underneath the operating deck inside the containment. The tank is designed to meet seismic requirements and is constructed as an integral part of the containment internal structures, but is isolated from the steel containment vessel. The bottom of the IRWST is above the RCS loop elevation so that the borated refueling water can drain by gravity into the RCS after it is sufficiently depressurized. The IRWST is connected to the RCS through both direct vessel injection lines. The IRWST and contents are at the existing temperature and pressure in the containment. Vents are installed in the roof of the tank. These vents are normally closed to contain water vapor and radioactive gases within the tank during normal operation and to prevent debris from entering the tank from the containment operating deck. The vents open with a slight pressurization of the IRWST. These vents provide a path to vent steam released by the spargers or generated by the PRHR heat exchanger into the containment atmosphere. Other vents also open on small-pressure differentials, such as during a LOCA, to prevent damage to the tank. Overflows are provided from the tank to the refueling cavity to accommodate volume and mass increases during PRHR heat exchanger or ADS operation, while minimizing the floodup of the containment. The IRWST does not contain materials in the tank or the recirculation path that could plug the outlet screens. The tank contains one PRHR heat



**FIGURE 2.30**  
AP1000 passive decay heat removal system.

exchanger and two depressurization spargers. The top of the PRHR heat exchanger tubes are located underwater and extend down into the IRWST. The spargers are also submerged in the tank, with the spargers' midarms located below the normal water level. The tank is sized to provide the flooding of the refueling cavity for normal refueling, the post-LOCA flooding of the containment for RCS long-term cooling mode, and to support the PRHR heat exchanger operation. Flow out of the IRWST during the injection mode includes conservative allowances for spill flow during a direct vessel injection line break. The IRWST can provide sufficient injection until the containment sump floods up high enough to initiate recirculation flow. The injection duration varies greatly depending upon the specific event. A direct vessel injection line break more rapidly drains the tank and speeds containment floodup. The containment floodup volume for a LOCA is  $<73,500 \text{ ft}^3$  (excluding the volume of the IRWST itself) below a containment elevation of 108 ft. Connections to the tank provide for transfer to and from the RCS/refueling cavity via the RNS, purification and sampling via the spent fuel pit cooling system, and remotely adjusting boron concentration to the CVCS. Also, the RNS can provide cooling of the IRWST. The water level and temperature in the tank are monitored by indicators and alarms. The operator can take action, as required, to meet the technical specification requirements for system operability.

The PRHR exchanger consists of inlet and outlet channel heads connected together by vertical C-shaped tubes (Figure 2.30). The tubes are supported inside the IRWST. The top of the tubes is several feet below the tank water surface. The PRHR heat exchanger is designed to meet seismic requirements. The heat exchanger inlet piping connects to an inlet channel head located near the outside top of the tank. The inlet channel head and tubesheet are attached to the tank wall via an extension flange. The heat exchanger is supported by a frame which is attached to the IRWST floor and ceiling. The extended flange is designed to accommodate thermal expansion. The heat exchanger outlet piping is connected to the outlet channel head, which is vertically below the inlet channel head, near the tank bottom. The outlet channel head has an identical structural configuration to the inlet channel head. Both channel head tubesheets are similar to the steam generator tubesheets and they have manways for inspection and maintenance access. The PRHR heat exchanger is designed to remove sufficient heat so that its operation, in conjunction with available inventory in the steam generators, provides RCS cooling and prevents water relief through the pressurizer safety valves during loss of main feedwater or main feedline break events. PRHR heat exchanger flow and inlet and outlet line temperatures are monitored by indicators and alarms. The operator can take action, as required, to meet the technical specification requirements or follow emergency operating procedures for control of the PRHR heat exchanger operation. Two reactor coolant depressurization spargers are provided as shown in Figure 2.31. Each one is connected to an ADS discharge header (shared by three ADS stages). They are also submerged in the IRWST. Each sparger has four branch arms inclined downward. The connection of the sparger branch arms to the sparger hub are submerged below tank overflow level by  $\leq 11.5 \text{ ft}$ . The spargers are designed to meet seismic requirements. The spargers perform a nonsafety-related function: minimizing plant cleanup and recovery actions following automatic depressurization. They are designed to distribute steam into the IRWST, thereby promoting more effective steam condensation. The first three stages of ADS valves discharge through the spargers and are designed to pass sufficient depressurization venting flow, with an acceptable pressure drop, to support the depressurization system performance requirements. The installation of the spargers prevents undesirable and/or excessive dynamic loads (water hammer) on the IRWST and other structures. Each sparger is sized to discharge at a flow rate that supports ADS performance, which in turn allows adequate PXS injection.

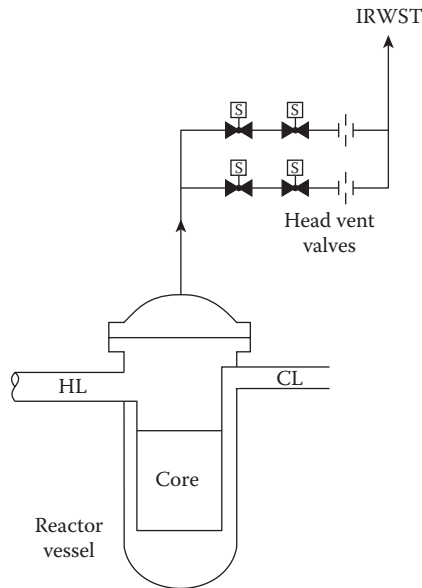


**FIGURE 2.31**  
AP1000 passive safety injection.

## 2.19 Safety Design Rationale for Venting the Reactor Vessel Head

The Three Mile Island accident in 1979 was exacerbated by uncertainty over the existence of a steam bubble in the reactor vessel head as well as in the pressurizer. The crew were reluctant to pressurize the coolant, believing that it may result in system rupture. The time delay resulted in partial core melting because of inadequate emergency cooling. Since that event, it has been a requirement to vent the reactor vessel head remotely to be certain there is no bubble there during an accident (or at any time). The requirements for high point vents are provided by the reactor vessel head vent valves and the ADS valves. The ADS system has been described. The primary function of the reactor vessel head vent is for use during plant startup to properly fill the RCS and vessel head. Both reactor vessel head vent valves and the ADS valves may be activated and controlled from the main control room. The AP1000 does not require use of a reactor vessel head vent to provide safety-related core cooling following a postulated accident. The reactor vessel head vent valves (Figure 2.32 and Table 2.13) can remove noncondensable gases or steam from the reactor vessel head to mitigate a possible condition of inadequate core cooling or impaired natural circulation through the steam generators resulting from the accumulation of noncondensable gases in the RCS. The design of the reactor vessel head vent system is in accordance with NRC requirements. The reactor vessel head vent valves can be operated from the main control room to provide an emergency letdown path which is used to prevent pressurizer overfill following long-term loss of heat sink events. An orifice is provided downstream of each set of head vent valves to limit the emergency letdown flow rate. The first-stage valves of the ADS are attached to the pressurizer and provide the capability of removing noncondensable gases from the pressurizer





**FIGURE 2.32**  
Reactor vessel head vent system.

**TABLE 2.13**

Reactor Vessel Head Vent System Design Parameters

System design pressure (psig)	2485
System design temperature (°F)	650
Number of remotely-operated valves	4
Vent line, nominal diameter (in.)	1
Head vent capacity (lbm/s) (assuming a single failure, RCS pressure at 1250 psia)	8.2

steam space following an accident. Venting of noncondensable gases from the pressurizer steam space is not required to provide safety-related core cooling following a postulated accident. Gas accumulations are removed by remote manual operation of the first-stage ADS valves. The discharge of the ADS valves is directed to the IRWST. The PRHR heat exchanger piping and the core makeup tank inlet piping in the PXS include high point vents that provide the capability of removing noncondensable gases that could interfere with heat exchanger or core makeup tank operation. These gases are normally expected to accumulate when the RCS is refilled and pressurized following refueling shutdown. Any noncondensable gases that collect in these high points can be manually vented. The discharge of the PRHR heat exchanger high point vent is directed to the IRWST. The discharge of the core makeup tank high point vent is directed to the reactor coolant drain tank. The reactor vessel head vent arrangement is designed to remove noncondensable gases or steam from the RCS via remote manual operations from the main control room through a pair of valves. The system discharges to the IRWST. The reactor vessel head vent system is designed to provide an emergency letdown path that can be used to prevent long-term pressurizer overfill following loss of heat sink events. The reactor vessel head vent is designed to limit the emergency letdown flow rate to within the capabilities of the normal makeup system. The reactor vessel head vent system can also vent noncondensable gases from the reactor head in case

of a severe accident. The system vents the reactor vessel head by using only safety-related equipment. The reactor vessel head vent system satisfies applicable requirements.

---

## 2.20 Other Passive Emergency Systems

In addition to passive systems inside the containment, there are passive systems for cooling the containment itself if it is filled with steam and gives high-pressure signals. All passive systems are intended for keeping the radioactivity contained and the fuel and other equipment protected from further degradation for the long-term following the DBA. It is assumed that the reactor cannot be accessed by personnel during this time (about 1.5 months). The equipment has other uses during normal or upset conditions, but the safety function is not compromised by other uses, such as fire protection or refueling water makeup. The passive containment cooling water storage tank (PCCWST) is incorporated into the shield-building structure above the containment vessel. The inside wetted walls of the tank are lined with stainless steel plate. It is filled with demineralized water and has the minimum required usable volume for the passive containment cooling function, over 756,000 gallons. The passive containment cooling system functions as the safety-related ultimate heat sink. The passive containment cooling water storage tank is seismically designed and missile protected. The surrounding reinforced concrete supporting structure is part of the protection for the passive cooling system. The welded seams of the plates forming part of the leak tight boundary are examined by liquid penetrant after fabrication to confirm that the boundary does not leak. The tank also has redundant level measurement channels and alarms for monitoring the tank water level and redundant temperature measurement channels to monitor and alarm for potential freezing. To maintain system operability, a recirculation loop that provides chemistry and temperature control is connected to the tank. The tank is constructed to provide sufficient thermal inertia and insulation such that draindown can be accomplished without heater operation. In addition to its containment heat removal function, the passive containment cooling water storage tank also serves as a source of makeup water to the spent fuel pool and a seismic Category I water storage reservoir for fire protection following an SSE. The PCCWST suction pipe for the fire protection system is configured so that actuation of the fire protection system will not infringe on the usable capacity allocated to the passive containment cooling function. System actuation consists of opening the passive containment cooling water storage tank isolation valves. This allows the passive containment cooling water storage tank water to be delivered to the top, external surface of the steel containment shell. The flow of water, provided entirely by the force of gravity, forms a water film over the dome and side walls of the containment structure. The cooling action reduces the pressure inside containment.

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## Suggested Readings

AP1000 design control document. Submittal to the Nuclear Regulatory Commission as document number APP-GW-GL-700, September 2004.

Patent 5112573 gives the composition of ZIRLO™ and its heat treatment. Filed August 28, 1989. Issued May 12, 1992.

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# 3

## *Boiling Water Reactors*

Kevin Theriault

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## 3.1 Introduction

### 3.1.1 Boiling Water Reactor Background

The boiling water reactors (BWRs) nuclear plant, like the pressurized water reactor (PWR), has its origins in the technology developed in the 1950s for the United States Navy nuclear submarine program. The first BWR nuclear plant to be built was the 5 MWe Vallecitos Plant (1957) near San Jose, California. The Vallecitos plant confirmed that BWR plants could successfully and safely produce electricity for a grid. The first large-scale BWR, Dresden I, followed in 1960, and since then the BWR design subsequently underwent a series of evolutionary changes with one purpose in mind: simplicity.

The major difference between the PWR and BWR is that the latter is a direct cycle nuclear system with heat generation occurring in the fuel region and water boiling in the envelope of the fuel bundles. This will be explored later.

There are approximately 92 operational BWRs in the world today and several advanced BWRs (ABWRs) currently under construction. This design comprises about 25% of the total number of units in operation globally. Current and former vendors are ASEA-Atom, Kraftwerken Union, Hitachi, Toshiba, and General Electric. Consolidation of the industrial supply base has led to continued partnerships in the nuclear supply chain.

The BWR design has been simplified in two key areas: reactor systems and containment design. Refer to Table 3.1 to see the evolution of simplification. The first BWR, Dresden 1 was, interestingly enough, not a true BWR. The design was based upon dual steam cycle, not the direct steam cycle that characterizes BWRs. Steam was generated in the reactor but then flowed to an elevated steam drum and a secondary steam generator before making its way to the turbine. The first step down the path of simplicity that led ultimately to the ABWR was elimination of the external steam drum by introducing two technical innovations: the internal steam separator and dryer.

General Electric selected the BWR as the most promising nuclear power concept because of its inherent advantages in control and design simplicity and established an atomic power equipment business in 1955 to offer it commercially. Aside from its heat source, the BWR generation cycle is substantially similar to that found in fossil-fueled power plants.

### 3.1.2 BWR-6 Product Line

The BWR-6 product line is capable of producing 20% more power from the same size pressure vessels as used in the BWR-5 product line without increasing the size of the respective buildings or support systems. Power output capabilities range from approximately 600–1400 MWe gross. Principal design features include

- Compact jet pumps with increased coolant circulation capability.
- Increased capacity from steam separators and dryers.
- More fuel bundles in standard pressure vessels and improvements in reactor internals arrangement.

**TABLE 3.1**

Evolution of the GE BWR

Product Line	First Commercial Operation Date	Representative Plant/Characteristics
BWR/1	1960	Dresden 1 Initial commercial-size BWR
BWR/2	1969	Oyster Creek Plants purchased solely on economics Large direct cycle
BWR/3	1971	Dresden 2 First jet pump application Improved ECCS: spray and flood capability
BWR/4	1972	Vermont Yankee Increased power density (20%)
BWR/5	1977	Tokai 2 Improved ECCS Valve flow control
BWR/6	1978	Confrentes Compact control room Solid-state nuclear system protection system
ABWR	1996	Kashiwazaki-Kariwa 6 Reactor internal pumps Fine-motion control rod drives Advanced control room, digital and fiber optic technology Improved ECCS: high/low pressure flooders
ESBWR	Under review	TBD Natural circulation Passive ECCS

- Smaller-diameter fuel rods, longer in active fuel length and arranged in  $8 \times 8$  bundles within the same external outline as the previous  $7 \times 7$  design. This lowers the kilowatt rating per length of fuel and permits increased heat output per bundle.
- Improved control and instrumentation systems incorporating the latest solid-state electronics technology.
- Improved operator-machine interface systems for better control of the plant.

### 3.1.3 ABWR

Development of the ABWR took place during the 1980s under the sponsorship of the Tokyo Electric Power Company (TEPCO) (Figures 3.1 and 3.2). The stated purpose of the development effort was to design a BWR plant that included a careful blend of (1) the best features of worldwide operating BWRs, (2) available new technologies, and (3) new modular construction techniques. Safety improvements were, as always, the top priority. Anticipating the economic challenges that lay ahead, special attention was paid to systematically reduce the capital cost and incorporate features into the plant design that would make maintenance significantly easier and more efficient.

Development of the ABWR occurred in a series of steps. Phase 1 was a conceptual design study that determined the feasibility of the ABWR concept. Phase 2, in which most of the development took place, included more detailed engineering and the testing of new technologies and design features. The purpose of Phase 3 was to put the finishing touches on the design and systematically reduce capital costs, which proved to be a highly

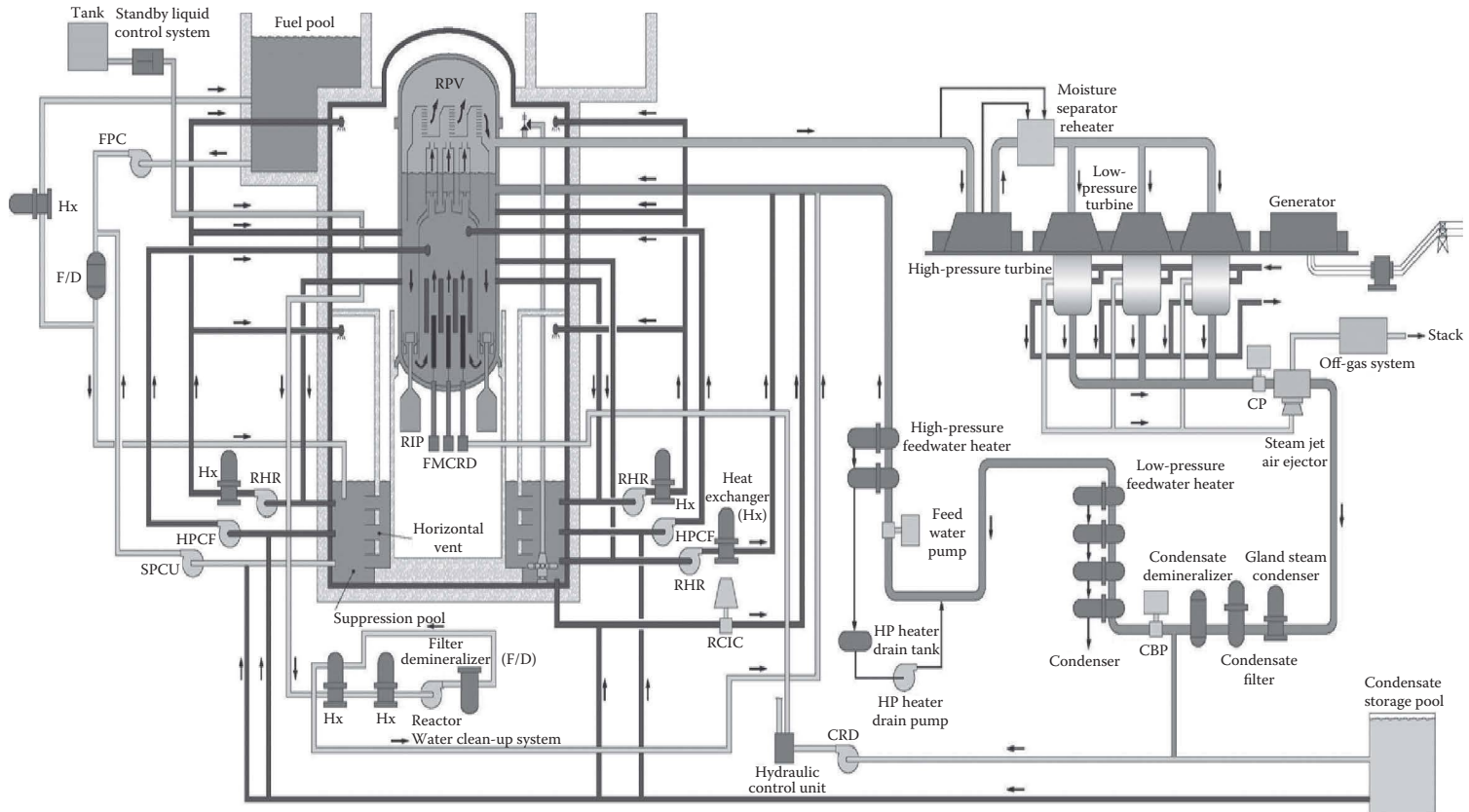
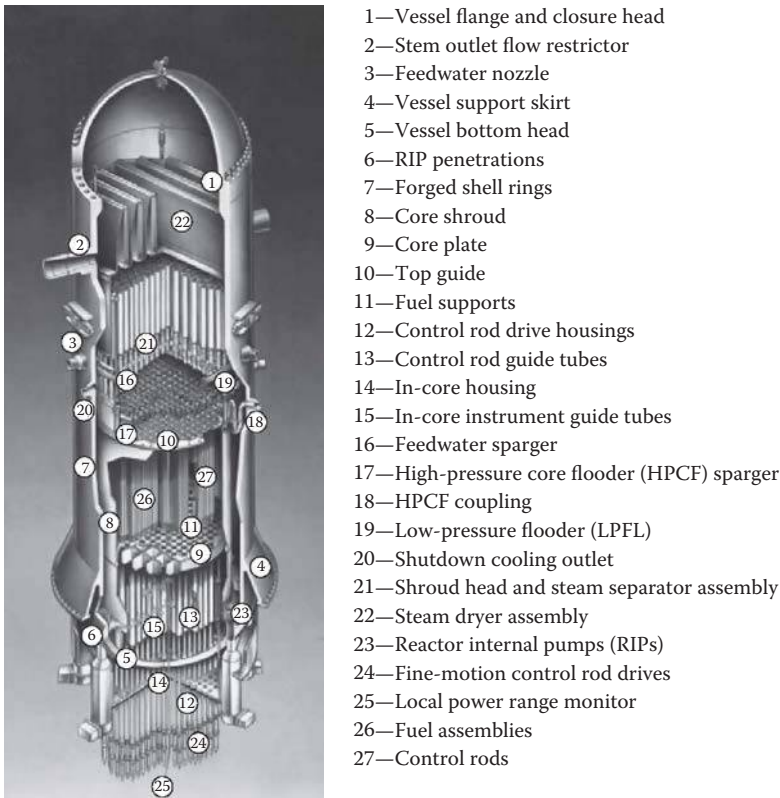


FIGURE 3.1  
ABWR major systems.



**FIGURE 3.2**  
 ABWR reactor assembly.

successful and, in hindsight, fortuitous endeavor. The development phases came to an end in 1988 when TEPCO announced that the next Kashiwazaki-Kariwa units to be constructed would be ABWRs.

With the selection of the ABWR for the K-6&7 projects, the detailed project engineering began. Licensing activities with the Japanese regulatory agency, the Ministry of International Trade and Industry (MITI), also started at this time and, interestingly, were conducted in parallel for some time with the review of the ABWR in the United States by the Nuclear Regulatory Commission (NRC). MITI and the NRC, in fact, held several meetings to discuss their respective reviews.

By 1991, the detailed design was essentially complete and MITI concluded its licensing review. An Establishment Permit, or license, was issued in May 1991. Excavation began later that year on September 17, bringing a decade of development work to a successful conclusion.

The key design objectives for the ABWR were established during the development program. The key goals, all of which were achieved, are as follows:

- Design life of 60 years
- Plant availability factor of 87%
- Less than one unplanned scram per year



- Refueling interval of 18–24 months
- Operating personnel radiation exposure limit <1 Sv/year
- Reduced calculated core damage frequency by at least a factor of 10 over previous BWRs (goal <10<sup>-6</sup>/year)
- Radwaste generation <100 m<sup>3</sup>/year
- Construction schedule of 48 months

### 3.1.4 Economic Simplified BWR

The economic simplified BWR (ESBWR) builds on the very successful ABWR technology and construction programs, as well as the simplified BWR (SBWR) development program (Figures 3.3 and 3.4). As of this publication production the ESBWR is being certified by the U.S. NRC. The key goals are

- Cost advantage over competing base load typical generating technologies
- Plant availability factor of 95%
- Design life of 60 years
- Less than one unplanned scram per year
- Refueling interval of 18–24 months
- Operating personnel radiation exposure limit <1 Sv/year
- Reduced calculated core damage frequency by at least a factor of 10 over previous BWRs (Goal <10<sup>-6</sup>/year)
- Radwaste generation less than that of the 10% best operating BWRs
- Construction schedule of 48 months
- 20% reduction in capital cost (\$/kWh) versus previous 1100 MWe class BWRs typically complex safety systems

Table 3.2 is a comparison table for the key features of the described product lines.

### 3.1.5 Summary Description

The direct cycle BWR system (Figure 3.5) is a steam generation and steam utilization system consisting of a nuclear core located inside a reactor vessel and a conventional turbine generator and feedwater supply system. Associated with the nuclear core are auxiliary systems to accommodate the operational and safeguard requirements and necessary controls and instrumentation. Water is circulated through the reactor core, producing saturated steam, which is separated from recirculation water, dried in the top of the vessel, and directed to the steam turbine generator. The turbine employs a conventional regenerative cycle with condenser deaeration and condensate demineralization. The basic heat balance for a BWR system is shown in Figure 3.6.

The steam produced by the nuclear core is, of course, radioactive. The radioactivity is primarily N16, a very short-lived isotope (half-life of 7 s) so that the radioactivity of the steam exists from the reactor vessel only during power generation. Carryover of long-lived radioactive particles by the steam supply to the turbine and condensate system is virtually nonexistent.

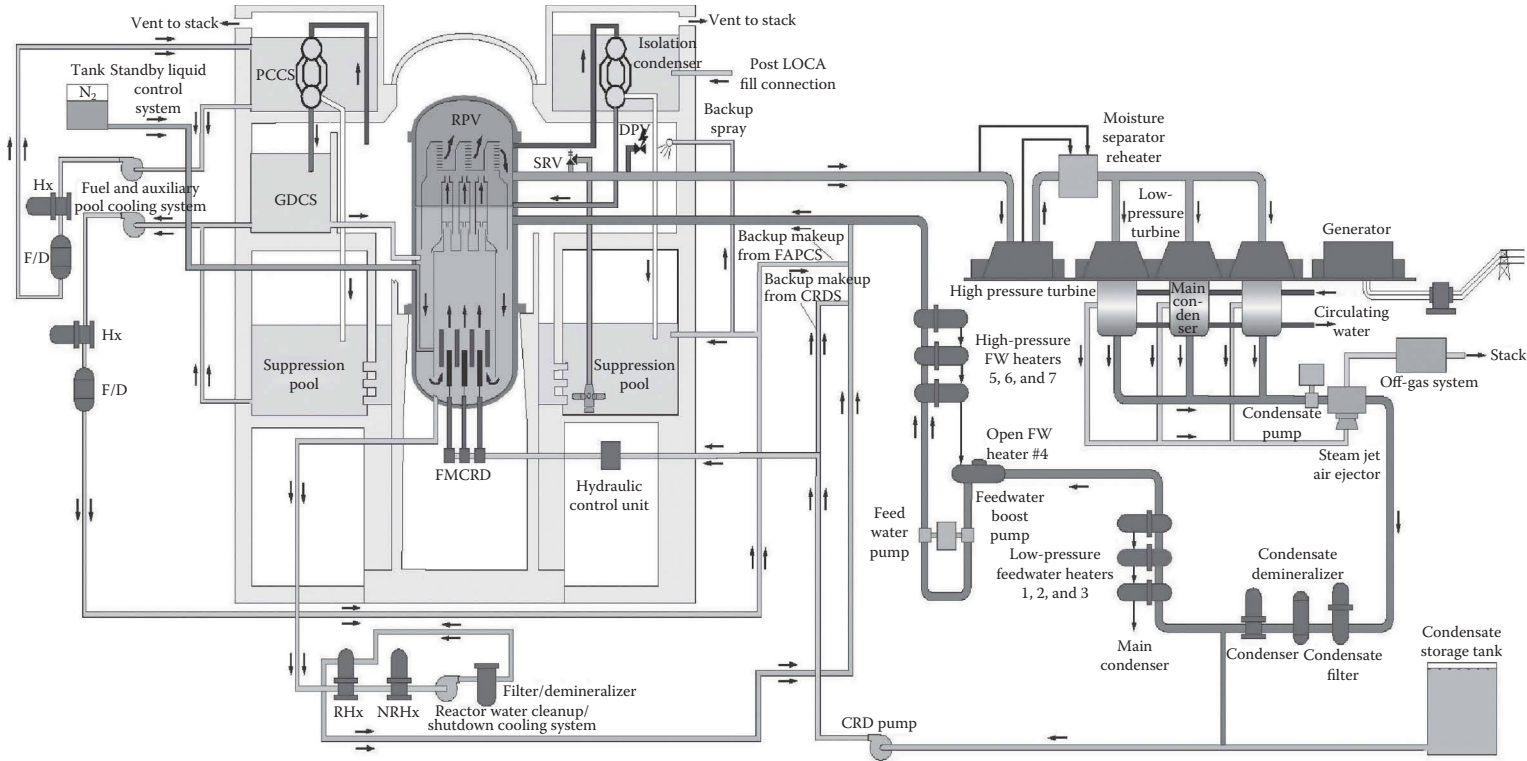


FIGURE 3.3  
ESBWR major systems.



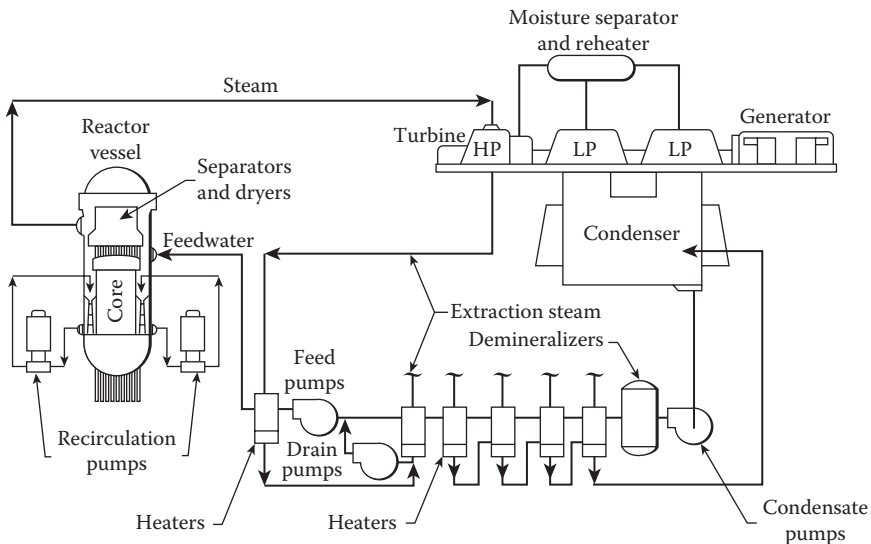
**FIGURE 3.4**  
ESBWR reactor pressure vessel and internals.

The nuclear core, the source of the heat, consists of fuel assemblies and control rods contained within the reactor vessel and cooled by the recirculating water system. A 1220-MWe BWR-6 core consists of 748 fuel assemblies and 177 control rod assemblies, forming a core array about 16 ft (4.9 m) in diameter and 14 ft (4.3 m) high. The power level is maintained or adjusted by positioning control rods up and down within the core. The BWR core power level is further adjustable by changing the recirculation flow rate through the core without changing control rod position. This unique BWR feature helps achieve the superior load-following capability of the BWR.

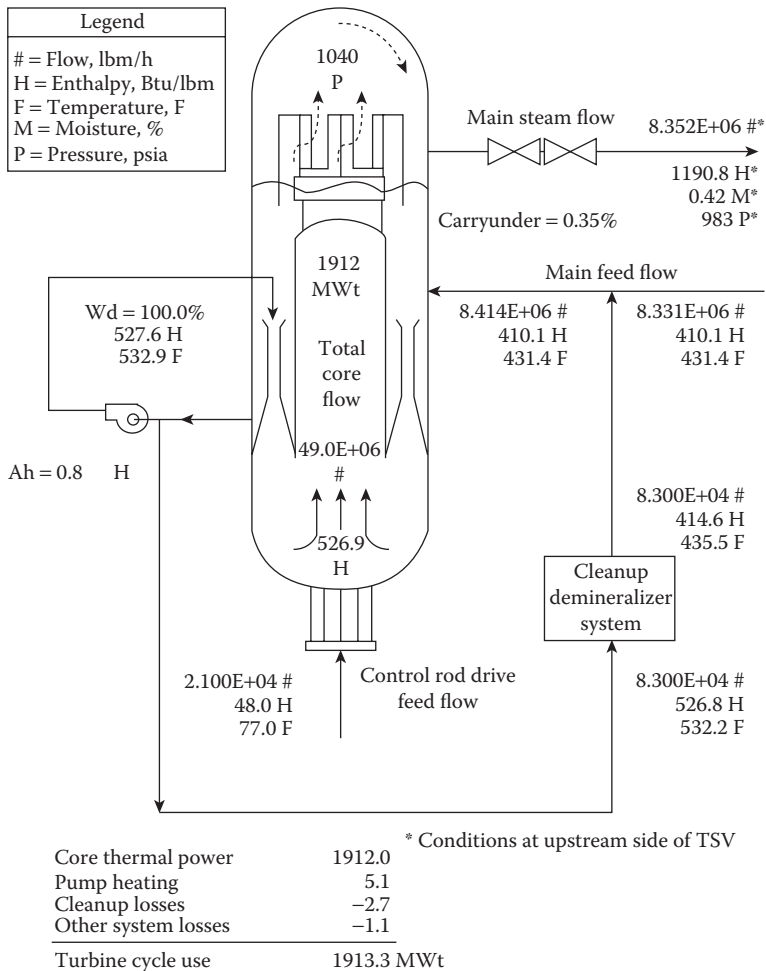
The BWR is the only light water reactor system that employs bottom-entry control rods. Bottom-entry and bottom-mounted control rod drives allow refueling without removal of control rods and drives and allow drive testing with an open vessel prior to initial fuel loading or at each refueling operation. The hydraulic control rod drive system, which incorporates

**TABLE 3.2**  
Comparison of Key ESBWR Features to Previous BWRs

Feature	BWR/6	ABWR	ESBWR
Recirculation system inside RPV	Two external loop recirculation system with jet pumps	Vessel-mounted reactor internal pumps	Natural circulation
Control rod drives	Locking piston CRDs	Fine-motion CRDs	Fine-motion CRDs
ECCS	Two-division ECCS plus HPCS	Three-division ECCS	Four-division, passive, gravity-driven
Reactor vessel	Welded plate	Extensive use of forged rings	Extensive use of forged rings
Primary containment	Mark III—large, low pressure, not inerted	Compact, inerted	Compact, inerted
Isolation makeup water	RCIC	RCIC	Isolation condensers, passive
Shutdown heat removal	Two-division RHR	Three-division RHR	Nonsafety system combined with RWCU
Containment heat removal	Two-division RHR	Three-division RHR	Passive
Emergency AC Alternate shutdown	Three safety-grade D/G Two SLC pumps	Three safety-grade D/G Two SLC pumps	Two nonsafety D/G Two SLE accumulators
Control and instrumentation	Analog, hardwired, single channel	Digital, multiplexed, fiber optics, multiple channel	Digital, multiplexed, fiber optics, multiple channel
In-core monitor calibration	TIP system	A-TIP system	Gamma thermometers
Control room	System-based	Operator task-based	Operator task-based
Severe accident mitigation	Not specifically addressed	Inerting, drywell flooding, containment venting	Inerting, drywell flooding, core catcher



**FIGURE 3.5**  
Direct cycle reactor system.



**FIGURE 3.6**  
Typical heat balance diagram.

mechanical locking of the rod at the selected position, provides positive driving and positioning of the control rods. Pressurized accumulators that provide a rod insertion force far greater than any gravity or mechanical system carry out rapid control rod insertion.

The core flow of a BWR is the sum of the feedwater flow and the recirculation flow (typical of any boiler). An important and unique feature of the BWR product line is the application of jet pumps inside the reactor vessel. These pumps generate about two-thirds of the recirculation flow within the reactor vessel. The jet pumps also contribute to the inherent safety of the BWR design under loss-of-coolant emergency conditions. Like most boilers, the BWR can deliver at least 10% power in a natural recirculation mode without operation of the recirculation pumps.

The BWR operates at constant pressure and maintains constant steam pressure similar to most fossil boilers. The integration of the turbine pressure regulator and control system in conjunction with the reactor water recirculation flow control system permits automated changes in steam flow to accommodate varying load demands on the turbine. Power changes of up to 25% of rated power can be accomplished automatically by recirculation

flow control alone, thus providing automatic load-following capability for the BWR without altering control rod settings.

The nuclear boiler system is supported by the specialized functions of its auxiliary system. Several auxiliary systems are used for normal plant operation:

- Reactor water cleanup (RWCU) system
- Shutdown cooling function of the residual heat removal (RHR) system
- Fuel building and containment pools cooling and filtering system
- Closed cooling water system for reactor service
- Radioactive waste treatment system

The following auxiliary systems are used as backup (standby) or emergency systems:

- Standby liquid control (SBLC) system
- Reactor core isolation cooling (RCIC) system
- RHR system
- Low-pressure coolant injection (LPCI)
- Steam condensing
- Containment spray
- Suppression pool cooling
- High-pressure core spray (HPCS) system
- Low-pressure core spray (LPCS) system
- Automatic depressurization

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## 3.2 Nuclear Boiler Assembly

### 3.2.1 Introduction

The nuclear boiler assembly consists of the equipment and instrumentation necessary to produce, contain, and control the steam power required by the turbine generator. The principal components of the nuclear boiler are as follows:

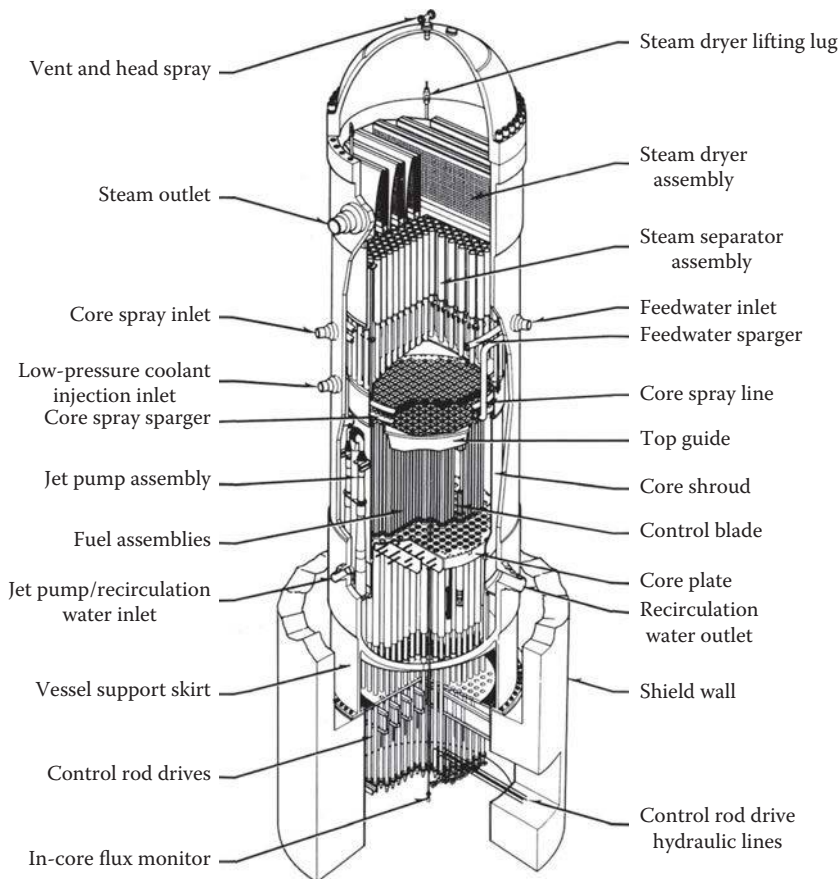
- *Reactor vessel and internals:* Reactor pressure vessel, jet pumps for reactor water recirculation, steam separators and dryers, core spray, and feedwater spargers and core support structure
- *Reactor water recirculation system:* Pumps; control and equipment isolation valves; piping and its suspension devices, restraints, and suppressors; used in providing and controlling core flow
- *Main steam lines:* Safety/relief and containment isolation valves; piping up to and including outboard containment isolation valve, and its restraints, suppressors, and guides
- *Control rod drive system:* Control rods, control rod drive mechanisms, and hydraulic system for insertion and withdrawal of the control rods

### 3.2.2 Reactor Assembly

The reactor assembly (Figure 3.7) consists of the reactor vessel, its internal components of the core, the shroud, the top guide assembly, the core plate assembly, the steam separator and dryer assemblies, and the jet pumps. The reactor assembly also includes the control rods, control rod drive housings, and control rod drives. Each fuel assembly that makes up the core rests on an orificed fuel support mounted on top of the control rod guide tubes. Each guide tube, with its fuel support piece, bears the weight of four assemblies and is supported by a control rod drive penetration nozzle in the bottom head of the reactor vessel. The core plate provides lateral guidance at the top of each control rod guide tube. The top guide provides lateral support for the top of each fuel assembly.

Control rods occupy alternate spaces between fuel assemblies and may be withdrawn into the guide tubes below the core during plant operation. The rods are coupled to control rod drives mounted within housings, which are welded to the bottom head of the reactor vessel. The bottom-entry drives do not interfere with refueling operations. A flanged joint is used at the bottom of each housing for ease of removal and maintenance of the rod drive assembly.

Except for the Zircaloy in the reactor core, these reactor internals are stainless steel or other corrosion-resistant alloys. All major internal components of the reactor can be removed



**FIGURE 3.7**  
Reactor assembly.

except the jet pump diffusers, the core shroud, the jet pump, and high-pressure coolant injection inlet piping. The removal of the top guide assembly and the core plate assembly is a major task, and it is not expected that these components would require removal during the life of the plant. The removal of other components such as fuel assemblies, in-core assemblies, control rods, and fuel support pieces is performed on a routine basis.

### **3.2.2.1 Reactor Vessel**

The reactor vessel is a pressure vessel with a single full-diameter removable head. The base material of the vessel is low alloy steel, which is clad on the interior except for nozzles with stainless steel weld overlay to provide the necessary resistance to corrosion. Since the vessel head is exposed to a saturated steam environment throughout its operating lifetime, stainless steel cladding is not used over its interior surfaces.

Fine-grained steels and advanced fabrication techniques are selected to maximize structural integrity of the vessel. BWR vessels have the lowest neutron exposure of any light water reactor, and the annulus space that carries recirculating water and feedwater downward between the core shroud and the vessel reduces radiation experienced by the vessel wall material. Vessel material surveillance samples are located within the vessel to enable periodic monitoring of exposure and material properties. Provisions are made for irradiating tensile and impact specimens for a program of monitoring and evaluating radiation-induced changes in vessel. Such programs have been conducted in most General Electric-designed power reactors, and considerable data have been accumulated on the performance of vessel materials after irradiation. The initial selection of high-quality materials, coupled with a continuing evaluation program, permits the vessel to meet the requirements of operability and safety throughout its design lifetime.

The vessel head closure seal consists of two concentric metal O-rings. This seal system has been demonstrated to perform without detectable leakage at all operating conditions. These conditions include cold hydrostatic testing, heating and cooling, and power operation. To monitor seal integrity, a leak detection system is used. Vessel supports, internal supports, their attachments, and adjacent shell sections are designed to take combined loads, including control rod drive reactions, earthquake loads, and jet reaction thrusts. The vessel is mounted on a supporting skirt, which is bolted to a concrete and steel cylindrical vessel pedestal, which is integrated with the reactor building foundation.

Many features have been incorporated in the design of the vessel and its associated piping to simplify the refueling operation. Steam outlet lines are welded to the vessel body, thereby eliminating the need to break flanged joints in the steam lines when removing the head for refueling. Another design feature is the seal between the vessel and the surrounding drywell, which permits flooding of the space (reactor well) above the vessel.

### **3.2.2.2 Core Shroud**

The shroud is a cylindrical, stainless steel structure that surrounds the core and provides a barrier to separate the upward flow through the core from the downward flow to the annulus. The discharge plenum of the core shroud is formed by the following connections; a flange at the top of the shroud mates with a flange on the top guide, which in turn mates with a flange on the steam separator assembly. The jet pump discharge diffusers penetrate the peripheral shelf of the shroud support below the core elevation to introduce the coolant into the inlet plenum. The peripheral shelf of the shroud support is welded to the vessel wall to prevent the jet pump outlet flow from bypassing the core and to form a chamber



around the core, which can be re-flooded in the event of a loss-of-coolant accident (LOCA). The shroud support carries the weight of the shroud, the steam separators, the jet pump system, and the seismic and pressure loads in normal and fault conditions of operation.

Two ring spargers, one for LPCS and the other for HPCS, are mounted inside the core shroud in the space between the top of the core and the steam separator base. The core spray ring spargers are provided with spray nozzles for the injection of cooling water. The core spray spargers and nozzles do not interfere with the installation or removal of fuel from the core. A nozzle for the injection of the neutron absorber (sodium pentaborate) solution is mounted below the core in the region of the recirculation inlet plenum.

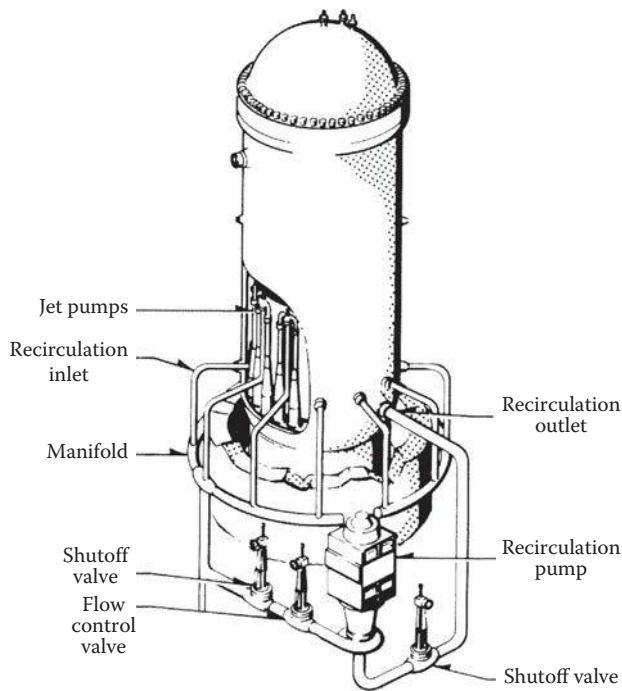
The steam separator assembly consists of a domed base on top of which is welded an array of standpipes with a three-stage steam separator located at the top of each standpipe. The steam separator assembly rests on the top flange of the core shroud and forms the cover of the core discharge plenum region. The seal between the separator assembly and core shroud flanges is a metal-to-metal contact and does not require a gasket or other replacement sealing devices. The separator assembly is bolted to the core shroud flange by a long hold-down bolt for ease of removal and extension above the separators. During installation, the separator base is aligned on the core shroud flange with guide rods and finally positioned with locating pins. The objective of the long-bolt design is to provide direct access to the bolts during reactor refueling operations with minimum-depth underwater tool manipulation during the removal and installation of the assemblies, which makes it unnecessary to engage threads in making up the shroud head. A tee-bolt engages in the top guide and its nut is tightened to only nominal torque. Final loading is established through differential expansion of the bolt and compression sleeve. The fixed axial flow-type steam separators have no moving parts and are made of stainless steel. In each separator, the steam–water mixture rising through the standpipe impinges on vanes giving the mixture a spin to establish a vortex wherein the centrifugal forces separate the water from the steam in each of the three stages. The steam then leaves the separator at the top and passes into the wet steam plenum below the dryer. The separated water exits the lower end of each stage of the separator and enters the pool that surrounds the standpipes to join the down comer annulus flow.

### **3.2.2.3 Steam Dryer**

The steam dryer assembly is mounted in the reactor vessel above the steam separator assembly and forms the top and sides of the wet steam plenum. Vertical guides on the inside of the vessel provide alignment for the dryer assembly during installation. Pads extending inward from the vessel wall support the dryer assembly, and it is held down in position during operation by the vessel head. Steam from the separators flows upward and outward through the drying vanes. These vanes are attached to top and bottom supporting members forming a rigid integral unit. Moisture is removed and carried by a system of troughs and drains to the pool surrounding the separators and then into the recirculation down comer annulus.

### **3.2.3 Reactor Water Recirculation System**

The function of the reactor water recirculation system (Figure 3.8) is to circulate the required coolant through the reactor core. The system consists of two loops external to the reactor vessel, each containing a pump with a directly coupled water-cooled (air–water) motor, a flow control valve, and two shutoff valves.



**FIGURE 3.8**  
BWR vessel arrangement for jet pump recirculation system.

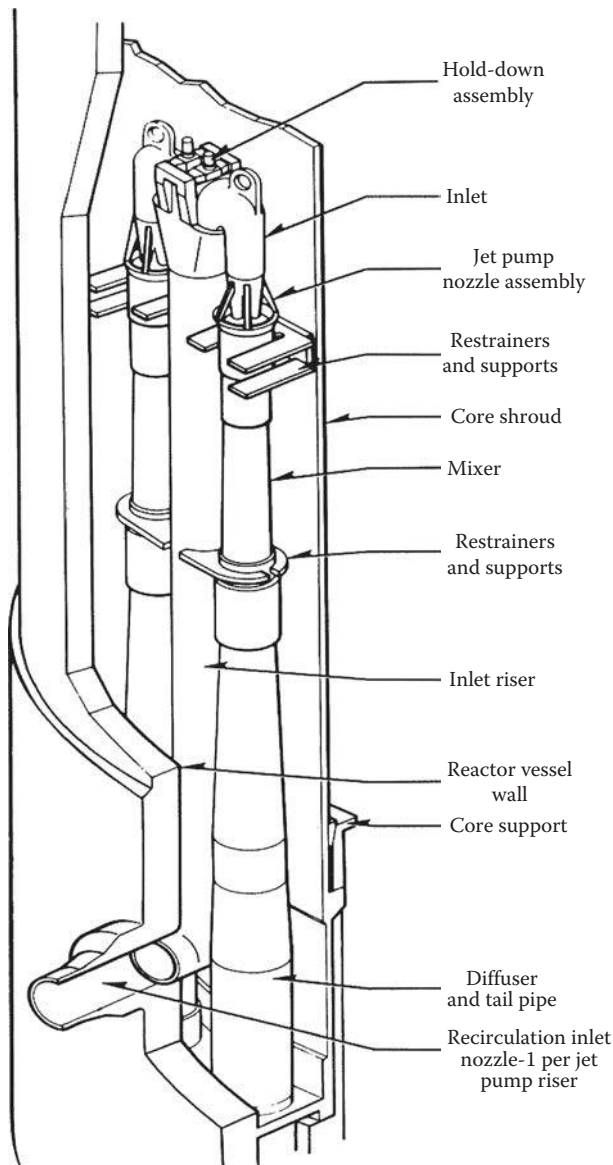
High-performance jet pumps located within the reactor vessel are used in the recirculation system. The jet pumps, which have no moving parts, provide a continuous internal circulation path for a major portion of the core coolant flow.

The recirculation pumps take suction from the downward flow in the annulus between the core shroud and the vessel wall. Approximately one-third of the core flow is taken from the vessel through the two recirculation nozzles. There, it is pumped at a higher pressure, distributed through a manifold to which a number of riser pipes are connected, and returned to the vessel inlet nozzles. This flow is discharged from the jet pump nozzle into the initial stage of the jet pump throat where, due to a momentum exchange process, it induces surrounding water in the down comer region to be drawn into the jet pump throat where these two flows mix and then diffuse in the diffuser, to be finally discharged into the lower core plenum. The jet pump diffusers are welded into openings in the core shroud support shelf, which forms a barrier between the lower plenum and the suction side of the jet pump. The flow of water turns upward, where it flows between the control rod drive guide tubes and enters into the fuel support where the flow is individually directed to each fuel bundle through the nose-piece. Orifices in each fuel support piece provide the desired flow distribution among the fuel assemblies. The coolant water passes along the individual fuel rods inside the fuel channel where it is heated and becomes a two-phase, steam–water mixture. The steam–water mixture enters a plenum located directly above the core and bounded by the separator dome, which opens to the separator array of fixed steam separators. The steam is separated from the water and passes through a dryer where any remaining water is removed. The saturated steam leaves the vessel through steam line nozzles located near the top of the vessel body end and is piped to the turbine. Water collected in the support tray of the dryer is routed through drain lines, joins the water leaving the separators, and flows downward in the annulus between the

core shroud and the vessel wall. Feedwater is added to the system through spargers located above the annulus and joins the downward flow of water. A portion of this downward flow enters the jet pumps and the remainder exits from the vessel as recirculation flow.

**3.2.3.1 Jet Pump Assembly**

The jet pumps (Figure 3.9) are located in the annular region between the core shroud and the vessel inner wall. Each pair of jet pumps is supplied driving flow from a single riser pipe. The jet pump assembly is composed of two jet pumps and contains no moving parts.



**FIGURE 3.9**  
Jet pump assembly.

Each jet pump consists of an inlet mixer, a nozzle assembly with five discharge ports, and a diffuser. The inlet mixer assembly, a replaceable component, is a constant-diameter section of pipe with a diffuser entrance section at the lower end and the drive nozzle at the upper end. The nozzle assembly can be removed by disconnecting the removable split flange. The jet pump diffuser is a gradual conical section terminating in a straight cylindrical section at the lower end that is welded into the shroud support. The overall length of the jet pumps is approximately 19 ft (5.8 m). Instrumentation monitors jet pump flow passages to ascertain their individual and collective flow rates under varying operating conditions.

### ***3.2.3.2 Operating Principle of the Jet Pump***

The driving flow enters the nozzle section at a high pressure and is accelerated to a high velocity because of the constriction at the nozzle outlet. The suction flow enters at a low pressure, which is further reduced as the flow is accelerated through the converging suction inlet nozzle. These two streams merge in the mixing section, where a pressure rise occurs because of the velocity profile rearrangement and the momentum transfer caused by the mixing. The rate of pressure rise decreases near the end of the mixing section because mixing is essentially completed. A diffuser is located downstream from the mixing section to slow the relatively high velocity mixed streams. This converts the dynamic head into static head. The jet pump system readily accommodates the full spectrum of flow rates required for load following.

### ***3.2.3.3 Safety Feature of the Jet Pump***

The safety feature of post-accident core flooding capability with a jet pump design allows flooding at no less than two-thirds of the core height. There is no recirculation line break that can prevent re-flooding of the core to the level at the top of the jet pump.

### ***3.2.3.4 Pumps and Motors***

The reactor recirculation pumps are vertically mounted, centrifugal, mechanical seal type and are constructed of stainless steel. The pumps operate at 25% of rated speed during startup and are powered from a low-frequency motor generator set. Following startup, the pumps operate at constant speed and are powered from auxiliary power.

The pump shaft seal assembly consists of multiple mechanical seals built into a cartridge or cartridges, which can be readily replaced with spare cartridges without removing the motor from the pump. Each seal carries an equal portion of the total pressure differential and is capable of sealing against maximum pump operating pressure. A throttle bushing located in the pump casing minimizes leakage in the unlikely event of a gross failure of all shaft seals. Cooling coils that circulate water from the closed cooling water system for reactor service controls the temperature of the seal cavity. The temperatures of the cavity and cooling water for each pump are recorded, and on high temperature activate an alarm in the control room.

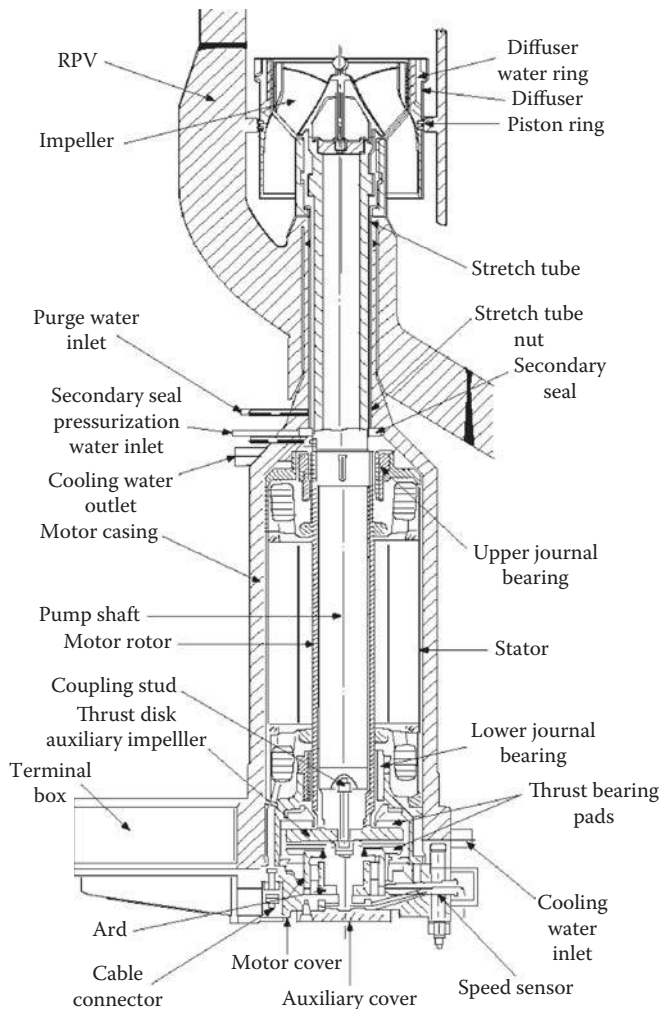
The drive motor for each pump is a vertical water-cooled (air-water heat exchanger), totally enclosed, three-phase, squirrel-cage induction motor designed to operate at constant speed. Cooling water to the air-water cooler for motor windings cooling and through coils in the bearing oil reservoir for motor bearing cooling is provided from the closed cooling water system. Temperature recorders and high-temperature alarms are located in the control room for motor windings, bearing oil reservoirs, and cooling water.

### 3.2.3.5 Valves and Piping

The recirculation loop piping is of welded construction. The piping, associated valves, and pumps are hung using constant-support hangers, thereby minimizing resultant stresses at the point of attachment to the reactor vessel. All recirculation piping is restrained to prevent pipe whipping as a result of jet action forces that may arise if a pipe break were to occur. The shutoff and bypass valves are motor-operated gate valves and the flow control valve is a ball-type with electrohydraulic actuator.

The flow control valve is on the discharge side of the pump. One shutoff valve is on the suction side of the pump and the other is downstream of the flow control valve. This allows maintenance in parallel with the refueling operation. No special reactor pressure vessel water level considerations are necessary. The stainless steel valves have double sets of valve stem packing to provide a highly reliable seal.

The ABWR design enhancements include elimination of the external recirculation loops and pumps and installation of reactor internal pumps (Figure 3.10). The ESBWR transitions



**FIGURE 3.10**  
Cross section of reactor internal pump.

from a forced recirculation mode to a natural circulation mode and eliminates the need for reactor recirculation pumps.

### 3.2.4 Main Steam Lines

Steam exits from the vessel several feet below the reactor vessel flange through four nozzles. Carbon steel steam lines are welded to the vessel nozzles and run parallel to the vertical axis of the vessel, downward to the elevation where they emerge horizontally from the containment. Two air-operated isolation valves are installed on each steam line, one inboard and one outboard of the containment penetration. The safety/relief valves are flange-connected to the main steam line for ease of removal for test and maintenance. A flow-restricting nozzle is included in each steam line as an additional engineered safeguard to protect against a rapid uncovering of the core in case of a break of a main steam line.

#### 3.2.4.1 Safety/Relief Valves

The safety/relief valves are dual-function valves discharging directly to the pressure suppression pool. The safety function provides protection against overpressure of the reactor primary system. The relief function provides power-actuated valve opening to depressurize the reactor primary system. The valves are sized to accommodate the most severe of the following two pressurization transient cases determined by analysis:

- Turbine trip from turbine design power, failure of direct scram on turbine stop valve closure, failure of the steam bypass system, and reactor scrams from an indirect scram
- Closure of all main steam line isolation valves, failure of direct scram based on valve position switches, and reactor scrams from an indirect scram

For the safety function, valves open at spring set point pressure and close when inlet pressure falls to 96% of spring set point pressure.

For the pressure-relief function, the valves are power-actuated manually from the control room or power-actuated automatically upon high pressure. Separate power circuits supply each valve. Valves that are power-actuated automatically upon high pressure close when pressure falls to a preset closing pressure. The pressure-relief function set point is below that for operation of valves for the safety (spring-actuated) function. By operating at the lower set point for the pressure-relief function, the re-closing set point of the valves provides a higher differential shutoff pressure than a spring reset valve, assuring leak tightness of the valves. The pressure-relief function may be used (in the event the main condenser is not available as a heat sink after reactor shutdown) to release steam generated by core decay heat until the RHR system steam condensing function is initiated.

To limit the cycling of relief valve to one valve subsequent to their initial actuation during a main steam line isolation event, two valves (one a backup to the other) have the feature of automatically changing normal set pressures (opening and closing) following their initial actuation at normal set pressures to a lower level, thereby limiting the pressure cycles to a level where the other relief valves will not reopen. In conjunction with these two valves, the set pressure for the closing of the other valves is changed automatically, which allows them to stay open longer before closing to accommodate pressure swings. Manual valve operation and resetting of valve set pressure to their normal levels after the transient is by the control room operator.

Selected safety/relief valves are associated with the automatic depressurization of the primary system under assumed LOCA conditions. These valves have two independent logic channels powered from different power sources, either of which can initiate depressurization. Valves open automatically and remain open until the pressure falls to a preset closure pressure. These valves open automatically upon signals of high drywell pressure and low reactor water level and confirmation of one LPCI function of the RHR system or LPCS system running. Initiation signals need not be simultaneous. The valves remain open until the primary system pressure is reduced to a point where the LPCI function of the RHR system and/or the LPCS system can adequately cool the core. The initiation of automatic depressurization is delayed from 90 to 120 s to allow the operator to terminate the initiation should the HPCS system initiation and acceptable reactor vessel level have been confirmed.

The valves used for automatic depressurization can be manually power actuated to open at any pressure. The signal for manual power actuation is from redundant control room switches from different power sources.

In the unlikely event that the RHR shutdown suction line is unavailable during reactor shutdown to cool reactor water and during the period when the LPCI function of the RHR system and/or the LPCS system pumps are injecting water into the reactor vessel, safety/relief valves used for automatic depressurization can be used to pass water from the reactor vessel to the suppression pool via valve discharge lines. For this to occur, the reactor vessel floods to a level above the vessel main steam line nozzles, selected safety/relief valves are opened from the control room to pass reactor water to the suppression pool.

Each steam line has two containment isolation valves, one inside and one outside the containment barrier. The isolation valves are spring-loaded pneumatic piston-operated globe valves designed to fail closed on loss of pneumatic pressure or loss of power to the pilot valves. Each valve has an air accumulator to assist in the closure of the valve upon loss of the air supply, electrical power to the pilot valves, and failure of the loaded spring. Each valve has an independent position switch initiating a signal into the reactor protection system scram trip circuit when the valve closes.

The isolation valves close upon (1) low water level in the reactor vessel, (2) high radiation from the steam line, (3) high flow rate in the main steam line, (4) low pressure at inlet to the turbine, (5) high ambient and differential steam line tunnel temperature (outside the containment), (6) low condenser vacuum (unless procedurally bypassed), and (7) high turbine building temperature. The signal for closure comes from two independent channels; each channel has two independent tripping sensors for each measured variable. Once isolation is initiated, valves continue to close and cannot be opened except by manual means. Independent remote manual switches located in the control room may each operate the valves. Lights in the control room indicate the position of the valve.

A shutoff valve is used in each steam line outboard of the external containment isolation valve and functions as a backup to the isolation valve. The shutoff valve is part of leakage control system to prevent possible release of nuclear steam which could leak through the main steam containment isolation valves following a LOCA. Independent containment inboard and containment outboard divisions are used to establish a pressurized barrier between the containment barrier and the environs. Out leakage is effectively eliminated and in leakage is directed into the containment from the pressurized volume. Both divisions are powered from auxiliary and standby AC power. While either of the two divisions is sufficient to establish the necessary pressure barrier, both are initiated in the control

room by a remote manual switch after it has been determined that a LOCA has occurred. The system will not actually initiate unless the pressure levels of the air supply and the reactor vessel are within the permissive interlock set points.

The main steam line isolation valves remain closed if the steam line pressure is greater than the air pressure interlock set point. When the interlock is cleared, air is admitted to raise the pressure in the main steam lines to a predetermined level to establish the containment pressure boundary.

### 3.2.5 Control Rod Drive System

Positive core reactivity control is maintained by the use of movable control rods interspersed throughout the core. These control rods thus control the overall reactor power level and provide the principal means of quickly and safely shutting down the reactor. The rods are moved vertically by hydraulically actuated, locking piston-type drive mechanisms. The drive mechanisms perform a positioning and latching function and a scram function with the latter overriding any other signal. The drive mechanisms are bottom-entry, upward-scramming drives which are mounted on a flanged housing on the reactor vessel bottom head. Here, they cause no interference during refueling and yet they are readily accessible for inspection and servicing. Hydraulic connections to the drive mechanism are made at ports in the face of the housing flange.

The control rod drive system consists of several locking-piston control rod drive mechanisms, a hydraulic control unit for each drive mechanism, a hydraulic power supply for the entire system, and instrumentation and controls with necessary interconnections. The locking-piston-type control rod drive mechanism is a double-acting hydraulic piston that uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion. The drive mechanism can position the rods at intermediate increments over the entire core length. The control rod can be uncoupled from below the vessel without removing the reactor vessel head, or with the vessel head removed for refueling, without removing the drive mechanism. Some of the advantages of the bottom-mounted drive arrangement are detailed below.

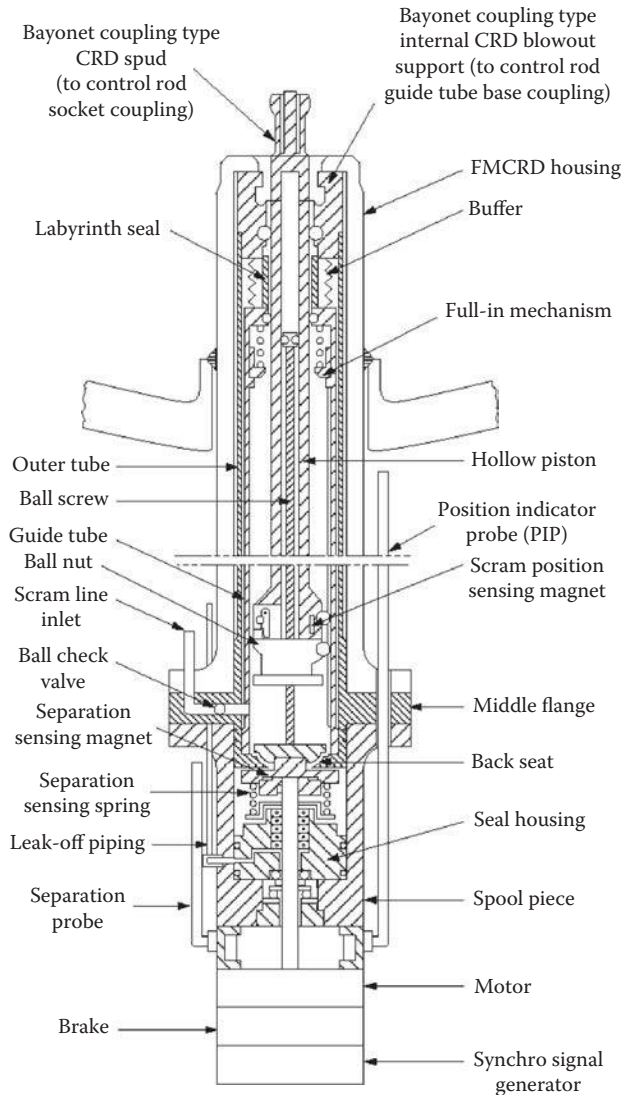
- The drives do not interfere with refueling and are operative even when the head is removed from the reactor. Furthermore, this location makes them more accessible for inspection and servicing. Such an arrangement makes maximum use of the water in the reactor as a neutron shield, while yielding the least possible neutron exposure to drive components.
- The locking piston drive provides the highest scram forces and operating force margins of all known types of drive mechanisms. This provides high operational reliability, particularly in the scram function.
- The use of water of reactor quality as the operating fluid eliminates the need for special hydraulic systems, with their inherent leakage and maintenance problems.
- The continuous inflow of high-purity water through the drives minimizes the contamination deposits within the drives from foreign material that may be in the reactor vessel.



- By using high-purity water as the operating fluid, the drives can use simple internal piston seals, which allow leakage into the reactor vessel. Dynamic shaft or push rod seals and their attendant systems and wear problems are eliminated.
- Control rod entry from below the core provides the best axial flux shaping and resultant fuel economy for the BWR.

The number of drives supplied with a reactor is selected to give the optimum power distribution in the core and to give the operator the maximum degree of control flexibility during startup, maneuvering, and flux shaping.

Enhancement of the ABWR and ESBWR converts from the complex mechanical-hydraulic control rod drive system to a simple fine motion control rod drive (Figure 3.11).



**FIGURE 3.11**  
Fine-motion control rod drive cross section.

### 3.3 Reactor Core Design

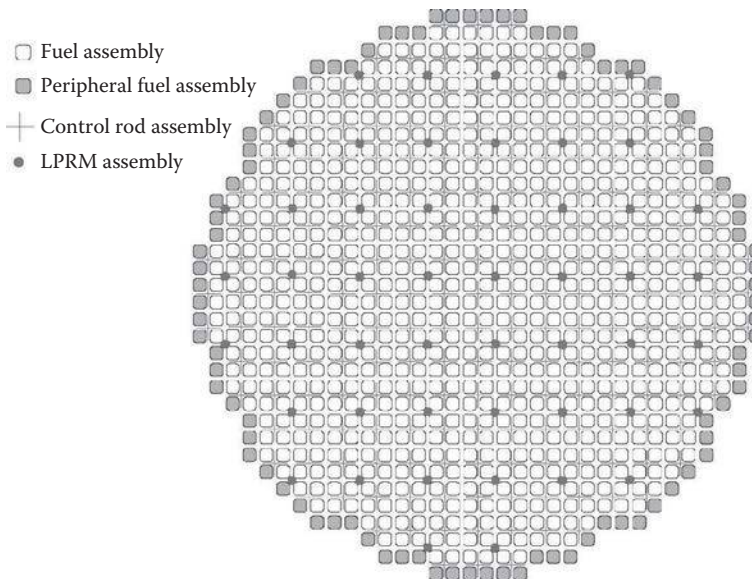
#### 3.3.1 Introduction

The design of the BWR core and fuel is based on the proper combination of many design variables and operating experience. These factors contribute to the achievement of high reliability, excellent performance, and improved fuel cycle economy.

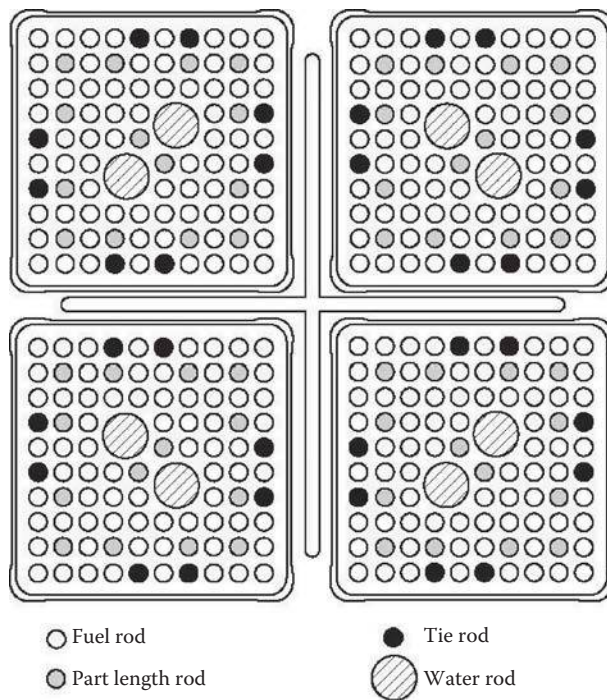
Discussed in this section are such design parameters as moderator-to-fuel volume ratio, core power density, thermal-hydraulic characteristics, fuel exposure level, nuclear characteristics of the core and fuel, heat transfer, flow distribution, void content, cladding stress, heat flux, and the operating pressure. Design analyses and calculations employed in this scope design have been verified by comparison with data from operating plants. General Electric continually evaluates this combination of design variables to be certain that changing conditions, which may significantly affect fuel cycle economics, are properly considered, and that the resulting final core design represents an optimum combination of the variables. The basic core configuration is shown in Figure 3.12, and a single fuel module cell is shown in Figure 3.13.

Several important features of the BWR core design are summarized below.

- The BWR core mechanical design is based on conservative application of stress limits, operating experience, and experimental test results. The moderate pressure level characteristics of a direct cycle reactor (approximately 1000 psia [6900 kPa]) reduce cladding temperatures and stress levels.
- The low coolant saturation temperature, high heat transfer coefficients, and neutral water chemistry of the BWR are significant, advantageous factors in minimizing Zircaloy temperature and associated temperature-dependent corrosion and hydride buildup. This results in improved cladding performance at long



**FIGURE 3.12**  
ABWR core configuration.



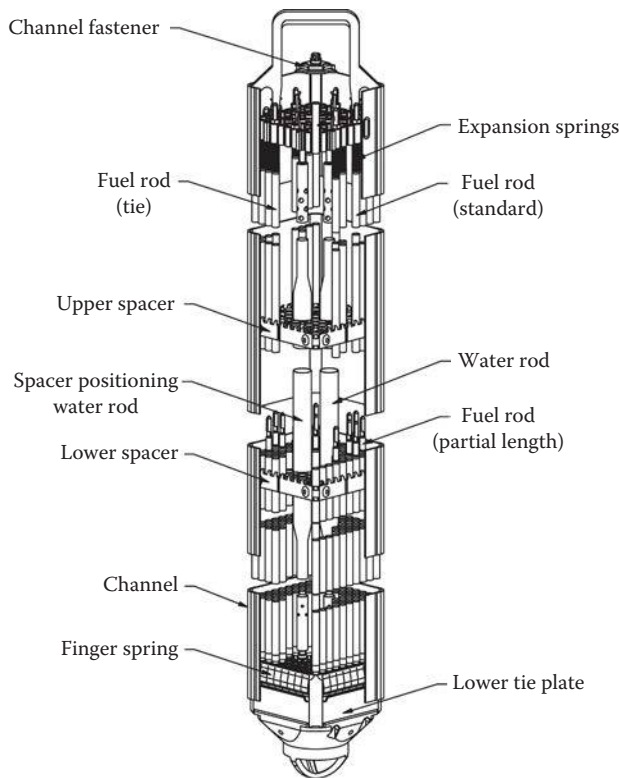
**FIGURE 3.13**  
Fuel module (cell).

exposures. The relatively uniform fuel cladding temperatures throughout the BWR core minimize migration of the hydrides to cold cladding zones and reduce thermal stresses.

- The basic thermal and mechanical criteria applied in the BWR design have been proven by irradiation of statistically significant quantities of fuel. The design heat fluxes and linear thermal outputs (approximate maximum of 13.4 kW/ft [44 kW/m]) are similar to values proven in fuel assembly irradiation.
- The design power distribution used in sizing the core represents a worst expected state of operation. Provisions for nonoptimal operation allow operational flexibility and reliability.
- The reactor is designed so that the peak bundle power at rated conditions is significantly less than the critical power limit.
- Because of the large negative moderator density (void) coefficient of reactivity, the BWR has a number of inherent advantages. These are the use of coolant flow as opposed to control rods for load following, the inherent self-flattening of the radial power distribution, the ease of control, the spatial xenon stability, and the ability to override xenon to follow load.

### 3.3.2 Core Configuration

The reactor core of the BWR is arranged as an upright cylinder containing many fuel assemblies and located within the reactor vessel. The coolant flows upward through the core. Important components of this arrangement are described in the following sections.



**FIGURE 3.14**  
GE14 fuel assembly.

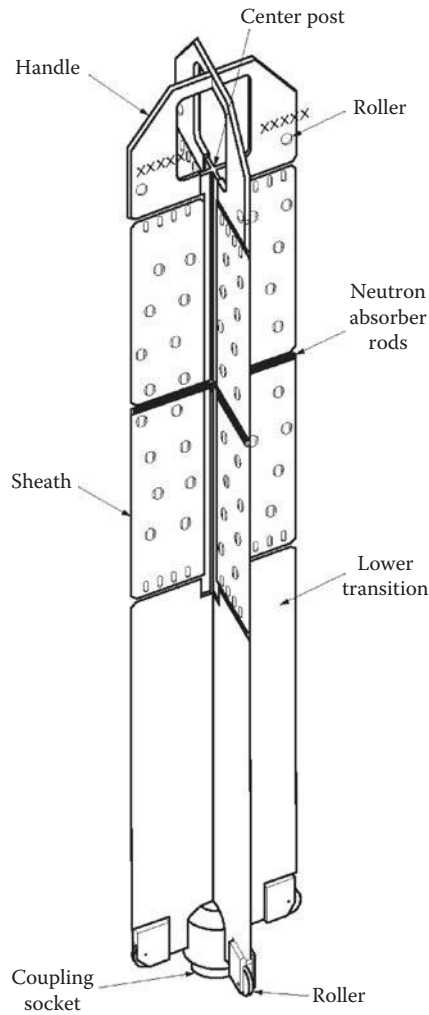
### 3.3.3 Description of Fuel Assembly

The BWR core comprises essentially only two components: fuel assemblies (Figure 3.14) and control rods (Figure 3.15).

#### 3.3.3.1 Fuel Rod

A fuel rod consists of  $\text{UO}_2$  pellets and a Zircaloy 2 cladding tube.  $\text{UO}_2$  pellets are manufactured by compacting and sintering  $\text{UO}_2$  powder in cylindrical pellets and grinding to size. The immersion density of the pellets is approximately 95% of theoretical  $\text{UO}_2$  density.

A fuel rod is made by stacking pellets into a Zircaloy 2 cladding tube, which is evacuated, back-filled with helium under a pressure of three atmospheres, and sealed by welding Zircaloy end plugs in each end of the tube. The Zircaloy tube is 0.483 in. (12.3 mm) in diameter, 160-1/4 in. (4.07 m) long, with a 32-mil (0.81 mm) wall thickness. The pellets are stacked to an active height of 150 in. (3.8 m), with the top 9.5 in. (241 mm) of tube available as a fission gas plenum. A plenum spring is located in the plenum space to exert a downward force on the pellets; this plenum spring keeps the pellets in place during the pre-irradiation handling of the fuel bundle. The selected dimensions result in a 9-mil (0.23 mm) diameter gap between the pellet and the tube.



**FIGURE 3.15**  
ABWR control rod.

### 3.3.3.2 Design Basis of Fuel Rods

The BWR fuel rod is designed as a pressure vessel. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, is used as a guide in the mechanical design and stress analysis of the fuel rod.

The rod is designed to withstand the applied loads, both external and internal. The fuel pellet is sized to provide sufficient volume within the fuel tube to accommodate differential expansion between fuel and cladding. Overall fuel rod design is conservative in its accommodation of the mechanisms affecting fuel in a BWR environment.

### 3.3.3.3 Fuel Bundle

Each fuel bundle contains 100 rods, which are spaced and supported in a square ( $10 \times 10$ ) array by lower and upper tie plates. The lower tie plate has a nosed piece, which fits into the

fuel support piece and distributes coolant flow to the fuel rods. The upper tie plate has a handle for transferring the fuel bundle. Both tie plates are fabricated from stainless steel and are designed to satisfy flow and mechanical strength considerations. Mechanically, these parts are designed to stay within the yield strength of the material during normal handling operations.

Three types of rods are used in a fuel bundle: tie rods, water rods, and standard fuel rods. The eight tie rods in each bundle have threaded end plugs which screw into the lower tie plate casting and extend through the upper tie plate casting. A stainless steel hexagonal nut and locking tab is installed on the upper end plug to hold the assembly together. These tie rods support the weight of the assembly only during fuel handling operations when the assembly hangs by the handle; during operation, the lower tie plate supports the fuel rods. Two rods in the interior foursome within each bundle (diagonally opposite the control blade) are water rods, that is, tubes of Zr-2 cladding without  $\text{UO}_2$  fuel. Small holes are located at the lower and upper ends, allowing water to be driven through the rod, thus introducing moderating material within the bundle interior. One water rod also serves as the spacer-positioning rod, being mechanically locked to each of the seven spacer assemblies, thereby fixing the axial position of each spacer. The fuel rod spacers are equipped with Inconel springs to maintain rod-to-rod spacing. The remaining 54 fuel rods in a bundle are standard rods having a single tube of fuel pellets the same length as the tie rods. The end plugs of the spacer-capture rod and the standard rods have pins, which fit into anchor holes in the tie plates. An Inconel expansion spring located over the top end plug pin of each fuel rod keeps the fuel rods seated in the lower tie plate while allowing them to expand axially by sliding within the holes in the upper tie plate to accommodate differential axial thermal expansion.

The initial core has an average enrichment ranging from approximately 1.7 wt% U-235 to approximately 2.0 wt% U-235 depending on initial cycle requirements. Individual fuel bundle enrichments in the initial core are of three or four average enrichments, ranging from that of natural uranium, 0.711 wt% U-235, to a maximum of approximately 2.2 wt% U-235. The design of the initial core achieves an optimal balance of fuel economy, operating margins, and ease of transition to equilibrium cycle refueling.

Different U-235 enrichments are used in fuel bundles to reduce local power peaking. Low enrichment uranium rods are used in corner rods and in the rods nearer the water gaps; higher enrichment uranium is used in the central part of the fuel bundle. Selected rods in each bundle are blended with gadolinium burnable poison. The fuel rods are designed with the characteristics described below.

*Poison:* A material that absorbs neutrons unproductively and hence removes them from the fission chain reaction in a reactor, thereby decreasing its reactivity.

*Mechanical end fittings, one for each of the enrichments:* End fittings are designed so that it is not mechanically possible to complete assembly of a fuel bundle with any high-enrichment rods in positions specified to receive a low enrichment.

### 3.3.3.4 Bundle Features

The design has two important features:

- The bundle design places minimum external forces on a fuel rod; each fuel rod is free to expand in the axial direction.
- The unique structural design permits the removal and replacement, if required, of individual fuel rods.

### 3.3.3.5 Fuel Channel

A fuel channel encloses the fuel bundle; the combination of a fuel bundle and a fuel channel is called a fuel assembly. The channel is a square-shaped tube fabricated from Zircaloy 4; its outer dimensions are 5.455 in. (138.6 mm) × 5.455 in. (138.8 mm) × 166.9 in. (4.2 m) long. The reusable channel makes a sliding seal fit on the lower tie plate surface. The channel fastener assembly (consisting of a spring and a guard) attaches the reusable channel to the upper tie plate and a capscrew secured by a lock washer. Spacer buttons are located on two sides of the channel to properly space four assemblies within a core cell. The fuel channels direct the core coolant flow through each fuel bundle and also serve to guide the control rod assemblies.

The use of the individual fuel channel greatly increases operating flexibility because the fuel bundle can be separately orificed and thus the reload fuel design can be changed to meet the newest requirements and technology. The channels also permit fast in-core sampling of the bundles to locate possible fuel leaks.

### 3.3.4 Core Design

The reactor core is designed to operate at rated power with sufficient design margin to accommodate changes in reactor operations and reactor transients without damage to the core. In order to accomplish this objective the core is designed, under the most limiting operating conditions and at 100% of rated power, to meet the bases detailed below.

- The maximum linear heat generation rate, in any part of the core, is <13.4 kW/ft (44 kW/m).
- Less than 0.1% of the core experiences transition boiling during the worst expected transient.

#### 3.3.4.1 Power Distribution

The design power distribution is divided into several components for convenience. The relative assembly power peaking factor is the maximum fuel assembly average power divided by the reactor core average assembly power. The axial power peaking factor is the maximum heat flux of a fuel assembly divided by the average heat flux in that assembly. The local power peaking factor is the maximum fuel rod heat flux at a horizontal plane in an assembly divided by the average fuel rod heat flux at that plane. Peaking factors vary throughout an operating cycle, even at steady-state full power operation, because they are affected by withdrawal of control rods to compensate for fuel burnup.

#### 3.3.4.2 Axial Distribution

Because of the presence of steam voids in the upper part of the core, there is a natural characteristic for a BWR to have the axial power peak in the lower part of the core. During the early part of an operating cycle, bottom-entry control rods permit a reduction of this axial peaking by locating a larger fraction of the control rods in the lower part of the core. At the end of an operating cycle, the higher accumulated exposure and greater depletion of the fuel in the lower part of the core reduces the axial peaking. The operating procedure is to locate control rods so that the reactor operates with approximately the same axial power shape throughout an operating cycle.

### 3.3.4.3 Relative Assembly Power Distribution

The maximum-to-average fuel bundle peaking or radial distribution is reduced in a BWR core because of greater steam voids in the center bundles of the core. A control rod operating procedure is also used to maintain approximately the same radial power shape throughout an operating cycle.

### 3.3.4.4 Local Power Distribution

The local power distribution is reduced by the use of several uranium enrichments in a fuel bundle. Lower uranium enrichments are located near the water gaps, and higher enrichments are located in the center of a fuel bundle.

### 3.3.4.5 Core Thermal Hydraulics

*Central UO<sub>2</sub> temperature:* The maximum UO<sub>2</sub> temperature will occur in new fuel operating at the maximum linear heat generation rate of 13.4 kW/ft (44 kW/m). Based on published conductivity data, the maximum temperature is approximately 3400°F (1871°C).

*Core orificing:* Fixed orifices accomplish control of core flow distribution among the fuel assemblies. These orifices are located in the fuel support pieces and are not affected by fuel assembly removal and replacement.

The core is divided into two orifice zones. The outer zone of fuel assemblies, located near the core periphery, has more restrictive orifices than the inner zone, so flow to the higher power fuel assemblies is increased. The orificing of all fuel assemblies increases the flow stability margin.

Three types of boiling heat transfer must be considered in defining thermal limits: nucleate boiling, transition boiling, and film boiling. Nucleate boiling is the extremely efficient mode of heat transfer in which the BWR is designed to operate. Transition boiling is manifested by an unstable fuel cladding surface temperature which rises suddenly as steam blanketing of the heat transfer occurs, then drops to the nucleate boiling temperature as the steam blanket is swept away by the coolant flow, then rises again. At still higher bundle powers, film boiling occurs, which results in higher fuel cladding temperatures. The cladding temperature in film boiling (and possibly the temperature peaks in transition boiling) may reach values that could cause weakening of the cladding and accelerated corrosion. Overheating is conservatively defined as the onset of the transition from nucleate boiling. The core and fuel design basis has been defined, accommodating uncertainties such that margin is maintained between the most limiting operating condition and the transition boiling condition at all times in core life.

### 3.3.4.6 Thermal-Hydraulic Analysis

A computer program is used to analyze the thermal and hydraulic characteristics of the reactor core as a whole. The geometric, hydraulic, and thermal characteristics of the core design are represented, including number of fuel assemblies in each orifice zone of the reactor core, fuel assembly dimensions, friction factors and flow restrictions, and the flow characteristics of the fuel orifices, inlet plenum region of the reactor, along with bypass and leakage flow paths around the fuel assembly channels. Individual cases are analyzed by providing reactor power, flow, inlet enthalpy, and appropriate power distribution factors as input to the above computer program. The output of the



program includes the calculated flow distribution among the several channel types and a detailed analysis of the heat fluxes, steam quality, void fraction, and maximum critical power ratio (MCPR) at as many as 24 axial nodes for the average and peak power fuel assemblies in each orifice zone.

Comparisons of the analytical models used with fuel assembly design details such as fuel-rod-to-fuel-rod and fuel-rod-to-fuel-assembly-channel clearances and spacer configurations have been made to ensure that the computer programs adequately represent the actual core and fuel design, and that design correlations are applicable.

In addition, fuel thermal design calculations, including calculation of fuel rod temperature,  $\text{UO}_2$  pellet thermal expansion characteristics, and rate of  $\text{UO}_2$  swelling due to irradiation have been performed. Thermal effects of irradiation, including reduction in local power peaking factor due to U-235 depletion, buildup of plutonium near the surface of the pellet, and effect of gap width and gas composition on gap conductance, have been considered in confirming that the thermal-hydraulic performance objectives will be met.

### 3.3.4.7 Core Nuclear Characteristics

Nuclear calculations are based on nuclear data selected from the best current sources of information throughout the nuclear industry and on mathematical computer codes developed by General Electric for the BWR.

#### 3.3.4.7.1 Reactivity Coefficients

In a BWR, two reactivity coefficients are of primary importance: the fuel Doppler coefficient and the moderator density reactivity coefficient. The moderator density reactivity coefficient may be broken into two components: that due to temperature and that due to steam voids.

#### 3.3.4.7.2 Fuel Doppler Reactivity Coefficient

As in all light water-moderated and low-enrichment reactors, the fuel Doppler reactivity coefficient is negative and prompt in its effect, opposing reactor power transients. When reactor power increases,  $\text{UO}_2$  temperature increases with minimum time delay, resulting in higher neutron absorption by resonance capture in the U-238.

#### 3.3.4.7.3 Moderator Density Reactivity Coefficient

During normal plant operations, the steam void component of the moderator density reactivity coefficients is of prime importance. The steam void component is large and negative at all power levels. At full rated power, the steam voids are equivalent to approximately 3% reactivity.

- The fuel assembly design is such that the moderator density reactivity coefficient of the water within the fuel channel is negative for all conditions of operation. The in-channel moderator coefficient is smallest at the cold, zero power condition.
- The large and negative moderator density coefficient at operating power levels is due to the steam void effect. This steam void effect results in the operating advantages listed below.
- *Xenon override capability*: Because the steam void reactivity effect is large compared with xenon reactivity, the BWR core has the excellent capability of overriding the xenon effect, thereby increasing power after a power decrease.

- *Xenon stability*: The steam void reactivity is the primary factor in providing the high xenon stability characteristic.
- *Load changing by flow control*: Because the fuel Doppler reactivity opposes a change in load, the void effect must be and is larger than the fuel Doppler effect to provide load changing capability by flow (or moderator density) control.
- *Thermal-hydraulic stability*: The negative void effect is an important contributor to reactor thermal-hydraulic stability.

### 3.3.4.8 Reactivity Control

The movable boron-carbide ( $B_4C$ ) control rods are sufficient to provide reactivity control from the cold shutdown condition to the full load condition. Additional reactivity control in the form of solid burnable poison is used only to provide reactivity compensation for fuel burnup or depletion effects.

The movable control rod system is capable of maintaining the reactor in a subcritical condition when the reactor is at ambient temperature (cold), zero power, zero xenon, and with the strongest control rod fully withdrawn from the core, in order to provide greater assurance that this condition can be met in the operating reactor, the core design is based on calculating a reactivity less than 0.99%, or a 1% margin on the *stuck rod* condition.

Supplementary solid burnable poisons are used to assist in providing reactivity compensation for fuel burnup. For all operating cycles, the supplementary control is provided by gadolinium mixed into a portion of the  $UO_2$  reload fuel rods.

### 3.3.4.9 Margin between Operating Limits and Damage Limits

Two mechanisms that could result in fuel damage (i.e., perforation of the cladding) are

- Severe overheating of the fuel cladding
- Fracture of the fuel cladding due to excessive strain resulting from  $UO_2$  thermal expansion

Although significant weakening of the fuel cladding due to overheating is not expected to occur until well into the film boiling region, fuel damage is conservatively defined as the onset of transition boiling. This, by definition, corresponds to  $MCPR = 1.00$ . In addition to this limit, a statistical margin of approximately 6% is made to allow for the various uncertainties in predicting and detecting the actual boiling state. So, during the worst expected transient, the  $MCPR$  is not permitted to go below a value of approximately 1.06. An additional margin for the effects of the worst transient produces the normal operating limit. A typical value for this operating limit is  $MCPR = 1.23$ . During full power operation, the fuel will typically operate with an  $MCPR > 1.30$ . The difference between the actual operating value of  $MCPR$  and the operating limit is termed the *operating margin*.

A value of 1% plastic strain of Zircaloy cladding is conservatively defined as the limit below which fuel damage is not expected to occur. Available data indicate that the threshold for damage in irradiated Zircaloy cladding is in excess of this value. The linear heat generation rate required to cause this amount of cladding strain is approximately 25 kW/ft (82 kW/m). The linear heat generation rate for the worst expected transient is approximately 16 kW/ft (52.5 kW/m). During normal full power operation, the maximum linear heat generation rate will not exceed 13.4 kW/ft (44 kW/m).

#### 3.3.4.10 Strain Localization

It was determined for  $7 \times 7$  BWR fuel that strain localization due to pellet-to-cladding interaction at pellet interfaces (ridging) and pellet cracks can cause a small but statistically significant number of fuel rod perforations during normal reactor operation. The fuel design improvements listed below have been made for  $8 \times 8$  BWR/6 fuel to reduce pellet-to-cladding localized strain.

- The fuel pellet length-to-diameter ratio is decreased from 2:1 to 1:1, which reduces ridging.
- The fuel pellet is chamfered, which reduces ridging.
- The maximum linear heat generation is decreased from 18.5 kW/ft (60.7 kW/m) to 13.4 kW/ft (44 kW/m), which reduces thermal distortion and ridging.
- The cladding heat treatment procedure is improved to reduce the variability of the cladding ductility.

#### 3.3.4.11 Reactor Stability

The large fuel time constant and inherent negative moderator feedback are major contributors to the stability of the BWR. The Doppler reactivity feedback appears simultaneously with a change in fuel temperature and opposes the power change that caused it, while heat conduction to water and the subsequent formation of steam voids must always transfer heat through the fuel material. Because the Doppler reactivity opposes load changes, it is desirable to maintain a large ratio of moderator-to-Doppler coefficient for optimum load following capability. The BWR takes advantage of its inherently large moderator-to-Doppler coefficient ratio by permitting a variation of coolant flow for load following.

Xenon instability is an oscillatory phenomenon of xenon concentration throughout the reactor that is theoretically possible in any type of reactor. If such a condition should occur, it can restrict load following performance, cause increased local power peaking in the core, and possibly reduce the fuel economics of the core. The BWR as designed by General Electric has characteristics that provide a large margin of damping to such oscillations. This is primarily brought about by the high negative power coefficient characteristic of the core. In addition, the use of in-core ion chambers for local monitoring of core conditions and for local reactivity adjustment brought about by the control rods and local steam void control provide complete knowledge of core conditions and adequate control capability. Xenon oscillations are local phenomena within the core. They are not evident when looking at core averaged values and, unless in-core instrumentation is provided, the presence of such oscillations may not be known until they have caused power peaking with possible core damage.

If the magnitude of the power coefficient of reactivity becomes too small, spatial xenon oscillations will occur and restrict reactor load following and performance. Even in the stable region, it is important to have well-damped power distributions and to select reactor load following variables which do not tend to encourage spatial xenon oscillations. Current BWR designs result in power coefficients well beyond the range of instability of xenon. This advantage of the BWR is of major importance for large, loosely coupled nuclear cores. Flow control further aids spatial xenon stability by providing a power shape, which remains relatively constant at varying reactor power levels.

The water-to-fuel volume ratio is determined from consideration of the reactivity coefficient for safe and stable operation. This ratio is selected to provide cold lattice coefficients,

which preclude detrimental startup transients. In parallel to the allowance of considerable margin in design for good load following and spatial xenon stability, the water-to-fuel volume ratio selected is close to the optimum for minimum fuel cycle costs.

### 3.3.5 Reactor Reactivity Control

#### 3.3.5.1 Control Rods

Control rods (Figure 3.15) using  $B_4C$  compacted in stainless steel tubes were introduced into service in 1961. Since then, this design has been the standard reference control element in all General Electric BWRs, and has replaced the 2% boron-steel rods previously used. Over the years,  $B_4C$  control rods have been produced routinely in quantity by tested manufacturing procedures. During the years since the first rods were placed in service, they have demonstrated excellent mechanical and nuclear performance.

The control rods perform dual functions of power distribution shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of selected patterns of rods. The rods, which enter from the bottom of the near-cylindrical reactor, are positioned in such a manner to counterbalance steam voids in the top of the core and affect significant power flattening. These groups of control elements, used for power flattening, experience a somewhat higher duty cycle and neutron exposure than the other rods.

The reactivity control function requires that all rods be available for reactor *scram* (prompt shutdown) or reactivity regulation. Control elements are therefore mechanically designed to withstand the dynamic forces resulting from a scram. They are connected to bottom-mounted, hydraulically actuated drive mechanisms that allow axial positioning for reactivity regulation or rapid scram insertion. The design of the rod-to-drive connection permits each blade to be attached or detached from its drive during refueling without disturbing the remainder of the control functions. The bottom-mounted drives permit the entire control function to be left intact and operable for tests with the reactor vessel open.

Control rods are cooled by the core leakage (bypass) flow. The core leakage flow is made up of recirculation flow that leaks through several leakage flow paths:

- Four holes in fuel assembly nosepiece (lower tie plate)
- The area between fuel channel and fuel assembly nosepiece
- The area between fuel assembly nosepiece tie and fuel support piece
- The area between fuel support piece and core plate
- The area between core plate and shroud
- Holes in the core plate for bypass flow control

#### 3.3.5.2 Control Rod Nuclear Characteristics

The control rod system is designed so that adequate shutdown capability is available at all times. To permit a margin for credible reactivity changes, the control system has the capability to shut down and maintain the core continuously subcritical with the maximum worth control rod fully withdrawn. This capacity is experimentally demonstrated when reactivity alterations are made to the reactor core. The use of mechanical control permits periodic tests on the core reactivity during refueling. Control rods are withdrawn adjacent to an inserted fresh fuel assembly to verify subcriticality and predicted excess reactivity

of the fuel. The control rod insertion rates on scram are sufficient to protect the reactor against damage in all transients, which are expected to occur during the life of the plant.

Control rods are used primarily for power distribution shaping and for shim control of long-term reactivity changes, which occur as a result of fuel irradiation. The flow control function, which is used to follow rapid load changes, reduces requirements on speed of control rod response and thus improves plant safety. Every 2–3 months, the control rod patterns are altered to provide more uniform fuel and control rod burnup. In normal daily operation, little control rod movement is required for depletion of reactivity. The resulting low frequency of control rod changes reduces the possibility of operator error.

With the normal control rod patterns required to maintain an acceptable power distribution in the operating core, an average control rod will be worth about 0.005 dk effective. The maximum worth of a rod in a typical power operation pattern will be about 0.01 dk effective. The notch increment dimensions and spacing of the rods are set to limit the reactivity insertion to about 0.0003 dk/k for any notch increment of control withdrawn. Preplanned withdrawal patterns and procedural patterns and procedural controls are used to prevent abnormal configurations yielding excessively high rod worth.

The velocity limiter is a mechanical device which is an integral part of the control rod assembly and protects against the low probability of a rod drop accident. It is designed to limit the free fall velocity and reactivity insertion rate of a control rod so that fuel damage would not occur. It is a one-way device in that control rod scram time (or fast insertion) is not significantly affected.

### 3.3.5.3 Supplementary Reactivity Control

The control requirements of the initial core are designed to be considerably in excess of the equilibrium core requirements because all of the fuel is fresh in the initial core. The initial core control requirements are met by use of the combined effects of the movable control rods and a supplemental burnable poison.

Only a few materials have nuclear cross sections suitable for burnable poisons. An ideal burnable poison must deplete completely in one operating cycle so that no poison residue exists to penalize initial U-235 enrichment requirements. It is also desirable that the positive reactivity from poison burnup matches the almost linear decrease in fuel reactivity from fission product buildup and U-235 depletion. A self-shielded burnable poison consisting of  $Gd_2O_3$  dispersed in a few selected fuel rods in each fuel assembly provides the desired characteristics.  $Gd_2O_3$  depletes as a cylinder with decreasing radius to provide a linear increase in reactivity. The concentration is selected so that the poison essentially depletes in the operating cycle. It is possible to improve power distributions by spatial distribution of the burnable poison.

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## 3.4 Reactor Auxiliary Systems

### 3.4.1 Introduction

Because the reactor is basically a water boiler, process systems are required which clean and control the chemistry of the water in the reactor vessel as well as protect the reactor core. Called the *reactor auxiliary systems*, these systems may be divided into two general

categories: systems necessary for normal nuclear boiler operations, including startup and shutdown; and systems that accommodate or provide backup in case of an abnormal condition.

Auxiliary systems used during normal plant operation include

- The RWCU system
- The fuel building and containment pools cooling and filtering system
- The closed cooling water system for reactor services
- The shutdown cooling function of the RHR system
- Radioactive waste treatment system
- Off-gas treatment system

Backup auxiliary systems used during abnormal plant operation include the RCIC system, the SBLC system, the steam condensing function of the RHR system (hot standby), and the suppression pool cooling function of the RHR system. Other process systems, commonly referred to as emergency core cooling systems (ECCS), are designed as safety systems to mitigate the consequences of postulated emergency situations that could otherwise lead to core damage and release of fission products to the environs. ECCS consists of the LPCI function of the RHR system, the HPCS system, the LPCS system, and automatic depressurization (blow down). The essential service water system and the area cooling systems (which service the areas where ECCS equipment is located) are also required during abnormal plant operation.

### 3.4.2 RWCU System

The purpose of the RWCU system is to maintain high reactor water quality by removing fission products, corrosion products, and other soluble and insoluble impurities. In addition, the system provides a means for water removal from the primary system during periods of increasing water volume.

The cleanup system is sized to process the water volume of the reactor system in approximately 3–3.5 h. The system can be operated during startup, shutdown, and refueling operation, as well as during normal plant operations. Water is removed from the reactor through the reactor recirculation pump suction line and returned through the feedwater line. Under normal operation, the water is removed at reactor temperature and pressure and pumped through regenerative and nonregenerative heat exchangers (NRHXs) where it is cooled, and then through the filter-demineralizer units. The flow continues through the shell side of the regenerative heat exchanger (RHX), where it is heated before returning to the reactor.

During startup and other times of increasing water volume, excess water is normally removed from the reactor by blowdown through the cleanup system to the main condenser, or alternately to the waste collector tank, or waste surge tank. During this operation, the return flow to the RHX is reduced, thereby reducing the cooling capability of this exchanger and correspondingly increasing the duty of the NRHX. The NRHX is designed to cool reactor water flow to the filter-demineralizer units to approximately 120°F during normal operation and reactor vessel blowdown. Cooling water is supplied to the NRHX by the closed cooling water system for reactor service.

The operation of the RWCS is controlled from the main control room. Filter resin backwashing and precoating operations are controlled from a local panel. The cleanup system

is isolated from the reactor automatically by closure of motor-operated isolation valves on any of the following signals:

- High temperature after the NRHX
- Low reactor water level
- SBLC solution injection
- High ambient temperature in the cleanup system equipment area
- High flow rate differential between system inlet and outlet
- High differential temperature across the system's ventilation system

### 3.4.3 Fuel Building and Containment Pools Cooling and Cleanup System

The fuel building and containment pools cooling and cleanup system accommodates the spent fuel cooling heat load as well as drywell heat transferred to the upper containment pool. The equipment for the cooling and cleanup systems consists of circulating pumps heat exchangers; filter-demineralizers; and the required piping, valves, and instrumentation. Pumping loops circulate pool water through the heat exchangers and fuel pool filters and returns the flow by discharging it through diffusers mounted in the fuel storage pool and in the containment pool. The suction for the circulating pumps is taken from the skimmer surge tank. The skimmer surge tank is fed from skimmers located at the top of these pools.

The upper containment pool has a shield wall with a removable gate between the reactor well, the fuel holding pool, and the fuel transfer pool. With the gate inserted in the slot, the upper containment pool can be drained for work at the pressure vessel flange level. With the pools full of water, the gates are removed during refueling operations to permit the transfer of fuel and equipment between pools.

The RHR system heat exchangers are also available to supplement the fuel pool cooling heat exchangers. RHR system heat exchangers are not normally required but may be needed if more than the normal number of spent fuel assemblies is stored in the pool. The system pumps and heat exchangers are located in the fuel building below the normal fuel pool water level. Heat exchangers are cooled by essential plant service water.

Essential service water may become radioactive because it collects corrosion products normally located in the fuel building. Pool water is usually filtered continually.

Because there are no drain connections at the bottom of the fuel storage pool, the spent fuel assemblies can never be exposed by an accidental valve opening or pipe break. Fuel is not stored in the upper containment pool during normal operation. A portable underwater vacuum system, similar to that used in swimming pools, can be used to clean pool walls, floors, and internals removed from the reactor vessel. Deposition at the water line of the pool walls is minimized by several surface skimmers. These devices remove a portion of the surface water and recycle it to the pool.

### 3.4.4 Closed Cooling Water System for Reactor Service

The closed cooling water system consists of a separate, forced circulation loop. This system uses water piped from the site service water source to provide a heat sink for selected nuclear system equipment. Its purpose is to provide a second barrier between the primary systems containing radioactive products and the service water system that is the final heat

sink, thereby eliminating the possibility of radioactive discharge into plant effluents that could result from heat exchanger leaks. The plant service water pumps provide coolant to the closed cooling water system for reactor service which in turn generally service the following equipment:

- Reactor recirculation pump seal coolers
- Reactor recirculation pump motor coolers
- Nonregenerative cleanup heat exchanger
- Clean sump coolers
- Sample coolers
- Drywell coolers
- Cleanup recirculation pump coolers
- Off-gas system glycol coolers
- Control rod drive supply pumps
- Radwaste concentrator condensers
- Radwaste concentrated waste tank
- Control rod drive supply pumps

Any possible radioactive leakage from the foregoing reactor equipment would be to, and would be confined in, the closed loop cooling water system, which is monitored continuously for radioactivity. A surge tank is used to accommodate system volume swell and shrinkage and to provide a means for adding makeup water and inhibitors.

The closed cooling water system design temperature depends on the maximum temperature of service water intake. The closed cooling water system satisfies the plant's full power load requirements. Extra cooling capability, with all spares operating, is adequate to handle plant startup duty.

#### 3.4.5 Emergency Equipment Cooling System

The emergency equipment cooling system (EECS) services certain equipment required for normal and emergency shutdown of the plant. The system provides cooling water for the RHR system pump motor and pump seal cooler and the HPCS and LPCS systems pump motors and pump seal coolers. Upon loss of normal ventilation, such as may occur upon loss of external AC power, the EECS provides ventilation and cooling for the HPCS and LPCS systems, the RHR system, and the reactor core ICS equipment as required preventing overheating. On failure of any single component, the EECS will service at least two RHR system pumps or one RHR system pump and the LPCS system pump, the HPCS system pump, and any standby core cooling system equipment being cooled by ventilation equipment being serviced by the EECS.

#### 3.4.6 SBLC System

The SBLC system is a redundant control system capable of shutting the reactor down from rated power operation to the cold condition in the postulated situation that the control rods cannot be inserted. The operation of this system is manually initiated from the reactor control room.



The equipment for the SBLC system is located in the reactor building and consists of a stainless steel storage tank; a pair of full capacity positive displacement pumps and injection valves; a test tank; and the necessary piping, valves, and instrumentation.

The SBLC system is adequate to bring the reactor from the hot operating condition to cold shutdown and to hold the reactor shutdown with an adequate margin when considering temperature, voids, Doppler effect, equilibrium, xenon, and shutdown margin. It is assumed that the core is operating at normal xenon level when injection of liquid control chemical is needed.

The liquid control chemical used is boron in the form of sodium pentaborate solution. It is injected into the bottom of the core where it mixes with the reactor coolant. The sodium pentaborate is stored in solution in the SBLC tank. Electric heaters automatically keep the solution above the saturation temperature. The system temperature and liquid level in the storage tank are monitored, and abnormal conditions are annunciated in the control room.

### 3.4.7 RCIC System

The RCIC system maintains sufficient water in the reactor pressure vessel to cool the core and then maintains the nuclear boiler in the standby condition in the event the vessel becomes isolated from the turbine steam condenser and from feedwater makeup flow. The system also allows for complete plant shutdown under conditions of loss of the normal feedwater system by maintaining the necessary reactor water inventory until the reactor vessel is depressurized, allowing the operation of the shutdown cooling function of the RHR system. The system delivers rated flow within 30 s after initiation. A *water leg* pump keeps the piping between the pump and the discharge shutoff valve full of water to ensure quick response and to eliminate potential hydraulic damage on system initiation.

Following a reactor scram during normal plant operation, steam generation continues at a reduced rate due to the core fission product decay heat. The turbine bypass system directs the steam to the main condenser, and the feedwater system provides makeup water required to maintain the reactor vessel inventory.

In the event the reactor vessel becomes isolated from the main condenser, the relief valves automatically (or by operator action from the control room) maintain vessel pressure within desirable limits. In the event feedwater becomes unavailable, the water level in the reactor vessel drops due to continued steam generation by decay heat. Upon reaching a predetermined low level, utilizing one-out-of-two-twice logic, the RCIC system is initiated automatically to maintain safe standby conditions of the isolated primary system. The turbine-driven pump supplies makeup water from one of the following sources capable of being isolated from other systems: the condensate storage tank (first source), the steam condensed in the RHR heat exchangers (second source), or the suppression pool (an emergency source). The turbine is driven with a portion of the decay heat steam from the reactor vessel and exhausts to the suppression pool. The makeup water is pumped into the reactor vessel through a connection to the reactor feedwater line.

A design flow functional test of the RCIC system may be performed during normal plant operation by drawing suction from the condensate storage tank and discharging through a full flow test return line to the condensate storage tank. The discharge valve to the reactor feedwater line remains closed during the test and reactor operation remains undisturbed. If the system requires initiation while in the test mode, the control system automatically returns to the operating modes. Cooling water for pump and turbine operations and for the lube oil cooler and the gland seal condenser is supplied from the discharge of the pump.

The RCIC system operates independently of auxiliary AC power, plant service air, or external cooling water systems. System valves and auxiliary pumps are designed to operate by DC power from the station batteries.

Two turbine control systems include a speed governor limiting the speed to its maximum operating level and a control governor with automatic set-point adjustment, which is positioned by demand signal from a flow controller. Manual operation of the control governor is possible when in the test mode but automatically repositioned by the demand signal from the controller if system initiation is required. The operator can select manual control of the governor, and adjust power and flow to match decay heat steam generation.

The turbine and pump automatically shut down upon:

- Turbine overspeed
- High water level in the reactor vessel
- Low pump suction pressure
- High turbine exhaust pressure
- Automatic isolation signal

The steam supply system to the turbine is automatically isolated upon:

- Large pressure drop across two pipe elbows in the steam supply line
- High area temperature
- Low reactor pressure (two-out-of-two logic)
- High pressure between the turbine exhaust rupture diaphragms (two-out-of-two logic)

### 3.4.8 ECCS

The ECCS comprises the LPCI function of the RHR system, HPCS and LPCS systems, and automatic depressurization of the primary system. The ECCS is designed to perform the following functions:

- Prevent fuel cladding fragmentation for any mechanical failure of the nuclear boiler system up to, and including, a break equivalent to the largest nuclear boiler system pipe
- Provide this protection by at least two independent, automatically actuated cooling systems
- Function with or without external (off-site) power sources
- Permit testing of all the ECCSs by acceptable methods, including, wherever practical, testing during power plant operations
- Provide this protection for long time periods and from secure sources of cooling water with the capability of dissipating the rejected heat for a minimum of 30 days

The aggregate of the ECCS is designed to protect the reactor core against fuel cladding damage (fragmentation) across the entire spectrum of line break accidents. The operational capability of the various ECCSs to meet functional requirements and performance objectives is outlined in the following paragraphs.

The operation of the ECCS network is automatically activated by the reactor protection system upon redundant signals that are indicating low reactor vessel water level or high drywell pressure or a combination of indicators showing low reactor vessel water level and high drywell pressure.

During the first 10 min following initiation of operation of the ECCS, any one of the following three combinations satisfies the functional requirements of the system objectives:

- The operation of the automatic depressurization function, the HPCS system, and two LPCI loops of the RHR system (failure of division 1)
- The operation of the automatic depressurization function, the HPCS system, the LPCS system, and one LPCI loop of the RHR system (failure of division 2)
- The operation of the automatic depressurization function, three LPCI loops of the RHR system, and the LPCS system (failure of division 3)

In the event of a break in a pipe that is part of the ECCS, any one of the following four combinations satisfies the functional requirement:

- The operation of the automatic depressurization function and two LPCI loops of the RHR system
- The operation of the automatic depressurization function, one LPCI loop of the RHR system, and the LPCS system
- The operation of the automatic depressurization function, the HPCS system, and one LPCI loop of the RHR system
- The operation of the automatic depressurization function, the HPCS system, and the LPCS system

A combination of the HPCS system or the LPCS system plus any two other ECCS pumps provides two phenomenological cooling methods (flooding and spraying).

After the first 10 min following the initiation of operation of the ECCS and in the event of an active or passive failure in the ECCS or its essential support system, one of the following two combinations satisfies the performance objectives and the requirement for removal of decay heat from the containment.

- Two LPCI loops of the RHR system with at least one heat exchanger and 100% service water flow
- Either the HPCS system or the LPCS system, one LPCI loop of the RHR system with one heat exchanger, and 100% service water flow

The separation of redundant equipment of the various systems that make up the ECCS is maintained to assure maximum operational availability. Electrical equipment and wiring for the engineered safeguard features of the ECCS are broken into segregated divisions, further assuring a high degree of redundancy.

The power for operation of the ECCS is from regular AC power sources. Upon loss of regular power, operation is from on-site standby AC power sources, and the standby diesel-generator set is capable of accommodating full capacity of the LPCI and spray function. The HPCS system is completely independent of external power sources, having its own diesel generator as shown.

The operation of ECCS pumps is also possible from a local key lock hand switch and from the control room but automatic signals pre-empt all others. In the event of normal power failure while the system is operating or in the process of going into operation, the system will restart from the standby sources. All system alarms annunciate in the control room.

Although the feedwater system is not considered a part of the ECCS, under some circumstances it could refill the vessel or at least maintain a water level, depending upon the location of the postulated break, for a given spectrum of break sizes. In the case of turbine-driven feedwater pumps, this additional coolant source would still be available from the electrically driven condensate pumps.

#### **3.4.8.1 HPCS System**

The purpose of the HPCS system is to depressurize the nuclear boiler system and to provide makeup water in the event of a loss of reactor coolant inventory. In addition, the HPCS system prevents fuel cladding damage (fragmentation) in the event the core becomes uncovered due to loss of coolant inventory by directing this makeup water down into the area of the fuel assemblies. The makeup water is jetted as a spray over the area of the fuel assemblies from nozzles mounted in a sparger ring located inside the reactor vessel above the fuel assemblies. The HPCS system is an integral part of the total design for ECC, which provides for adequate core cooling and depressurization for all rates of coolant loss from the nuclear boiler.

The HPCS system includes a sparger ring with spray nozzles located inside the reactor pressure vessel, a motor-driven pump, diesel generator, valves, piping, and instrumentation necessary to provide an operating system with the capability of being tested during plant operation.

Cooling water for the operation and testing of the HPCS system is from the condensate storage tank. Upon depletion of this supply, the system automatically transfers water from the containment suppression pool. Water inventory lost from the nuclear boiler system drains to the drywell to weir wall level and then into the suppression pool, thereby providing an inexhaustible supply of cooling water allowing continued operation of the HPCS system until the operator from the control room manually stops it. System piping and equipment are maintained full of condensate water at all times to avoid time delays in filling the lines and to avoid hydraulic hammer.

The HPCS system can operate independently of normal auxiliary AC power, plant service air, or the emergency cooling water system. Operation of the system is automatically initiated from independent redundant signals indicating low reactor vessel water level or high pressure in the primary containment. The system also provides for remote-manual startup, operation, and shutdown. A testable check valve in the discharge line prevents backflow from the reactor pressure vessel when the reactor vessel pressure exceeds the HPCS system pressure such as may occur during initial activation of the system. A low flow bypass system is placed into operation until pump head exceeds the nuclear system pressure and permits flow into the reactor vessel.

The HPCS system can be tested during normal plant operation or when the plant is shut down. During normal plant operation, pump suction is from the condensate storage tank with a full flow return line to the condensate storage tank. During plant shutdown, pump suction is from the primary containment pressure suppression pool with a full flow return line to the suppression pool. The control system provides for the automatic transfer to the service mode upon the presence of ECC demand signal.

The integrity of the piping internal to the reactor vessel is determined by comparing the difference in pressure between spray sparger and the bottom of the core area with the pressure drop across the core. An increase in this comparison initiates an alarm in the control room.

#### 3.4.8.2 LPCS System

The function of the LPCS system is to prevent fuel cladding damage (fragmentation) in the event the core is uncovered by the loss of coolant. The cooling effect is accomplished by directing jets of water down into the fuel assemblies from spray nozzles mounted in a sparger ring located above the reactor core. The system is an integral part of the total design for ECC, which provides for adequate core cooling for all rates of coolant loss from the nuclear boiler. The system goes into operation once the reactor vessel pressure has been reduced and the operation of the other systems of the ECCS prove inadequate to maintain the necessary water level in the reactor vessel at the reduced vessel pressure.

The LPCS system includes a sparger ring with spray nozzles located in the reactor vessel above the core, a motor-driven pump, motor-operated valves, piping, valves, and instrumentation necessary to provide a system for required operation with the capability of being tested. The system is connected to the containment suppression pool for its supply of water for cooling and connectable to the RHR system for testing and flushing. The elevation of the pump, with respect to the minimum water level of the suppression chamber, ensures adequate net positive suction head. The system pump is protected from overheating during operation against high reactor vessel pressure or closed injection or test valves by a low-flow bypass line to the suppression pool. A *water leg* pump keeps the piping between the pump and the injection valve full of water to ensure quick response and to eliminate potential hydraulic damage on system initiation. In the event of complete loss of normal electrical power, the spray system may be operated (automatically or manually) from the standby diesel generator.

The operation of the LPCS system pump is initiated from independent, redundant signals indicating low-reactor-water level and/or high pressure in the drywell, both using a one-out-of-two-twice logic. (The same signals initiate starting of the standby diesel generators.) The motor-operated valve in the discharge line opens automatically upon activation of the pump and a permissive pressure differential across the valve. As the reactor vessel pressure decreases, the flow rate of water to the reactor vessel increases. A testable check valve in the discharge line located inside the containment precludes backflow from the reactor vessel when the vessel pressure is greater than the pump discharge pressure. The operation of the system can be initiated from the main control room.

Water lost from the reactor vessel collects in the drywell to the level of the weir wall and then flows into the suppression chamber. This establishes a closed loop allowing the spray system to continue to operate until the operator manually stops it.

A bypass line to the suppression pool capable of rated core spray flow permits testing while the power plant is in service. A motor-operated valve controls bypass flow and is operated by a key locked switch in the control room. The position of the valve (as is true for all air- or motor-operated valves) is indicated in the control room. The valve receives a signal to close, which pre-empts all others, in the event that operation of the LPCS system is required.

To allow for system testing during plant shutdown, reactor water, via a temporary connection (removable spool piece) to the RHR system, is discharged into the reactor vessel through the core spray sparger. The spool piece is removed prior to plant startup and the open pipe capped.

### 3.4.8.3 Automatic Depressurization Function

Blowdown, through selected safety/relief valves, in conjunction with the operation of the LPCI function of the RHR system and/or the LPCS system functions as an alternate to the operation of the HPCS system for protection against fuel cladding damage (fragmentation) upon loss of coolant over a given range of steam or liquid line breaks. The blowdown depressurizes the reactor vessel, permitting the operation of the LPCI function and/or the LPCS system. Blowdown is activated automatically upon coincident signals of low water level in the reactor vessel and high drywell pressure. A time delay of approximately 2 min after receipt of the coincident signals allows the operator time to bypass the automatic blowdown if the signals are erroneous or the condition has corrected itself. The operator can initiate blowdown from the control room at any time.

### 3.4.9 RHR System

The RHR system removes residual heat generated by the core under normal (including hot standby) and abnormal shutdown conditions. The LPCI function of the RHR system is an integral part of the ECCS. The design objectives of the system follow:

- To restore and maintain, if necessary, the water level in the reactor vessel after a LOCA so that the core is sufficiently cooled to prevent fuel cladding damage (fragmentation)
- To limit suppression pool water temperature
- To remove decay heat and sensible heat from the nuclear boiler system while the reactor is shut down for refueling and servicing
- To condense reactor steam so that decay and residual heat may be removed if the main condenser is unavailable (hot standby)
- To supplement the fuel and containment pools cooling and cleanup system capacity when necessary to provide additional cooling capability

The RHR system is made up of various subsystems with the following operational functions to satisfy these objectives.

#### 3.4.9.1 LPCI

The LPCI function in conjunction with the LPCS system, the HPCS system, and/or automatic depressurization of the nuclear boiler system (depending upon operability of the HPCS system or level of depletion of reactor vessel water) will restore and maintain the desired water level in the reactor vessel required for cooling after a LOCA.

In conjunction with the LPCS system, redundancy of capability for core cooling is achieved by sizing the RHR pumps so that the required flow is maintained with one pump not operating. Using a split bus arrangement for pump power supply (essential power system), two RHR pumps are connected to one bus and the third RHR pump and an LPCS pump are connected to the second bus to obtain the desired cooling capability. The pumps deliver full flow inside the core shroud when the differential pressure between the reactor vessel and the containment approaches 20 psi (138 Pa). The availability of the LPCI function is not required during normal nuclear system startup or cool down when the reactor vessel gage pressure is <135 psi (931 kPa). The operability of the pumps can be tested at any time during normal plant operation by bypassing the reactor vessel and pumping the flow back to the pressure suppression pool.

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## 3.5 Instrumentation and Controls

### 3.5.1 Introduction

The instrumentation of the BWR is generally associated with the control of the reactor, the prevention of the operation of the plant under unsafe or potentially unsafe conditions, the monitoring of process fluids and gases, and for monitoring of the performance of the plant. The control of the plant is from the control room. Instrumentation for monitoring the performance of the plant is located in the control room and locally.

Power output from the BWR is controlled by changes in reactor water recirculation flow rate or by the moving of control rods. As the reactor power output changes, the turbine initial pressure regulator adjusts the turbine admission valve to maintain nearly constant reactor pressure, admitting the new steam flow to produce the desired change in the turbine generator power output. The BWR is operated at constant reactor pressure because pressure changes caused by turbine throttle operation in response to load changes tend to bring about reactor power changes opposite to the desired change. However, small controlled pressure changes are used to improve load response.

### 3.5.2 Plant Startup

Startup of the plant from a cold standby condition to a power producing condition requires the

- Startup of the reactor water recirculation pumps
- Pumps brought to rated speed
- Manipulation of the reactor water recirculation flow control valves to provide the required flow
- Movement of control rods to attain the desired power level
- Monitoring of the reactor to record and monitor reactor behavior

The operator manually controls the startup of the plant from a cold standby condition.

#### 3.5.2.1 Reactor Startup and Operation

The operational sequence for the startup of the plant from a cold standby condition is as follows:

- The flow control valves are set at the minimum position, which corresponds to approximately 25% of rated flow.
- The reactor water recirculation pumps are started. Because the low-frequency motor generator sets cannot start the recirculation pump motors, the pump motors are started from auxiliary power and transferred to the low frequency motor generator sets when the pump motor nears full speed and after the starting current has dropped.
- Control rods are manually withdrawn according to a predetermined schedule to achieve criticality of the reactor. They are further withdrawn to approximately 32% of rated power with the reactor water flow control valves fully open and the recirculation pumps operating at low speed (25%). The rate at which power level is raised is usually limited by conditions of thermal expansion of the reactor vessel.

- At approximately 32% of rated power, the reactor water flow control valves are closed and the recirculation pump transferred to auxiliary power and operated at rated speed.
- From approximately 30% to approximately 40% of rated power, the control of power level is by manual control of recirculation flow by changes in control valve position from minimum position.
- Above approximately 65% of rated core flow, the recirculation flow control is automatic.
- Between approximately 38% and approximately 75% of rated power, control rods are normally used to change power level.
- Above approximately 75% of rated power, change in reactor water recirculation flow is normally used to change power level.
- Neutron monitoring channels monitor the nuclear behavior of the reactor. Counting channels are used in the subcritical range up through criticality. The neutron counting channels and/or intermediate range monitoring channels monitor the intermediate range, from criticality to the power range. The power range neutron monitors are used throughout the power range usually above 21% of rated power.

During initial power operation, an operating curve is established relating reactor power to recirculation flow. The first point of the curve is full flow and rated power. When a rod pattern is established for this point, recirculation flow is reduced in steps at the same rod pattern, and the relationship of flow to power is plotted. Other curves are established at lower power ratings and other rod patterns as desired. During operation, flow control adjustment, rod positioning, or a combination of the two may change reactor power, while adhering to established operating curves. A rod withdrawal interlock is used to prevent unscheduled rod withdrawal, which would result in an excessive power-to-flow ratio. The operating curves are evaluated periodically, usually during startups, to compensate for changing reactivity coefficients. Although control rod movement is not required when the load is changed by recirculation flow adjustment, long-term transient reactivity effects are normally compensated for by control rod adjustment.

### **3.5.2.2 Turbine Startup**

While the reactor temperature is being increased, the turning gear rotates the turbine. When reactor steam is available, the shaft seal steam is applied and the mechanical vacuum pump is started. After a partial vacuum is established in the main condenser, heating of the turbine and steam flow from the reactor are accomplished by first establishing a flow of steam to the condenser through the bypass valves. This flow is gradually transferred to the turbine until rated speed is achieved after which the unit is synchronized with the system. The initial pressure regulator controls the bypass flow during this initial period and the governor controls the turbine. The initial pressure regulator assumes normal control of the turbine admission valve after the unit is synchronized and a small amount of load is applied.

### **3.5.3 Power Operation**

After the generator is synchronized to the electrical system and is producing a substantial output, the power output is adjusted to meet the system requirements by manual adjustment of control rods, manual or automatic adjustment of reactor recirculation flow, or a combination of these two methods.



### **3.5.3.1 Control Rod Adjustment**

Withdrawing a control rod reduces the neutron absorption and increases core reactivity. Reactor power then increases until the increased steam formation just balances the change in reactivity caused by the rod withdrawal. The increase in boiling rate ends to raise reactor pressure, causing the initial pressure regulator to open the turbine admission valves sufficiently to maintain a constant pressure. When a control rod is inserted, the converse effect occurs.

The rate of power increase is limited to the rate at which control rods can be withdrawn. Control rods can be operated one at a time, or in groups of four rods in a symmetrical pattern. Single rods or rod groups can be withdrawn continuously or in incremental steps (notch steps). Continuous movement is usually limited to subcritical and heatup conditions. Control rod movement is the normal method of making large changes in reactor power, such as daily or weekly load shifts requiring reduction and increases of more than 25% of rated power.

### **3.5.3.2 Recirculation Flow Control**

The BWR is unique in that reactor power output can be varied over a power range of approximately 25% of the operating power level by adjustment of the reactor recirculation flow without any movement of control rods. This is the normal method used for load following and maneuvering the reactor and allows for load following at rates of up to 1% of rated power per second.

Reactor power change is accomplished by using the negative power coefficient. An increase in recirculation flow temporarily reduces the volume of steam in the core by removing the steam voids at a faster rate. This increases the reactivity of the core, which causes the reactor power level to increase. The increased steam generation rate increases the steam volume in the core with a consequent negative reactivity effect, and a new constant power level is established. When recirculation flow is reduced, the power level is reduced in a similar manner.

The adjustment of the flow control valve changes the recirculation flow rate. To change reactor power, a demand signal from the operator or a load speed error signal from the speed governing mechanism is supplied to the master controller. A signal from the master controller adjusts the position setting of the controller for each valve. This signal is compared with the actual position of the valve associated with each controller. The resulting error signal causes adjustment of the valve position to reduce the error signal to zero, and the reactor power change resulting from the change in recirculation flow causes the initial pressure regulator to reposition the turbine control valves.

Automatic load control is accomplished by supplying a speed-load error signal from the turbine governor to the master controller. The energy storage capability of the water in the reactor system is used to increase the speed of response of the automatic load control system. An automatic, temporary change in the set point of the pressure regulator is produced when there is a demand for a change in turbine output. If an increase in load is demanded, the pressure set point is lowered and water in the reactor system flashes to produce extra steam flow to the turbine. If a decrease in load is demanded, the pressure set point is raised which causes the turbine control valve to move toward the closed position.

### **3.5.3.3 System Control**

Control signals between the reactor and the turbine provide two functions required for normal operation. A signal from the initial pressure regulator is provided to the turbine

admission valves to maintain a nearly constant reactor pressure. A signal from the speed-load governing mechanism to the master flow controller establishes the necessary reactor recirculation flow required to meet the system power requirements.

If, while under normal load, the turbine speed decreases or the speed-load changer setting is increased, a positive speed-load signal is transmitted to the initial pressure regulator and the master flow controller. The increase in signal causes a momentary decrease in the pressure setting of the initial pressure regulator and causes the master controller to increase the flow demand to the recirculation system flow valve controller.

Decreasing the pressure setting of the initial pressure regulator causes a signal to be sent to the turbine admission valves, instructing them to open rapidly by an amount and for a length of time, which is a function of the speed-load error. This gives a limited rapid initial response to a speed-load error by increasing the steam flow from the reactor vessel. The allowable duration of this transient increased steam flow is limited by the fact that increased steam flow tends to reduce the reactor pressure and power level.

The increased flow demand to the recirculation system flow valve controller causes the flow control valve to open wider, causing an increase in reactor recirculation flow. The increased flow increases the reactor power output by sweeping out steam bubbles from the core faster, thus raising the effective density of the moderator. The increased steaming rate causes a slight increase in reactor pressure. The increase of pressure is sensed by the initial pressure regulator, which sends a signal instructing the turbine admission valves to open sufficiently to increase the turbine output to a level that will cancel the speed-load error.

*Daily load following:* Essentially any practical daily load following profile can be followed. There are no restrictions due to spatial xenon oscillations. (Early ascent above 95% of rated power would be subject to xenon override considerations.) Power levels can be readily reduced to any level during daily load following, including the power level where the turbine generator is supplying only house loads. Automatic load following provides the capability to accept large changes in load demand at operating power levels. The change in load demand may be initiated at any power level or reactor water recirculation flow combination in the automatic flow control range. This region lies between 28% and 75% of rated power and core flow rate ranging from approximately 65% to approximately 68% (constant flow control valve setting) and between 40% and rated power at rated core flow. For load reduction demands that exceed the range of the automatic flow control system, the main steam bypass system provides additional capability up to the bypass system capacity. The reactor operator would then establish a new control rod configuration to match the new power demand. For load increase demands that exceed the range of the automatic flow control system (assuming reduced flow initially), the power level will rise to that level corresponding to rated core flow and remain there until the control rods can be adjusted to increase power up to the desired level. Step demands for up to 25% of the power at rated core flow is accommodated by automatic reactor water recirculation flow control.

*Automatic dispatch operation:* Automatic reactor water recirculation flow control in combination with ganged control rods allows full participation in an automatic dispatch system with the combined purpose of meeting tie line regulation, spinning reserve, grid load rejection, and daily load following requirements. During such operation, automatic reactor water recirculation flow control meets the rapid changes in load demand required by tie line regulation, while simultaneously providing margins for spinning reserve or grid load rejection. The unit operator would adjust control rods to preserve the desired automatic margins during the slower changes in base power level required by daily load following.

#### **3.5.3.4 Turbine Bypass Valve**

A fast response, modulating-type valve, controlled by the steam bypass pressure regulator system, is used to perform three basic functions. The primary function is to reduce the rate of rise of reactor pressure when the turbine admission valves are moved rapidly in the closing direction. To perform this function, the bypass valve needs about the same speed of response as the turbine admission valves to prevent a pressure-induced reactor scram from high neutron flux when the turbine load is suddenly reduced by partial or complete closure of the turbine admission valves.

The second function of the bypass valve is to control reactor pressure during startup of the turbine. This allows the reactor power level to be held constant while the turbine steam flow is varied as the turbine is brought up to speed under the control of its speed governor.

The third function of the bypass valve is to help control reactor pressure after the turbine has been tripped. It is used to discharge the decay heat to the condenser and to control the rate of cooling of the reactor system.

#### **3.5.3.5 Pressure Relief Function**

A pressure relief function is used to control large pressure transients. This system will operate safety/relief valves following closure of the main steam isolation valves or the sudden closure of the turbine admission or stop valves and failure of the turbine bypass system to relieve the excess pressure. For this function, the safety/relief valves discharge steam from the steam lines inside the drywell to the suppression chamber. Each safety/relief valve is operated from its own overpressure signal for the relief function, and by direct spring action for the safety function.

To limit the cycling of safety/relief valves to one valve subsequent to their initial actuation during a main steam line isolation event, two valves (one a backup to the other) have the feature of automatically changing normal set pressures (opening and closing) following their initial actuation at normal set pressures to a lower level, thereby limiting the pressure cycles to a level where the other relief valves will not reopen. In conjunction with these two valves, the set pressure for the closing of the other valves is changed automatically which allows for them to stay open longer before closing to accommodate pressure swings. Manual valve operation and resetting of valve set pressure to their normal levels following the transient is by the control room operator.

#### **3.5.3.6 Reactor Feedwater Control System**

The reactor feedwater control system automatically controls the flow of reactor feedwater into the reactor vessel to maintain the water in the vessel within predetermined levels during all modes of plant operation. The control system utilizes signals from reactor vessel water level, steam flow, and feedwater flow.

The reactor feedwater control system provides the signal for the reduction of reactor water recirculation flow to accommodate reduced feedwater flow caused by failure of a single feedwater pump.

#### **3.5.4 Plant Shutdown**

For normal plant shutdown, reactor power and plant output are reduced by manual insertion of control rods. After turbine load is reduced to a minimum value, steam flow is maintained through the bypass valve and the generator is disconnected from the system.

Reactor power is further reduced to a low level, and the decay heat is rejected to the condenser through the turbine bypass valve. If the reactor is to be kept in the hot standby or steam condensing condition, criticality is maintained but fission power is reduced to a low level (about 0.01% of rated power is sufficient to maintain operating temperature). If refueling or other functions requiring access to the vessel are planned, all control rods are inserted and the reactor is cooled down by release of steam to the main condenser. The rate of cool down is normally controlled by periodically lowering the setting of the initial pressure regulator. After vessel gage pressure has been reduced sufficiently 1135 psi (930 kPa), the heat sink can be switched from the main condenser to the RHR system heat exchangers to get the reactor to the cold shutdown condition.

Reactor power is monitored from the source range up through the power operating range by suitable neutron monitoring channels, with all detectors inside the reactor core. This location of detectors provides maximum sensitivity to control rod movement during the startup period and provides optimum monitoring in the intermediate and power ranges. Three types of neutron monitoring are used: source range counting, intermediate range; voltage variance method, local power range; and DC ion chambers. A traversing in-core probe system provides for periodic calibration of the neutron detectors.

#### **3.5.4.1 Source Range Monitor**

In the source range, the neutron flux is monitored by fission counters, which are inserted to about the mid-plane of the core by the drive mechanisms, which move each chamber into the core through inverted thimbles. A range from below the source level to  $10^9$  nv is covered.

As startup progresses and the count rate approach the top of the meter range (about  $10^6$  cps), the counters are withdrawn downward to give a drop in apparent count rate. Criticality normally occurs before movement of the counters is necessary. The counters can be motor driven to any position within their limits of travel; however, two or three selected positions will provide the necessary range to achieve criticality and provide overlap with the intermediate range monitors.

When the reactor reaches the power range, the counters are moved to a position approximately 2 ft (0.61 m) below the core. This places the counters in a low neutron flux so that burnup and activation of the counters are minimized.

#### **3.5.4.2 Intermediate Range Monitors**

The intermediate range is from about  $10^8$  to  $1.5 \times 10^{13}$  nv. In this range, the neutron flux is monitored by a system using a voltage variance method (also known as mean square value or Campbell method). This method makes use of the AC component of voltage, which is due to the random nature of neutron pulses generated in a detection chamber. With small chambers located in the high temperature ambient of the reactor core, the AC component is used to measure neutron flux at lower power levels because cable leakage and gamma radiation have relatively little effect on the signal.

These fission chambers are also withdrawn during full power operation to maintain their expected life and to reduce activation. They are positioned with drive mechanisms similar to those used for the source range fission counters.

### 3.5.4.3 Local Power Range Monitor

In the power range, neutron flux is monitored by fixed in-core ion chambers, which are arranged in a uniform pattern throughout the core. These chambers cover a range of about 1%–125% of rated power on a linear scale. When a control rod or group of control rods is selected for movement, the readings from the adjacent detectors are displayed on the operator's control bench board together with a display of the position of the rod or group of rods.

Detector assemblies each contain four fission chambers and a calibration guide tube for a traversing ion chamber. The chambers are uniformly spaced in an axial direction and lie in four horizontal planes. Each ion chamber is connected to a DC amplifier with a linear output. Internal controls permit adjustment of the amplifier gain to compensate for the reduction of chamber sensitivity caused by burnup of its fissionable material. These detectors are individually replaced through the bottom of the reactor vessel.

The calibration guide tube included in each fixed in-core assembly permits the insertion of a traversing ion chamber to obtain vertical flux profiles and to calibrate the chambers. Each calibration guide tube extends nearly to the top of the active portion of the core and is sealed at the upper end. The tubes pass through the nozzles and seals beneath the reactor vessel and connect to an indexing mechanism located inside the containment. The indexing mechanism permits the traversing ion chamber to be directed to many different detector assemblies.

Fully inserting the traversing ion chamber into one of the calibration guide tubes and then taking data as the chamber is withdrawn obtains flux readings along the axial length of the core. The data goes directly to the computer. One traversing chamber and its associated drive mechanism is used for each group of seven to nine fixed in-core assemblies (depending on reactor size).

### 3.5.4.4 Average Power Range Monitor

Four average power range monitors (APRMs) measure the average power level. Each monitor measures bulk power in the core by averaging signals from as many as 24 detectors distributed throughout the core. The output signals from these monitors are displayed and are also used to operate trips in the reactor protection system.

## 3.5.5 Nuclear System Protection System

The nuclear system protection system is a four-channel electrical alarm and actuating system that monitors the operation of the reactor, which, upon sensing an abnormal condition, initiates action to prevent an unsafe or potentially unsafe condition. The system integrates the following functions:

*Reactor trip:* Monitors reactor operation and shuts down the reactor when certain limits are exceeded

*Nuclear system isolation:* Isolates the reactor vessel and all connections of the primary pressure boundary that penetrate the containment barrier

*Engineered safety feature actuation:* Actuates engineered safety feature systems

The nuclear system protection system uses *solid-state* electronic technology from sensor output to actuation device inputs, which include sensors, signal conditioning, and combinational logic and actuator logic. The system provides for the analog indication of major variables, separation of divisions, and online testability.

Logic bases for the nuclear system protection system functions are as follows:

- Reactor trip initiation for automatic control and reactor shutdown is based on a two-out-of-four logic.
- Nuclear system isolation by isolation valve closure in process lines penetrating the containment barrier is based on two-out-of-four logic for main steam isolation valves and a one-out-of-two taken twice logic for remainder of nuclear system isolation function.
- Engineered safety feature systems initiation is based on a one-out-of-two taken twice logic.

Sensors can be analog (such as process control transmitters) or digital (such as pressure switches or limit switches). Analog inputs for important variables drive indicators, which allow the operator not only to see the absolute value but also to compare readings in different channels. Both analog and digital signals are modified, if necessary, in signal conditioners to signals that are compatible with the solid-state logic. After conditioning, the digital signals go directly to the decision logic. Each conditioned analog signal is compared with the output of a set point generator in a bistable trip unit. When the preset level is exceeded, the bistable puts out a signal to the decision logic.

The decision logic is made up of solid-state circuitry that compares with various inputs. When a combination of inputs requires action, the logic circuitry provides a signal that seals into turn on a solid-state power gate that operates activation devices, such as contactors, circuit breakers, and solenoid pilot valves. Actuation devices in turn control power to the motors that operate valves and drive pumps, or control the air supply to pneumatically operated valves.

Simultaneous open and close manual switch conflicts are prevented by exclusive *OR* logic. Manual inputs may be momentary or maintained. The identity of the most recent momentary input is retained. When a maintained manual input is removed, the input channel reverts to automatic status.

Upon loss of AC power functions, which are normally energized (such as the reactor trip function) will provide fail-safe trip action. For such functions, loss of power to a sensor, its channel, or associated logic automatically produces a trip output. For normally de-energized functions (such as emergency core cooling) loss of power to a sensor, its channel, or associated logic leaves the state of the actuated equipment unchanged. Subsequent restoration of power will not introduce transients that could cause a change of state in the actuated equipment.

#### **3.5.5.1 Reactor Trip Function**

The nuclear system protection system initiates the rapid insertion of the control rods to shut down the reactor. The system is of the fail-safe design where it will trip on loss of electrical power but will not trip and cause a scram on the loss of a single power source. The four trip channels are physically separated from each other and from other equipment precluding the possibility of interactions that could cause possible false scrams or failure to scram. The logic requires a manual reset by the operator, which is automatically inhibited for 10 s. One reset switch is used for each trip channel. Failure of a single trip channel, division logic, or a system component will not prevent the normal protective action of the nuclear system protection system.

### **3.5.5.2 Nuclear System Isolation Function**

The nuclear system protection system provides for the closing of valves to isolate the containment, thereby preventing the release of steam and process fluids. The logic and equipment required for those valves, which are required to be open during ECC are part of each of the separate ECCs.

The lines, which penetrate the containment and are required, to be isolated during emergency core cooling consist of three groups:

*Reactor coolant pressure boundary isolation:* These are lines that connect directly to the reactor vessel and penetrate the drywell and containment barrier.

*Containment isolation:* These are lines that do not connect to the reactor vessel but penetrate the drywell and containment atmosphere.

*Closed system isolation:* These are the lines that penetrate the containment. However, they are neither part of the reactor coolant pressure boundary nor are they connected directly to the containment atmosphere.

All isolation valves except nontestable check valves are capable of remote manual control from the control room. Automatic closure signals override manual control signals. Once isolation has been initiated, valves close fully and will not reopen automatically when the signal clears. Valve position (except nontestable check valves) is indicated in the control room.

Power and control systems associated with containment isolation are multichannel, fail-safety. Failure of a single sensor circuit or system component does not prevent normal protective action. Separate routes from different, reliable power sources feed two valves in the same line. Control power and motive power for an electrically operated valve are supplied from the same source.

### **3.5.5.3 Engineered Safety Features Actuation Function**

The engineered safety features include the ECCS (HPCS system, LPCS system, LPCI function of the RHR system, and the automatic depressurization function of the nuclear boiler system) and the RCIC systems.

### **3.5.5.4 Divisional Separation**

Four divisional separations are used for reactor trip, isolation, and ECC inputs and outputs, both physically and electrically. Physical separation divisions are established by their relationship to the reactor vessel, which is divided into four quadrants. The sensors, logic, and output of the various systems are allocated to divisions.

Connections between divisions are isolated optically at the output of the originating cabinet or panel and buffered electrically at the receiving cabinet or panel. Connections to external devices, such as annunciators, indicators, and the computer, are similarly isolated and buffered.

### **3.5.5.5 Power Distribution**

AC and DC power are required for the nuclear system protection system. Power distribution is divided into four divisions.

An inverter supplied from either the AC emergency bus or the DC battery bus provides AC power for the scram solenoid pilot valves and the main steam isolation valves.

#### **3.5.5.6 Reset and Annunciation**

A momentary trip of any channel is annunciated and causes that channel to lock out until manually reset. Sufficient annunciation trip signals are used so that the operator can determine the particular sensor or sensors, which caused the channel trip. The computer also prints out the identification of sensors, which have caused scram and, if several variables are involved, it prints out the sequence of events in which they occurred.

#### **3.5.5.7 Backup Protection**

Two three-way normally de-energized solenoid valves are used to remove the main instrument air supply from all scram valves. If any of the scram pilot valves failed to operate properly during scram, then the associated control rods would be scrambled by the loss of air supply due to operation of the back-up scram valve.

Conditions monitored and inputs that activate the nuclear system protection system are as follows:

*High pressure in the drywell:* Abnormal drywell pressure trips the reactor, initiates the automatic depressurization function, the HPCS system, the LPCS system, and the RHR system.

*Low water level in the reactor vessel:* A low water level in the reactor vessel trips the reactor, causes nuclear system isolation, activates the automatic depressurization function, initiates the HPCS and LPCS systems, and initiates the RCIC system.

*High pressure in the reactor vessel:* High pressure in the reactor vessel will trip the reactor and initiate automatic depressurization function.

High neutron flux will cause a reactor trip.

*High water level in the scram discharge volume:* High water level in the control rod drive scram discharge volume will cause a reactor trip.

*Turbine stop valve closure and turbine control valve fast closure:* Turbine stop valve closure and fast closure of the turbine control valve will cause a reactor trip.

*Main steam line isolation:* The closure of the main steam line isolation valves will cause a reactor trip.

*High radiation activity near main steam line:* High radiation levels near the main steam lines will cause a reactor trip and nuclear system isolation.

*Leak detection:* Excessive leakage will cause nuclear system isolation.

*Low pressure at the turbine inlet:* Low pressure at the turbine inlet will cause nuclear system isolation.

#### **3.5.5.8 Bypass and Interlocks**

An operation mode switch on the reactor control panel controls the interlocking and bypassing of the protection system for the various operational modes. Following are the modes and interlocks provided.



*Shutdown:* This mode is for use when the reactor is to be shut down and maintenance work performed. All rods must be fully inserted and none can be withdrawn.

*Refuel:* This mode is used during refueling operations. It allows a single control rod to be withdrawn for test purposes.

*Startup and standby:* This mode is for starting up of the reactor and bringing it to a maximum of about 5% rated power. It also permits keeping the reactor critical while the turbine and associated equipment are being serviced with the main steam line isolation valves closed.

*Run:* This mode is for normal operation. The intermediate range flux scram is bypassed and all other function bypasses are removed. However, bypassing of some individual instruments for maintenance may be accomplished where permitted by operating procedures.

Interlocks are used on the intermediate range neutron monitors to ensure that all units are operating properly and on the proper range. Control rod withdrawal is blocked if the ratio of reactor power to recirculation flow exceeds a predetermined value.

### 3.5.6 Rod Control and Information

The primary purpose of the rod control and information (RC&I) function is to effect control rod motion as requested by the operator. It displays all information, which is relevant to the movement of rods. In addition to enabling the operator to move rods, this function also enforces adherence to operating restrictions, which limit the consequences of a potential rod drop accident. At higher power levels, it limits rod movement so that rods cannot be withdrawn to the point of generating excessive heat flux in the fuel. Unit conditions are considered in determining which restrictions are applied to a given rod movement request.

Rod position is sensed by a series of sealed glass reed switches contained within a tube inside the drive piston. Two switches are spaced every 3 in. (76 mm) with each of the dual switches feeding a separate channel. These signals are multiplexed inside the containment and transmitted to the control room. The rod position information function decodes these data and makes them available to other parts of the RC&I function, to the process computer and to the operator. The detection of an invalid input caused by a failed reed switch is indicated. The status of the scram valves and accumulators on the hydraulic control unit is monitored, and these data are available to the operator and the computer.

The speed and capacity of the RC&I function permit the control of more than one rod at a time. Up to four rods can be operated simultaneously. The position of each rod in a gang is monitored.



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# 4

## *Heavy Water Reactors*

Alistair I. Miller, John Luxat, Edward G. Price, and Paul J. Fehrenbach

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## 4.1 Introduction

In the 1950s and 1960s, heavy water reactor (HWR) technology was explored in most of the countries investigating the application of nuclear fission to energy production. However, it was in Canada that this line of reactors was initially selected as the preferred type, which would become known as CANada Deuterium Uranium (CANDU). The choice was influenced by the early development work in Canada within the Manhattan Project, which took advantage of the superior characteristics of heavy water moderation for the production of plutonium. The attraction for ongoing development was mostly in the comparative simplicity of a system that did not depend on isotopic enrichment of uranium for the fuel. Further simplicity was introduced with the choice of pressure tubes (rather than a pressure vessel) to contain the operating pressure. The use of natural uranium and of pressure tubes make CANDU technology relatively easily accessible. Fuel manufacture has been successfully developed virtually everywhere where CANDUs have been built, and the dependence on very specialized large component fabricators to produce pressure vessels has been avoided.

This approach is possible because of heavy water's excellence as an economical moderator: the normal—mass 1 atom, often called *protium*—hydrogen in light water is more effective in reducing the energy of neutrons than the heavier deuterium atom—the stable, mass 2 form of hydrogen—but deuterium has a far lower propensity to absorb neutrons than protium. This low-absorption property—often called *good neutron economy*—permits a chain reaction with natural uranium. On account of needing more collisions to achieve moderation, HWR cores are bigger than those for light-water-moderated reactors. This has mixed benefits and disadvantages: it is a source of added costs but a larger source of natural cooling and the comparative difficulty of achieving criticality means that maintaining or achieving criticality in abnormal or unintended circumstances is less likely or impossible.

Light-water-moderated reactors must bear the cost of enriching all of their fuel in  $^{235}\text{U}$  throughout their lives. Heavy-water-moderated (HWM) reactors avoid this, but must bear the initial cost of producing heavy water. Once produced, however, only minor losses of heavy water occur (typically  $\leq 1\%/a$  as make-up). Uranium enrichment and heavy water production are isotope separations of comparable difficulty. The separation factors exploited in isotope separation are larger for deuterium and protium than for  $^{235}\text{U}$

and  $^{238}\text{U}$ , but this advantage is balanced by the relatively high natural abundance of  $^{235}\text{U}$  (0.71%) compared with deuterium ( $\sim 0.015\%$ ). Overall, the basic costs of light water reactors (LWRs) and HWRs appear to be very comparable.

HWRs are versatile. They can use natural uranium as a fuel, and their good neutron economy gives them superior capabilities to burn almost any fertile or fissile fuel. Some uranium enrichment—though less than for LWRs—may be advantageous. This possibility includes the ability to burn fuel discharged from LWRs, extending its energy output by around 30%. Thorium can be burned in various ways, at or close to sustained breeding. Plutonium and actinides can be consumed. India—which has abundant thorium resources—has pioneered in R&D to use thorium as the main fuel in CANDU-type reactors.

This chapter summarizes the more important aspects of nuclear reactors in which heavy water ( $\text{D}_2\text{O}$ ) is used as the moderator. For a more comprehensive description of HWRs and associated references, the reader is referred to IAEA (2002) Technical Report Series No. 407, on which this chapter draws extensively.

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## 4.2 Characteristics of HWRs

There are four types of HWRs that have been constructed and operated to produce electricity:

1. Pressure tube heavy water-cooled, HWM reactor
2. Pressure tube boiling light water, HWM reactor
3. Pressure vessel heavy water-cooled, HWM reactor
4. Gas-cooled HWM reactor

The dominant type of HWR is the pressure tube heavy water-cooled, HWM reactor as defined by the CANDU and Indian series of reactors. In the following section, the characteristics of the CANDU 6 reactor are used to describe the important features of these reactors.

Key temperature and pressure parameters of Atomic Energy of Canada Limited's (AECL) CANDU 6 and India's proposed advanced HWR are summarized in Table 4.1.

### 4.2.1 Pressure Tube-Type HWRs

#### 4.2.1.1 Introduction

The CANDU series of reactors is designed to use natural uranium, but it can also use SEU or a variety of fuels. Typically, the reactor core is contained in a cylindrical austenitic stainless steel tank (calandria) that holds the heavy water moderator at low temperatures ( $< 80^\circ\text{C}$ ) and low pressure ( $\sim 0.1$  MPa). The ends of the cylinder are closed with two parallel end shields that are perforated with holes for the fuel channels, the holes being arranged in a square lattice pattern. Thin-walled Zircaloy 2 tubes are fastened to each inner tube sheet and act as stays for the end shields to form a leak tight tank. The holes in each end shield are connected with stainless steel tubes (lattice tubes) (Figure 4.1). Each fuel channel consists of a Zr-2.5%Nb pressure tube joined to martensitic stainless steel end fittings

**TABLE 4.1**

Key Temperature and Pressure Parameters of AECL's CANDU-6 and ACR-1000 Reactors

	CANDU-6	ACR-1000
Reactor outlet header pressure (MPa(g))	9.9	11.1
Reactor outlet header temperature (°C)	310	319
Reactor inlet header pressure (MPa(g))	11.2	12.5
Reactor inlet header temperature (°C)	260	275
Single channel flow (kg/s)	28	28
Number of channels	380	520
Steam generator temperature (°C)	260	275.5
Steam generator pressure (MPa(g))	4.6	5.9
Turbine inlet pressure (MPa(g))	4.4	5.7
Turbine inlet temperature (°C)	258	273
Turbine outlet pressure (kPa(a))	4.9	4.9
Turbine outlet temperature (°C)	32.5	32.5
Condenser pressure (kPa(a))	4.9	4.9

and occupies the tubular holes or lattice sites formed by each combined lattice tube and calandria tube. The fuel channel end fittings are supported on a pair of sliding bearings at each end, and the pressure tube is supported and separated from the calandria tube by annular spacers. Figure 4.1 illustrates the CANDU design.

#### 4.2.1.2 Design and Operating Characteristics

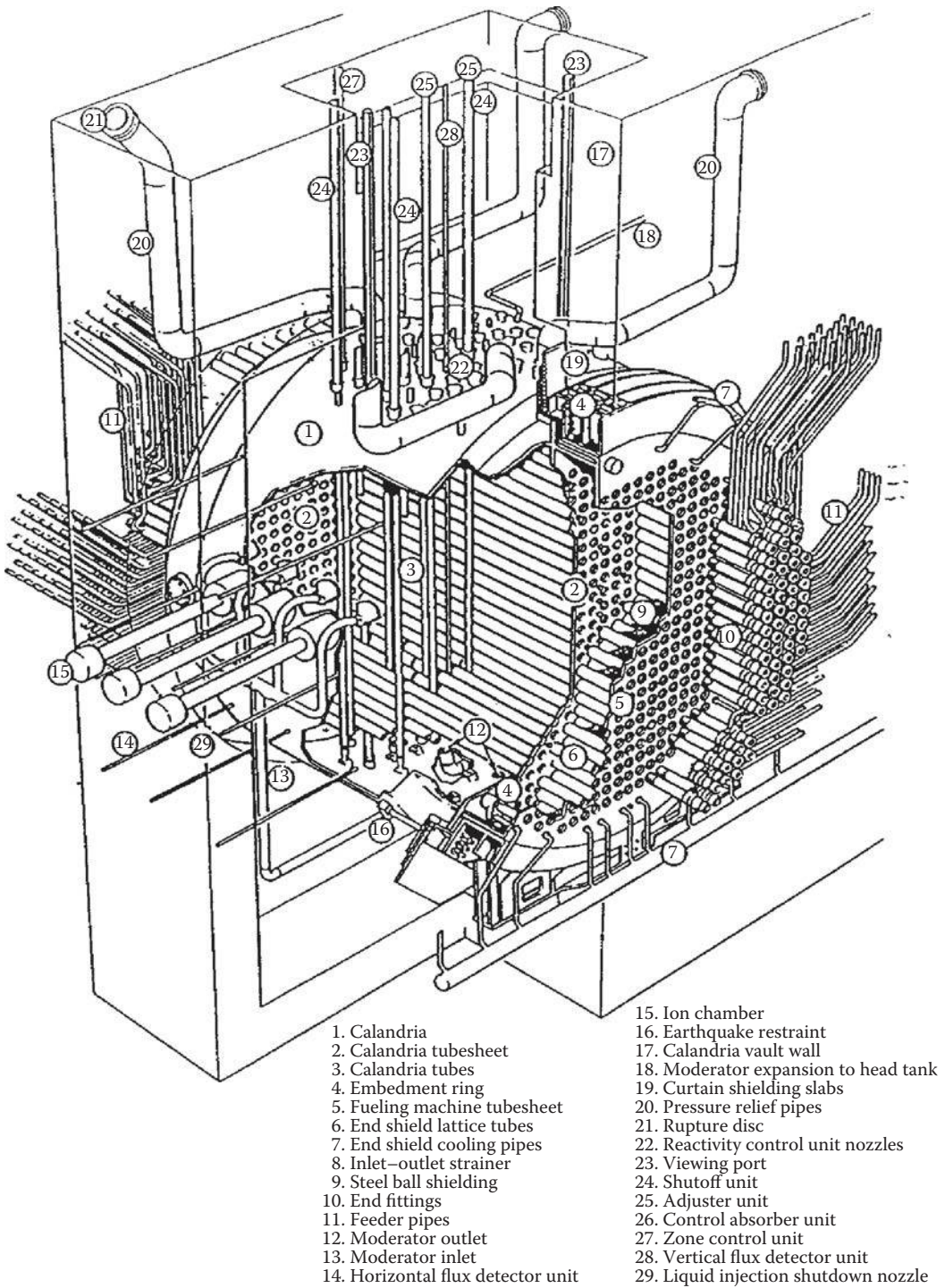
The pressure tube, heavy water-cooled, HWM reactor has certain characteristics which facilitate operation and safety analysis, and which provide fuel options. These are summarized in Sections 4.2.1.2.1 through 4.2.1.2.10.

##### 4.2.1.2.1 Pressure Tubes As the Reactor Pressure Boundary

Pressure tubes are thin-walled components with a simple geometry. This facilitates repetitive manufacture and inspection, both pre-service and in-service. Pressure tubes are replaceable and can be replaced at the end of their life to extend the reactor life, typically for 25–30 years.

As a result of the thin walls, there is no concern as regards overstressing the reactor pressure boundary under a fast cool down (e.g., steam main break). Zirconium alloy pressure tubes are sensitive to a hydride-related defect production and growth phenomena called delayed hydride cracking. A large analytical and experimental effort has been undertaken to establish the conditions under which a defect will penetrate the wall of the pressure tube and leak before it reaches an unstable length. A leak is detected through ingress of coolant to the annulus gas system, allowing time for a shutdown to replace the tube. Even if a pressure tube should fail, the damage is limited to the channel itself and some surrounding in-core components. Other channels will not fail.

The pressure tube geometry means that fuel elements are always within a few centimeters of the moderator, which can act as an emergency heat sink for postulated severe accidents such as a loss of coolant accident (LOCA) combined with loss of emergency core cooling (LOECC). This also provides an inherent limit to metal–water reactions in a severe accident because the fuel bundle is close to the emergency heat sink.



**FIGURE 4.1**  
 CANDU PHWR (schematic).

The horizontal channel orientation means that *graceful* sagging occurs in the event of a beyond-design-basis severe core damage accident, that is, assuming a LOCA with LOECC and loss of moderator cooling, the fuel channels would slump onto the bottom of the calandria, resulting in heat transfer to the water in the shield tank (at which point some melting may occur). Pressure tubes preclude the possibility of a sudden, large, high-pressure melt ejection occurring and eliminate one potential challenge to containment integrity.

There are no large high-pressure pipes directly connected to the reactor structure, so there are no overturning forces placed on the reactor from a large LOCA.

#### 4.2.1.2.2 Fuel

Fuel characteristics are as follows:

- Use of natural uranium fuel allows the storage and handling of new fuel with minimal criticality concerns because the fuel bundles require heavy water to become critical.
- On-power fueling means that there is very little reactivity hold-up needed in the reactor control system (and no need for boron in the coolant to hold down reactivity, resulting in a simpler design). The control rod reactivity worth can therefore be kept quite small (2 mk per rod or less).
- The high neutron economy, and hence low reactivity hold-up, of HWRs means that the reactor is very unlikely to become critical after any postulated beyond-design-basis severe core damage accident.
- Low remaining fissile content in spent fuel means that there are no criticality concerns in the spent fuel bay.

Fuel design is simple and performs well. Typically, the defect rate in operating CANDUs is <0.1% of all bundles (even smaller, of the order of 0.001%, in terms of fuel elements).

#### 4.2.1.2.3 Fueling Characteristics

On-power refueling and a failed fuel detection system allow fuel that becomes defective in operation to be located and removed without shutting down the reactor. This reduces the radiation fields from released fission products, allows access to most of the containment while the reactor is operating and reduces operator doses.

As a result of on-power fueling, the core state does not change after about the first year of operation. Thus, the reactivity characteristics remain constant throughout plant life, resulting in simpler operation and analysis. Ability to couple tools, similar in size to a fuel bundle, to the fueling machine allows it to be used for some inspections without necessitating removal of the pressure tube and in some instances without de-fueling the channels.

#### 4.2.1.2.4 Moderator Characteristics

The cool, low-pressure moderator removes 4.5% of the fuel heat during normal operation; about the same as the amount of decay heat removed shortly after shutdown. It can therefore act as a long-term emergency heat sink for a LOCA plus LOECC; the heat transfer is effective enough to prevent melting of the  $\text{UO}_2$  fuel and preserve channel integrity.

The HWR has an inherent prompt shutdown mechanism (besides the engineered shutdown systems [SDSs] and the control system) for beyond-design-basis severe core damage accidents. If steam is introduced into the moderator as a result of, for example, multiple



channel failures, then the immediate effect of loss of moderation would cause the reactor to shut down.

In the case of a channel failure, the moderator acts as an energy-absorbing *cushion*, preventing failure of the calandria vessel. Even for beyond-design-basis severe core damage accidents, where a number of channels are postulated to fail, the calandria may leak (pressure relief rupture discs) but would retain its gross structural integrity.

The low-pressure, low-temperature moderator contains the reactivity mechanisms and distributes the chemical trim, boron, for reactivity purposes and gadolinium nitrate for shutdown purposes.

#### 4.2.1.2.5 Heat Transport System Characteristics

Given the economic value of heavy water, the designers of HWRs pay great attention to preventing coolant leaks. Leak detection equipment is highly sensitive and leaks from any source can be detected very early. The heat transport system (HTS) contains minimal chemical additives (only LiOH for pH control and H<sub>2</sub> to produce a reducing chemistry).

#### 4.2.1.2.6 Tank Characteristics

The shield tank contains a large volume of water surrounding the calandria. In the case of beyond-design-basis accidents (BDBAs), for example, severe core damage accidents such as a LOCA plus LOECC plus loss of moderator heat removal plus failure of make-up to the moderator, the shield tank can provide water to the outside of the calandria shell, ensuring that it remains cool and therefore intact, thereby confining the damaged core material within the calandria. Recent HWR designs have added make-up to the shield tank and steam relief to ensure that this remains effective. Heat can be transferred from the debris through the thin-walled calandria shell to the shield tank without the debris melting through. This inherent *core catcher* provides debris retention and cooling functions. Because a severe core damage sequence can be stopped in the calandria, the challenge to containment is much reduced.

#### 4.2.1.2.7 Reactivity Control Characteristics

HWRs using natural uranium have a positive void coefficient, which leads to positive power coefficients. This is accommodated in the design by employing independent fast-acting SDSs based on poison injection into the moderator and spring assisted shut-off rods.

The long prompt neutron lifetime (about 1 ms) means that for reactivity transients even above prompt critical, the rate of rise in power is relatively slow. For example, the reactor period for an insertion of 5 mk is about 0.85 s<sup>-1</sup>, whereas for 7 mk it is about 2.4 s<sup>-1</sup>. The SDSs are, of course, designed to preclude prompt criticality.

Separation of coolant and moderator and the slow time response of moderator temperature eliminates moderator temperature feedback effects on power transients. The only way of diluting moderator poison (if present) is through an in-core break, which is small and hence would have an effect that is slow relative to SDS capability.

Reactivity control mechanisms penetrate the low-pressure moderator, not the coolant pressure boundary. They are therefore not subject to pressure-assisted ejection in the event of an accident and can be relied upon to perform their function.

Bulk power and spatial control are fully automated with digital control and computerized monitoring of the plant state, which simplifies the job of the operator and reduces the chances of operator error.

Control is through adjusters and the shut-off rods. These are of simple design with relatively large tolerances (e.g., loose fit in guide tubes). They do not interact with the fuel

bundles at all and are not, therefore, subject to jamming in the event of an accident damaging the fuel.

In the case of a severe accident (LOCA plus LOECC), the damaged fuel is confined to the fuel channels, and therefore there is no risk of melting the control rods.

#### 4.2.1.2.8 Shutdown Cooling

HWRs have a shutdown cooling system that can remove decay heat after shutdown from full pressure and temperature conditions. It is not necessary to depressurize the HTS.

#### 4.2.1.2.9 Safety Systems

The safe operation of a reactor necessitates that the fuel be kept adequately cool at all times to prevent loss of fuel cladding integrity and the consequent dispersion of radioactive species into the coolant. The safety systems that prevent or mitigate fuel damage are described below.

*4.2.1.2.9.1 Systems That Shut Down the Reactor in the Case of Accidents* The emergency core cooling system (ECCS) fulfils this purpose. It is a system that refills the reactor fuel channels with light water to remove residual or decay heat from the fuel. The fuel requires heavy water for the reactor to go critical and the light water of the ECCS suppresses criticality. There is no need to add boron to the ECCS water.

*4.2.1.2.9.2 Systems That Prevent Release of Radioactivity into the Environment* The major system fulfilling this function is the containment building. Current HWRs have a containment isolation system that has been demonstrated by on-power testing to have a probability of unavailability of  $<10^{-3}$  years/year. The building volumes are relatively large, resulting in low design pressures. Details of the operation of the safety systems are given in Section 4.4.

Most HWRs have two, independent, diverse, reliable, testable, redundant, fail-safe SDSs (as well as the control system). The two systems do not share instrumentation, logic actuation devices, or in-core components. One system uses rods, the other liquid poison injection. Each of the SDSs is effective, by itself, for all design basis accidents. With each one demonstrated by on-power testing to a reliability of 999 times out of 1000 attempts, the risk of a transient or accident occurring without shutdown is negligible.

Each safety SDS has the ability during an accident to shut down the reactor from the most reactive state to zero power cold conditions. Moderator poison is only needed in the long term (hours) to compensate for xenon decay.

The positive void coefficient, while it must be compensated for in an accident by the SDSs, has the advantage of resulting in fast and responsive neutronic trips for a number of accidents. It also ensures an inherent power reduction for rapid cool-down accidents such as steam main failure.

Most HWRs have two sources of emergency electrical power: Group 1 Class III diesels and separate, independent, seismically qualified Group 2 Class III diesels. This greatly reduces the risk of station blackout.

#### 4.2.1.2.10 Licensing

The HWR regulators' licensing philosophy usually places the onus on the proponent to demonstrate that the plant is safe while the regulator audits the result. The regulator does not prescribe the design in detail, thereby avoiding the conflict of interest inherent in reviewing its own design. Besides encouraging innovation, this process places full

responsibility for safety on the organization that owns and operates the plant, consistent with IAEA recommendations.

HWR regulations typically specify the classes of accident to be considered in the design. These include not only failures of an operating system (e.g., LOCA) but also such failures combined with a failure of the mitigating system (e.g., LOCA plus LOECC, with credit for only one SDS in any accident). The latter are design basis accidents in HWRs and must meet dose limits using deterministic analysis. The requirement to include these *dual* failures means that the least unlikely severe accidents are within the design basis and must not cause severe core damage. This results in a robust design.

Although the list of *design basis* accidents is specified in part in regulations, the proponent is required to demonstrate that the analysis has covered a complete set. This ensures that the scope of analysis is comprehensive.

Regulatory requirements in most HWR jurisdictions imply the use of probabilistic safety assessment (PSA), not just after the design is complete, but very early on in the design phase, when any identified weaknesses can still be rectified relatively inexpensively.

### 4.2.1.3 Nuclear Steam Supply System

#### 4.2.1.3.1 Introduction

The CANDU 6 is used as the basis for describing the features of the CANDU HWR. All CANDU 6 power plants are fundamentally identical, although there are differences in detail that largely result from different site conditions and from improvements made in the newer designs. The basic design features of the current generation of Indian 220-MWe HWRs and the 500-MWe versions are also generally similar except in some quantitative details.

The heat produced by controlled fission in the fuel is transferred to the pressurized heavy water coolant and circulated through the fuel channels and steam generators in a closed circuit. In the steam generators, the heat is used to produce light water steam. This steam is used to drive the turbine generator to produce electricity. The nuclear steam supply system (NSSS) is illustrated in Figure 4.2.

#### 4.2.1.3.2 Fuel and Fuel Handling System

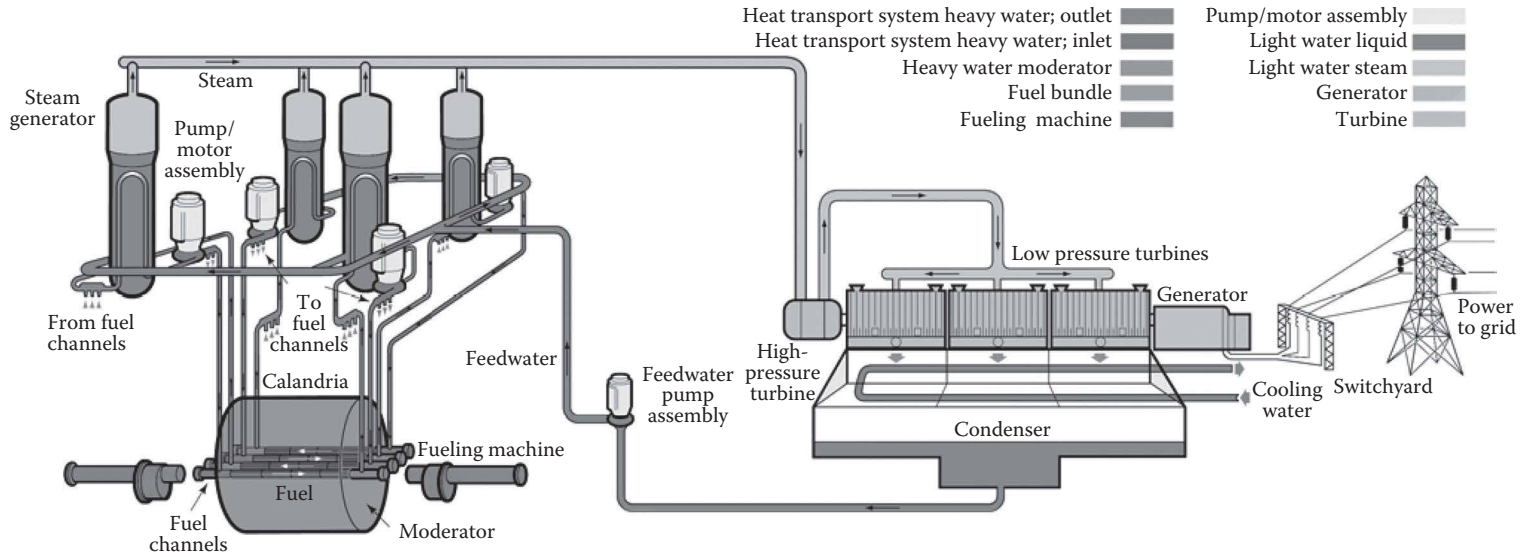
The fuel handling system

- Provides facilities for the storage and handling of new fuel
- Refuels the reactor remotely while it is operating at any level of power
- Transfers the irradiated fuel remotely from the reactor to the storage bay

The fuel-changing operation is based on the combined use of two remotely controlled fueling machines, one operating at each end of a fuel channel. Either machine can load or receive fuel. New fuel bundles, from one fueling machine, are inserted into a fuel channel in the same direction as the coolant flow—flow direction alternates between adjacent channels—and the displaced irradiated fuel bundles are received into the second fueling machine at the other end of the fuel channel.

Typically, either 4 or 8 of the 12 fuel bundles in a fuel channel are exchanged during a refueling operation. In the case of a CANDU 6 (700 MWe) reactor, a mean of 10 natural uranium fuel channels are refueled each week.

The fueling machines receive new fuel while connected to the new fuel port and discharge irradiated fuel while connected to the discharge port. The entire operation is



**FIGURE 4.2**  
Nuclear steam supply system.

directed from the control room through a pre-programmed computerized system. The control system provides a printed log of all operations and permits manual intervention by the operator. New fuel is received in the new fuel storage room located in the service building. This room accommodates 6 months' fuel inventory and can store, temporarily, all the fuel required for the initial fuel loading.

When required, fuel bundles are transferred to the new fuel transfer room located in the reactor building. Fuel bundles are identified and loaded manually into the magazines of the two new fuel ports. Transfer of new fuel bundles into the fueling machines is remotely controlled.

Irradiated fuel received in the discharge port from the fueling machine is transferred remotely onto an elevator that lowers it into a discharge bay filled with light water. The irradiated fuel is then conveyed, under water, through a transfer canal into a reception bay, where it is loaded onto storage trays or baskets and passed into the storage bay.

The discharge and transfer operations are remotely controlled by station staff. Operations in the storage bays are carried out under water, using special tools aided by cranes and hoists. Defective fuel is inserted into cans under water to limit the spread of contamination before transfer to the fuel bay. The storage capacity of the bays is sufficient to accommodate a minimum of 10 years' accumulation of irradiated fuel. Neither new nor irradiated CANDU fuel can achieve criticality in air or light water, regardless of the storage configuration. Thus, dry storage of fuel is possible after interim storage in the spent fuel bay. Safeguarding of fuel is facilitated by putting an identification number on each bundle. Numbers are recorded at various stages during fuel usage.

#### 4.2.1.3.3 HTS

The primary HTS (PHTS) in a CANDU 6 unit consists of two loops arranged in a figure-of-eight configuration with the coolant making two passes in opposite directions through the core during each complete circuit. The two PHTS pumps in each loop operate in series, causing the coolant to transport the fission heat generated in the fuel to the steam generators where it is transferred to light water, producing steam to drive the turbine. Each loop has one inlet and one outlet header at each end of the reactor core. The coolant is fed to each of the fuel channels through individual feeder pipes and returned from each channel through individual feeder pipes to the outlet headers.

Other key features of the circuit are listed below.

- Steam generators consist of an inverted U-tube bundle housed within a cylindrical shell, usually of a lightbulb shape. The steam generators include an integral preheater on the secondary side of the U-tube outlet section, and integral steam separating equipment in the steam drum above the U-tube bundle.
- Heat transport pumps are centrifugal motor-driven pumps, mounted with the shaft vertical and with a single suction and double discharge.
- In the event of electrical power supply interruption, cooling of the reactor fuel is maintained for a short period of time by the rotational momentum of the heat transport pumps during reactor power rundown, and by natural convection after the pumps have stopped.
- Chemistry control is relatively simple because chemicals do not have to be added to the PHTS for reactivity control.
- Carbon steel piping, which is ductile and relatively easy to fabricate and to inspect, is used in the HTS. Low concentrations of chromium are nowadays added to the steel to prevent flow-assisted corrosion from outlet water undersaturated in iron.

#### 4.2.1.3.4 Heat Transport Auxiliary Systems

There are four auxiliary systems attached to the HTS, which are required to perform specific functions, as detailed below.

1. Heat transport pressure and inventory control system provides
  - a. Pressure and inventory control for each HTS loop
  - b. Overpressure protection
  - c. Controlled degassing flow

The system consists of a pressurizer, D<sub>2</sub>O feed pumps, feed-and-bleed valves, D<sub>2</sub>O storage tank, degasser condenser, liquid relief valves, and safety valves.

The pressure in the PHTS of a CANDU 6 reactor is controlled by a pressurizer connected to the outlet headers at one end of the reactor. Pressure in the pressurizer is controlled by heaters in the pressurizer and by steam bleed. Heavy water in the pressurizer is heated electrically to pressurize the vapor space above the liquid. The volume of the vapor space is designed to cushion pressure transients, without allowing excessively high or low pressures to be generated in the HTS. (Nuclear power plants that do not allow the coolant to boil in the channels, do not use a pressurizer, and rely on the feed-and-bleed system for control.)

The pressurizer also accommodates the change in volume of the reactor coolant occurring in the HTS when the reactor moves from zero power to full power. This permits the reactor power to be increased or decreased rapidly, without imposing a severe demand on the D<sub>2</sub>O feed-and-bleed components of the system. The coolant inventory is adjusted by the feed-and-bleed circuit, and, with the pressurizer isolated, pressure can also be controlled by this system when the reactor is at low power or when the reactor is shut down. This feed-and-bleed circuit is designed to accommodate the changes in coolant volume that take place during heat-up and cool down.

2. D<sub>2</sub>O collection system's main purpose is to
  - a. Collect leakage from mechanical components
  - b. Receive D<sub>2</sub>O sampling flow
  - c. Receive D<sub>2</sub>O drained from equipment prior to maintenance

The collected D<sub>2</sub>O is pumped from the collection tank to the storage tanks of the pressure and inventory control system for reuse in the HTS. However, if the isotopic purity of the collection tank contents is low, the D<sub>2</sub>O can be pumped into drums for upgrading.

3. The shutdown cooling system, that is capable of
  - a. Cooling the HTS from 177°C down to 54°C, and holding the system at that temperature indefinitely
  - b. Providing core cooling during maintenance work on the steam generators and heat transport pumps when the HTS is drained down to the level of the headers
  - c. Being put into operation with the HTS at full temperature and pressure

The shutdown cooling system consists of two independent circuits, one located at each end of the reactor. Each circuit consists of a pump and a heat exchanger, connected between the inlet and outlet headers of both HTS loops. The system is normally full of D<sub>2</sub>O and isolated from the HTS by power-operated valves.

The shutdown cooling pumps are sized to ensure that boiling does not occur in any of the fuel channels at initial startup. During normal cool down, steam from the steam generators bypasses the turbine and flows into the turbine condenser, thereby reducing the HTS temperature to 177°C in approximately 30 min.

In order to achieve cool down from 177°C to 77°C, the isolating valves at the reactor headers are opened and all heat transport pumps are kept running. The heat transport pumps force a portion of the total core flow through the shutdown cooling heat exchangers where it is cooled by recirculated cooling water flowing around the heat exchanger coils.

After cooling to below 100°C, the heat transport pumps are shut down and the shut down cooling system pumps started. The system is then cooled to 54°C in this mode, enabling D<sub>2</sub>O to be drained down to the level just above the reactor headers, if required for maintenance of the steam generators or pumps.

4. The heat transport purification system
  - a. Limits the accumulation of corrosion products in the coolant by removing soluble and insoluble impurities
  - b. Removes accumulations of fine solids following their sudden release due to chemical, hydraulic, or temperature transients
  - c. Maintains the pD (pH of D<sub>2</sub>O) within the required range

Flow is taken from one reactor inlet header of each heat transport loop, passed through an interchanger, cooler, filter, and ion-exchange column before being returned through the interchanger to a pump inlet in each circuit. The pressure generated by the heat transport pump produces the flow through the purification system. The interchanger-cooler combination minimizes the heat loss in the D<sub>2</sub>O purification cycle.

#### 4.2.1.3.5 Moderator and Auxiliary Systems

The moderator absorbs 4.5% of reactor thermal power. The largest portion of this heat is from gamma radiation. Additional heat is generated by moderation (slowing down) of the fast neutrons produced by fission in the fuel and a small amount of heat is transferred to the moderator from the hot pressure tubes. For reactivity control, gadolinium, and occasionally boron, can be added or removed from the moderator fluid.

The moderator system includes two 100% capacity pumps, two 50% flow capacity heat exchangers cooled by recirculated light water, and a number of control and check valves. Connections are provided for the purification, liquid poison addition, D<sub>2</sub>O collection, supply, and sampling systems. The series/parallel arrangement of the moderator system lines and valves permits the output from either pump to be cooled by both of the heat exchangers and ensures an acceptable level of moderator cooling when either of the two pumps is isolated for maintenance. Reactor power must be reduced to about 60% if one moderator heat exchanger is isolated. The primary functions of the system are to

- Provide moderator cooling
- Control the level of heavy water in the calandria
- Maintain the calandria outlet temperature at approximately 70°C

The normal electric power supplied to the moderator system is backed up with an emergency power supply.

The heavy water in the calandria functions as a heat sink in the unlikely event of a LOCA in the HTS coinciding with a failure of emergency core cooling.

Helium is used as a cover gas for the moderator system because it is chemically inert and is not activated by neutron irradiation. Radiolysis of the heavy water moderator in the calandria results in the production of deuterium and oxygen gases. Circulation of the cover gas to catalytic recombiners reforms heavy water and prevents accumulation of these gases. The deuterium and oxygen concentrations are maintained well below levels at which an explosion hazard would exist.

The cover gas system includes two compressors and two recombination units that form a circuit for the circulation of cover gas through the calandria relief ducts. Normally, one compressor and both recombination units are operated, with the other compressor held on stand-by.

The moderator purification system

- Maintains the purity of  $D_2O$ , thereby minimizing radiolysis which can cause excessive production of deuterium in the cover gas
- Minimizes corrosion of components by removing impurities present in the  $D_2O$  and by controlling the pD
- Reduces, under operator command, the concentration of the soluble poisons, boron, and gadolinium, in response to reactivity demands
- Removes the soluble poison gadolinium after SDS2 has operated

Isolation valves in the purification system inlet and outlet lines are provided for maintenance purposes. The valves also allow drainage of the HTS coolant to just above the elevation of the headers without the need to drain the purification system. These valves close automatically in the event of LOCA.

The  $D_2O$  sampling system allows samples to be taken from the

- Main moderator system
- Moderator  $D_2O$  collection system
- Moderator purification system
- $D_2O$  cleanup system

Analyses may be performed on the samples to establish whether the chemistry of the heavy water falls within the specified range of chemistry parameters. These parameters include pD, conductivity, chloride concentration, isotopic purity, boron and gadolinium concentrations, tritium concentration, fluoride concentration, and organic content.

#### 4.2.1.3.6 Reactor Regulating System

The fundamental design requirement of the reactor regulating system is to control the reactor power at a specified level and, when required, to maneuver the reactor power level between set limits at specified rates. The reactor regulating system combines the reactor's neutron flux and thermal power measurements by means of reactivity control devices and a set of computer programs to perform three main functions:

1. Monitor and control total reactor power in order to satisfy station load demands
2. Monitor and control reactor flux shape



3. Monitor important plant parameters and reduce reactor power at an appropriate rate if any parameter is outside specified limits

Reactor regulating system action is controlled by digital computer programs that process the inputs from various sensing devices and activate the appropriate reactivity control devices.

All neutron flux measurement and control devices, both vertical and horizontal, are located in the low-pressure calandria, perpendicular to and between the horizontal fuel channels.

Computer programs provide

- Reactor power measurement and calibration
- Demand power routine
- Reactivity control and flux shaping
- Set-back routine
- Step-back routine
- Flux mapping routine

The principal instrumentation utilized for reactor regulation includes

- Ion chamber system
- Self-powered, in-core flux detectors
- Thermal power instrumentation

The nuclear instrumentation systems are designed to measure reactor neutron flux over the full operating range of the reactor. These measurements are required as inputs to the reactor regulating system and safety systems. The instrumentation for the safety systems is independent of that used by the reactor regulating system.

The reactivity control devices provide short-term global and spatial reactivity control. The devices are of two major types: mechanical and liquid.

The mechanical devices are the mechanical control absorbers and adjuster assemblies. The mechanical control absorbers comprise tubes containing cadmium (neutron absorber) that can be inserted to reduce power quickly. The adjuster assemblies comprise stainless steel tubes that are used to produce axial flattening of the fuel bundle powers as necessary. They can be removed from the core to add reactivity.

The liquid reactivity devices consist of the light water zone control units and the liquid poison addition system.

The function of the zone control system is to maintain a specified amount of reactivity in the reactor, this amount being determined by the deviation from the specified reactor power set point. If the zone control system is unable to provide the necessary correction, the program in the reactor regulating system draws on other reactivity control devices. Positive reactivity can be added by withdrawal of absorbers. Negative reactivity can be induced by insertion of mechanical control absorbers or by automatic addition of poison to the moderator.

The reliability of the reactor regulating system is of paramount importance and is achieved through having

- Direct digital control from dual redundant control computers
- Self-checking and automatic transfer to the stand-by computer on fault detection

- Control programs that are independent of each other
- Duplicated control programs
- Duplicated and triplicated inputs
- Hardware interlocks that limit the amount and rate of change of positive reactivity devices

#### 4.2.1.3.7 *Balance-of-Plant*

The balance-of-plant comprises the steam lines from the steam generators, the steam turbines and the alternating electrical generator, the condenser, various moisture separators and equipment to achieve de-aeration, demineralization, oxygen scavenging, reheating, and pH control of the feedwater returned to the steam generator.

The turbine generator system comprises steam turbines directly coupled to an alternating current (AC) electrical generator operating at synchronous speed.

The steam turbine is a tandem compound unit, generally consisting of a double-flow, high-pressure turbine and three double-flow, low-pressure turbines, which exhaust to a high vacuum condenser for maximum thermal efficiency. The condenser may be cooled by sea, lake, or river water, or by use of atmospheric cooling towers.

The generator is a high-efficiency, hydrogen-cooled machine arranged to supply AC at medium voltage to the electric power system.

**4.2.1.3.7.1 *Feedwater and Main Steam System*** Feedwater flows from the condenser via the regenerative feedwater heating system and is supplied separately to each steam generator. The feedwater is pumped into the steam generators by feedwater pumps with the flow rate to each steam generator regulated by feedwater control valves. A check valve in the feedwater line of each steam generator is provided to prevent backflow in the unlikely event of feedwater pipe failure. An auxiliary feedwater pump is provided to satisfy low-power feedwater requirements during shutdown conditions, or in the event that the main feedwater pumps become unavailable.

The chemistry of the feedwater to the steam generators is precisely controlled by demineralization, de-aeration, oxygen scavenging, and pH control. A blowdown system is provided for each steam generator that allows impurities collected in the steam generators to be removed to prevent their accumulation and possible long-term corrosive effects. In some reactors, the blowdown is collected and recirculated.

The heat supplied to the steam generators produces steam from the water that flows over the outside of the tubes. Moisture is removed from the steam by the steam separating equipment located in the drum (upper section) of the steam generator. The steam then flows via four separate steam mains, through the wall of the reactor building, to the turbine where they connect to the turbine steam chest via a main steam line isolation valve.

Steam pressure is normally controlled by the turbine governor valves, which admit steam to the high-pressure stage of the turbine. If the turbine is unavailable, up to 70% of full power steam flow can bypass the turbine and go directly to the condenser. Turbine bypass valves control pressure during this operation. Auxiliary bypass valves are also provided to permit up to 10% of full-power steam flow to discharge to the condenser during low-power operation. Steam pressure can be controlled by discharging steam directly into the atmosphere via four atmospheric steam discharge valves that have a combined capacity of 10% of full power steam flow. These valves are used primarily for control during warm-up or cool-down of the HTS.

Four safety relief valves connected to each steam main provide overpressure protection of the steam system.

**4.2.1.3.7.2 Turbine Generator System** The steam produced in the steam generators enters a single high-pressure turbine, and its water content increases as it expands through this high-pressure stage. On leaving this stage, the steam passes through separators where the moisture is removed. It then passes through reheaters where it is heated by live steam taken directly from the main steam lines. The reheated steam then passes through the low-pressure turbines and into the condenser where it condenses to water that is then returned to the steam generators via the feedwater heating system.

The steam turbine is a tandem compound unit, directly coupled to an electrical generator by a single shaft. It comprises one double-flow, high-pressure cylinder followed by external moisture separators, live steam reheaters, and three double-flow, low-pressure cylinders (recent and future plants have two low-pressure cylinders). The turbine is designed to operate with saturated inlet steam. The turbine system has main steam stop valves, governor valves, and reheat intercept and emergency stop valves, depending on the arrangement preferred by the architect/engineer. All of these valves close automatically in the event of a turbine protection system trip.

The generator is a three-phase, four-pole machine that typically operates at 1800 rpm to serve 60-Hz electrical systems, and at 1500 rpm to serve 50-Hz systems.

The associated equipment consists of a solid-state automatic voltage regulator that controls a thyristor converter which in turn supplies the generator field via a field circuit breaker, generator slip rings, and brush gear. The main power output from the generator to the step-up transformer is by means of a forced air-cooled, isolated phase bus duct, with tap offs to the unit service transformer, excitation transformer, and potential transformer cubicle.

The turbine condenser consists of three separate shells, each shell being connected to one of the three low-pressure turbine exhausts. Steam from the turbine flows into the shell where it flows over a tube bundle assembly through which cooling water is pumped and is condensed. The condenser cooling water system typically consists of a once-through circuit, using sea, lake, or river water. The condensed steam collects in a tank at the bottom of the condenser (termed the *hot well*). A vacuum system is provided to remove air and other noncondensable gases from the condenser shells. The condenser is designed to accept turbine bypass steam, thereby permitting the reactor power to be reduced from 100% to 70% if the turbine is unavailable. The bypass can accept 100% steam flow for a few minutes, and 70% of full-power steam flow continuously. On its return to the steam generators, condensate from the turbine condenser is pumped through the feedwater heating system. First, it passes through three low-pressure feedwater heater units, each of which contains two heaters fed by independent regenerative lines. This permits maintenance work to be carried out on the heaters with only a small effect on the turbine generator output. Two of the heater units incorporate drain cooling sections and the third a separate drain cooling stage.

Next, the feedwater enters a de-aerator where dissolved oxygen is removed. From the de-aerator, the feedwater is pumped to the steam generators through two high-pressure feedwater heaters, each incorporating drain cooling sections.

**4.2.1.3.7.3 Power System Station Services** The other major system of a nuclear plant is the electric power system. The normal electric power system comprises a main power output transformer, unit and service transformers, and a switchyard. This system steps up (increases) the generator output voltage to match the electric utility's grid requirements for

transmission to the load centers and also supplies the power needed to operate all of the station services. The main switchyard portion of the electric power system permits switching outputs between transmission lines and comprises automatic switching mechanisms and lightning and earthing protection to shield the equipment against electrical surges and faults.

The station services power supplies are classified according to their required levels of reliability. The reliability requirement of these power supplies is divided into four classes that range from uninterruptible power to power that can be interrupted with limited and acceptable consequences. The electric power system station services comprise the supply systems described below.

1. *Class IV power supply:* Power to auxiliaries and equipment that can tolerate long duration interruptions without endangering personnel or station equipment is obtained from a Class IV power supply. This class of power supply comprises

Two primary medium-voltage buses, each connected to the secondary windings of the system service and unit service transformers in such a way that only one bus is supplied from each transformer.

Two medium-voltage buses supplied from the secondary windings of two transformers on the primary medium voltage buses. These buses supply the main heat transport pumps, feed pumps, water circulation pumps, extractor pumps, and chillers.

A complete loss of Class IV power will initiate a reactor shutdown.

2. *Class III power supply:* AC supplies to auxiliaries that are necessary for safe shutdown of the reactor and turbine are obtained from the Class III power supply with a stand-by diesel generator backup. These auxiliaries can tolerate short interruptions in their power supplies. This class of power supply comprises

Two medium-voltage buses supplied from the secondary windings of the two transformers on the Class IV primary medium-voltage buses, which supply power to the pumps in the service water system, ECCS, moderator circulation system, shutdown cooling system, HTS feed lines, steam generator auxiliary feed line, and the air compressors and chillers.

Several low-voltage buses.

3. *Class II power supply:* Uninterruptible AC supplies for essential auxiliaries are obtained from the Class II power supply, which comprises

Two low-voltage AC three phase buses that supply critical motor loads and emergency lighting. These buses are each supplied through an inverter from a Class III bus via a rectifier in parallel with a battery.

Three low-voltage AC single-phase buses that supply AC instrument loads and the station computers. These buses are fed through an inverter from Class I buses, which are fed from Class III buses via rectifiers in parallel with batteries. In the event of inverter failure, power is supplied directly to the applicable low-voltage bus and through a voltage regulator to the applicable instrument bus. If a disruption or loss of Class III power occurs, the battery in the applicable circuit will provide the necessary power without interruption.

4. *Class I power supply:* Uninterruptible direct current (DC) supplies for essential auxiliaries are obtained from the Class I power supply, which comprises

Three independent DC instrument buses, each supplying power to the control logic circuits and to one channel of the triplicated reactor safety circuits. These buses are each supplied from a Class III bus via a rectifier in parallel with a battery.

Three DC power buses that provide power for DC motors, switchgear operation, and for the Class II AC buses via inverters. These DC buses are supplied from Class III buses via a rectifier in parallel with batteries.

5. *Automatic transfer system:* In order to ensure continuity of supply in the event of a failure of either the unit or system power, an automatic transfer system is incorporated on the station service buses. Transfer of load from one service transformer to the other is accomplished by

A manually initiated transfer of power under normal operating conditions, or an automatically initiated transfer for mechanical trips on the turbine.

A fast, open transfer of power is supplied automatically to both load groups of the Class IV power supply system, when power from one transformer is interrupted. This fast transfer ensures that the voltage and the phase differences between the incoming supply and the residual on the motors have no time to increase to a level that would cause excessive inrush currents.

A residual voltage transfer, comprising automatic closure of the alternate breaker after the residual voltage has decayed by approximately 70%. This scheme is time-delayed and may require load shedding and could result in reactor power cut-back. It is provided as a backup to the above transfers.

6. *Station battery banks:* The station battery banks are all on continuous charge from the Class III power supply and in the event of a Class III power disruption will provide power to their connected buses.
7. *Stand-by generators:* Stand-by power for the Class III loads is supplied by diesel generator sets, housed in separate rooms with fire-resistant walls. Redundant diesel generators are available, capable of supplying the total safe shutdown load of the unit. The Class III shutdown loads are duplicated, one complete system being fed from each diesel generator. In the event of a failure of Class IV power, diesel generators will start automatically. The generators can be up to the required speed and ready to accept load in less than 2 min. The total interruption time is limited to 3 min. Each generator automatically energizes half of the shutdown load through a load-sequencing scheme. There is no automatic electrical tie between the two generators, nor is there a requirement for them to be synchronized. In the event of one generator failing to start, the total load will be supplied from the other generator.
8. *Emergency power supply system:* The emergency power supply system can provide all shutdown electrical loads that are essential for safety. This system and its buildings are seismically qualified to be operational after an earthquake. The system provides a backup for one group of safety systems (SDS2, emergency water supply [EWS], and secondary control area) if normal electric supplies become unavailable or if the main control room becomes uninhabitable. The system comprises two diesel generating sets, housed in separate fire-resistant rooms, which are self-contained and completely independent of the station's normal services. There is adequate redundancy provided in both the generating distribution equipment and the loads.

*4.2.1.3.7.4 Station Instrumentation and Control* Digital computers are used for station control, alarm annunciation, graphic data display, and data logging. The system consists of two independent digital computers (DCCX and DCCY), each capable of station control. Both computers run continuously, with programs in both machines switched on, but only the controlling computer's outputs are connected to the station equipment. In the event that the controlling or directing computer fails, control of the station is automatically transferred to the *hot* stand-by computer. In the event of a dual computer failure, the station will automatically shut down.

Individual control programs use multiple inputs to ensure that erroneous inputs do not produce incorrect output signals. This is achieved by rejecting

- Analog input values that are outside the expected signal range
- Individual readings that differ significantly from their median, average, or other reference

A spare computer is provided as a source of spare parts for the station computers. It is also used for

- Program assembly and checkout
- Operator and maintainer training
- Fault diagnosis in equipment removed from the station computers

Computerized operator communication stations replace much of the conventional panel instrumentation in the control room. A number of human-machine communication stations, each essentially comprising a keyboard and color cathode ray tube monitor, are located on the main control room panels. The displays provided on the monitors include

- Graphic trends
- Bar charts
- Status displays
- Pictorial displays
- Historical trends

Printed copies of the displays on any display monitor the operator wishes to record can be obtained from the line printers. The digital computers are also used to perform the control and monitoring functions of the station and are designed to be

- Capable of handling normal and abnormal situations
- Capable of automatically controlling the unit at startup and at any pre-selected power level within the normal loading range
- Capable of automatically shutting down the unit if unsafe conditions arise
- Tolerant of instrumentation failures

The functions of the overall station control system are performed by control programs loaded into each of the two unit computers. The major control function programs are

- The reactor regulation program, which adjusts the reactivity control devices to maintain reactor power equal to its desired set point

- The steam generator pressure program, which controls steam generator pressure to a constant set point by changing the reactor power set point (normal mode), or by adjusting the station loads (alternate mode)
- The steam generator level control program, which controls the feedwater valves in order to maintain the water level in the steam generators at a reactor power-dependent level set point
- The HTS pressure program, which controls the pressurizer steam-bleed valves and heaters in order to maintain HTS pressure at a fixed set point

There are also programs for

- HTS control
- Moderator temperature control
- Turbine run-up and monitoring
- Fuel-handling system control

There are two modes of operation of the reactor: the *reactor-following-turbine* mode and the *turbine-following-reactor* mode.

In the reactor-following-turbine mode of operation, the turbine generator load is set by the operator: the steam generator pressure control program *requests* variations be made to reactor power in order to maintain a constant steam generator pressure. This control mode is termed reactor-follows-turbine or *reactor-follows-station loads*.

In the turbine-following-reactor control mode (i.e., turbine-follows-reactor), station loads are made to follow the reactor output. This is achieved by the steam generator pressure-control program, which adjusts the plant loads in order to maintain a constant steam generator pressure. This mode is used at low reactor power levels, during startup or shutdown, when the steam generator pressure is insensitive to reactor power. It is also used in some upset conditions when it may not be desirable to maneuver reactor power.

#### **4.2.1.4 Features of Other HWRs**

##### *4.2.1.4.1 Integrated Four-Unit CANDU HWRs*

Ontario Power Generation and Bruce Power utilities operate the majority of operating CANDU plants as four-unit stations either as 525-MW or 540-MW units (Pickering A and B) or 825-MW units (Bruce A and Bruce B) or 935-MW units (Darlington). These integrated four-unit stations, although nominally similar (featuring common control room area, emergency coolant injection, and electrical and service water systems) differ in the number of channels, number of fuel bundles in the channels, outlet temperatures, and support components inside and outside the channels. There are also differences in the shutdown mechanisms as well as the design of shield tank and shielding material.

The HTS differs also by having preheaters separate from the steam generators and in the number of steam generators and in Bruce A having the steam generators attached to a common steam drum. Also, the containment of each unit is connected to a large vacuum building by shafts and sealed with valves that can be opened after a severe system accident to draw radioactivity into the vacuum building.

#### 4.2.1.4.2 Carolinas–Virginia Tube Reactor

The Carolinas–Virginia tube reactor (CVTR) heavy water-cooled and moderated pressure tube reactor was built as a power demonstration reactor at Parr, South Carolina. Construction started in 1960 and the reactor was completed and connected to the grid by the end of 1963. The CVTR generated 19 MWe and, after about 4 years of operation, a planned experimental program having been completed, it was shut down and eventually decommissioned. The reactor circuit contained many of the features of later pressurized heavy water-cooled and moderated reactors, including a pressurizer and, notably, an oil-fired superheater to upgrade the quality of steam being fed to the turbine.

#### 4.2.1.4.3 Pressure Tube Boiling Light Water Coolant, HWM Reactors

Four countries have evaluated the reactor system in which light water is brought to boiling in vertically oriented pressure tubes, the steam–water mixture being sent to steam drum separators and the steam used directly to drive a turbine. The arrangement simulates the conventional recirculation boiler. The Russian RBMK is a similar type of reactor, except that graphite is used as the moderator.

Each country had different reasons for initiating studies of this type of reactor. In the United Kingdom, there was a search for a more economic thermal reactor for electricity production than the Magnox or the advanced gas-cooled reactors, and one that would avoid the use of graphite as a moderator as well as the use of a large-pressure vessel. Pressure vessel property changes during life and potential problems with resealing the vessel after refueling were then current concerns. In Canada in the early 1960s, there was concern that the heavy water coolant system in the PHWRs would not be sufficiently leak-tight to produce acceptable heavy water losses, and there was a desire to develop a less capital-intensive reactor by using light water coolant.

In Italy, the intention was to develop a reactor that was independent of enriched fuel, while in Japan the HWR was seen as part of a future fuel recycling strategy where spent fuel from PWRs would be recycled through HWRs to make use of the fissile material remaining in the fuel.

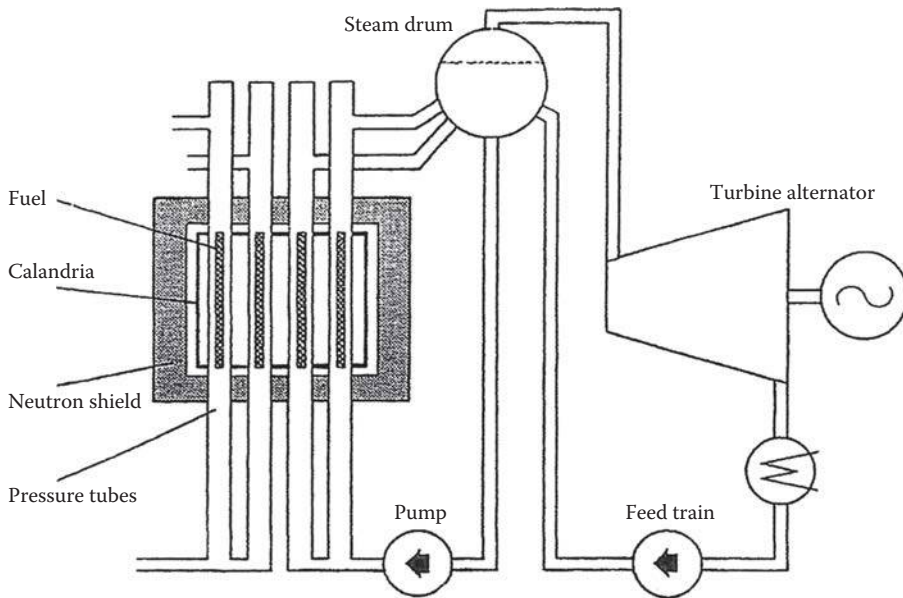
Thus, a prototype reactor was built in each country using the experience gained, which is described in subsequent sections. It should be noted that an advanced CANDU reactor (ACR) design was under development by AECL that incorporated a light water coolant, heavy water moderator, and enriched fuel with horizontal channels. However, work on this project has been suspended to concentrate on the updated CANDU 6 design. In India a 300 MWe vertical pressure tube reactor using HWM and boiling light water coolant is under development. This reactor is designed to use thorium fuel with additional fissile material to sustain a thermal breeding cycle.

The general characteristics of a pressure tube boiling light water HWM (BLW-HWM) system are illustrated in Figure 4.3. The pressure tubes (vertically oriented) contain the fuel and the light water entering at the bottom of the fuel channel is brought to boiling, about 10 wt% of water being converted to steam. The steam–water mixture passes to the steam drums and virtually dry, saturated steam is supplied directly to the turbine. The exhaust steam is condensed and returned to the water space in the steam drums via a feed heating train.

#### 4.2.1.4.4 Steam Generating HWR

Steam Generating HWR (SGHWR) started operation in 1968 with a designed output of 100 MWe. The use of light water coolant and heavy water moderator means that with the choice of the appropriate fuel-to-moderator ratio, the void coefficient could be made to





**FIGURE 4.3**  
Typical circuit for a boiling light water cooled heavy water moderated reactor.

approach zero and even to be slightly negative. The reactor operated for 23 years at average capacity factor of 60%. However, a commercial design was not considered economic and the program was cancelled in 1977.

#### 4.2.1.4.5 *Gentilly-1*

The Gentilly-1 pressure tube reactor was a 250-MWe HWM and boiling light water-cooled design fueled with natural uranium dioxide. The reactor concept had been developed in the early 1960s, and in 1966 the reactor was committed for construction. First, power was produced in 1971 and full power attained in May 1972. It was shut down in April 1979, and by 1984 had been decommissioned.

#### 4.2.1.4.6 *Fugen*

The 165-MWe Fugen reactor was the prototype of what was to be a line of 600-MWe reactors that would form, in conjunction with PWRs, the Japanese fuel recycling strategy. A 600-MWe design was found to be too costly and the development of MOX fuels led to the demise of the project.

#### 4.2.1.4.7 *Cirene*

The Cirene reactor was a 40-MWe prototype power plant constructed at Latina, 80 km south of Rome. Construction started in 1976 and completion was scheduled for 1984. Commissioning stopped in 1988, before work to reduce the positive void reactivity coefficient was complete, by the general moratorium on nuclear reactor operation imposed by the Italian Government following the Chernobyl accident.

#### 4.2.1.5 Summary

The comments given in the earlier section can be applied generally to this line of BLW-HWM reactors. The use of light water coolant, dispensing with steam generators, is economically attractive. These advantages have been offset by necessary design modifications in the fuel and other features, and by a necessary increase in the number of channels needed to achieve the same output as pressurized PHWR versions. However, provided these problems are addressed in revised designs, this line of reactors should be cheaper to build and to operate than the heavy water-cooled versions. A conceptual design for a reactor which uses plutonium and thorium fuel is being pursued by India as part of its overall plan for nuclear-based electricity generation and is an example where the advantages of this reactor type may still be realized.

#### 4.2.2 Characteristics of Pressure Vessel HWRs

HWRs of the pressure vessel type have been designed and constructed in Sweden, Germany, and Argentina. The main references of this line are the Ågesta reactor in Sweden (shutdown), the MZFR reactor in Germany, and the Atucha-1 and Atucha-2 reactors in Argentina.

##### 4.2.2.1 Ågesta

In Sweden, the first pressure vessel pressurized HWR was constructed at Ågesta. This was a project that combined the objectives of two separate concepts: one for a district heating reactor and the other for a heat and power reactor. The pressure vessel reactor was conceived as a 65 MWth prototype plant that was to supply district heating and electricity (10 MWe). The reactor was located in an underground chamber excavated in solid rock and serviced a suburb of Stockholm. The reactor operated with a good degree of reliability. Operation was interrupted over the summer months when district heating was not required. The reactor was shut down in 1975 and decommissioned because it had ceased to be an economical source of power.

##### 4.2.2.2 MZFR

The 57-MWe MZFR reactor was built by Siemens-KWU at the Karlsruhe Research Center for limited electricity supply and district heating. It was the prototype for the Atucha-1 and Atucha-2 reactors built in Argentina. The principal features are similar to those incorporated in Atucha-1.

##### 4.2.2.3 Atucha-1

The reactor core is approximately cylindrical and consists of vertical fuel assemblies located in the same number of fuel channels. The coolant channels are arranged on a triangular lattice pitch and penetrate the top and bottom plenums located inside a cylindrical pressure vessel containing the moderator heavy water at a similar pressure to the HTS.

As reactor heavy water coolant and the moderator heavy water are kept at nearly the same pressure, thin-walled tubes were sufficient to separate the fluids. The fuel channel tubes can thus be categorized as reactor internals. Also, the two systems use the same auxiliary systems to maintain water quality.

The moderator water at a temperature of 210°C is used to preheat the feedwater, producing a net efficiency of operation of approximately 29% for Atucha-1 and 32% for Atucha-2.

Reactivity can be controlled by *black* and *gray* absorbers arranged in groups or banks of three azimuthally symmetric absorber rods. These penetrate the vertical matrix of fuel channel tubes at an angle to the vertical. The reactivity can also be controlled by boron additions and by varying moderator temperature.

The fuel is a long string with 37 elements with extensions to allow the fueling machine to extract the fuel. It can be refueled on-power with a single fueling machine operating above the reactor vessel cover head.

The containment is a spherical stainless steel housing which is protected against external impacts by the surrounding reinforced concrete reactor building.

The HTS consists of the reactor vessel, two steam generators, two primary pumps, and the pressurizer that keeps pressure at approximately 11.65 MPa. The system has two loops, and for Atucha-1 the exit temperature from the pressure vessel is ~300°C and the inlet temperature of the return coolant into the pressure vessel is 265°C.

Atucha-2 is a larger version of Atucha-1 with more channels. Construction of the reactor was delayed for a number of years because of limitations in funding, but is now nearing completion and reached full power early in 2015.

#### **4.2.2.4 Characteristics of HWM, Gas-Cooled Reactors**

##### *4.2.2.4.1 Introduction*

Four gas-cooled pressure tube reactors of relatively small size were built in the 1960s with the object of exploring the use of CO<sub>2</sub> as a heat transport fluid in combination with heavy water moderation instead of graphite. The reactors had innovative fuel designs and most had the pressure tubes vertically oriented, although the most successful unit, the EL4 plant in France had the pressure tubes horizontal.

The potential advantages were low-neutron absorption by the coolant and high outlet coolant temperatures available at moderate pressures. The disadvantages lay in the relatively poor heat transfer and heat transport properties of CO<sub>2</sub>.

The advantage of using CO<sub>2</sub> is that the heat transport gas can be heated to higher temperatures than is possible with water and thus achieve higher thermal efficiencies at the turbine. Typically, the temperature reached by the CO<sub>2</sub> is about 500°C. The heat is exchanged in steam generators to produce the steam to drive turbines.

##### *4.2.2.4.2 The EL4 Reactor*

The EL4 reactor (70 MWe) was constructed at the Mont d'Aree site near Brennilis, France. The heavy water moderator is contained in a horizontal cylinder 4.6-m long and 4.8 m in diameter. The 216 fuel channels, arranged on a square pitch of 234 mm, are contained in Zircaloy tubes. The Zircaloy pressure tubes (107-mm inside diameter and 3.2-mm wall thickness) can operate at a low temperature by virtue of their being thermally isolated from the hot CO<sub>2</sub> gas by a stainless steel guide tube and by thermal insulation between the guide tube and the pressure tube.

The EL4 reactor started up in 1965. It had initial problems with steam generators, which were overcome in the first 2 years of operation, and it was not able to use beryllium alloy fuel cladding as intended. However, it operated successfully until 1985 when it was shut down, together with some other gas-cooled reactors, because Electricité de France decided to concentrate on PWRs.

The advantages of this reactor were the relatively low cost per unit of electricity and the low fields occurring in the reactor vault. As a result of the absence of activity transport, the reactor face and vault were accessible when the reactor was on-power.

#### 4.2.2.4.3 The Niederaichbach Reactor

The 100-MWe Niederaichbach reactor was designed by Siemens in the early 1960s and constructed between 1965 and 1970 in the Isar Valley, about 70 km northwest of Munich. The reactor contained 351 vertical channels on a square pitch of 24.5 cm. The channels penetrated a tank or calandria containing heavy water. Basic control was achieved by adding a burnable poison,  $\text{CdSO}_4$ , to the moderator. The moderator level could also be adjusted and the moderator dumped to shut down the reactor.

The Niederaichbach reactor reached full power in 1970 and was connected to the grid in 1973. It was shut down in 1974 when it was deemed to have become uneconomic compared with other water-cooled reactors, and the subsequent decommissioning activity had the objective of demonstrating the ability to return a reactor site to a greenfield condition.

#### 4.2.2.4.4 Lucens Reactor

The Lucens reactor was constructed in the period 1962–1968 in underground caverns at a site between Lausanne and Berne. It was a 30 MWth/8.3 MWe pressure tube reactor with  $\text{CO}_2$  cooling and was designed to combine features of the French reactors and the British Magnox units with heavy water moderation. The reactor only operated for a few months before a 3-month shutdown was required for maintenance. During the shutdown, water accumulated in some channels causing corrosion of the magnesium alloy fuel cladding and resulted in partial blockage of the fuel due to accumulation of corrosion products. At startup, the flow blockage remained undetected during the subsequent rise to power owing to flow bypass of the blocked subchannels. The cladding melted and further obstructed the flow, leading to a uranium fire, graphite column contact with the pressure tube as a result of bowing, and pressure tube failure by overheating and subsequent rupture. The calandria tube was also ruptured. As a result, the reactor was shut down and eventually decommissioned. Before commissioning, it was recognized that the design was not supported by the Swiss electrical utilities, and its operation was intended for experimental purposes.

### 4.2.3 Unique Features of HWR Fuel Channel Technology

Fuel channels are a common feature of HWRs of all types. Components of fuel channels can be grouped into three main elements:

- Pressure-retaining components, including the out-of-core channel extensions and the mechanical closures accessed by fueling machines in refueling the channels
- Channel support components, which are more obvious as the end bearings and spacer/calandria tube components in CANDU 6 horizontal channels
- Channel internals, which may include radiation-shielding plugs, thermal shielding plugs, flow straighteners/modifiers, fuel supports, and the fuel

Because many of the fuel channel designs were *one-offs*, there was little development of most concepts. In the case of the CANDU channel, development has been toward larger diameters and longer channels as the means of achieving higher power outputs at

higher temperatures. This part of the development has now reached a limit as regards pressurized water conditions, and development activities are now being directed toward achieving a longer channel life with limited modifications being made to the basic design.

In the previous sections, several reactor designs using heavy water moderation are briefly described. Based on the pressure tube boundary conditions, the fuel channel designs can be divided into three types:

- Channels with a high-temperature, high-pressure boundary
- Channels with a high-temperature, low-pressure boundary
- Channels with a low-temperature, moderate-pressure boundary

The aspects of fuel channel technology to be described in the various reactor designs will thus be addressed on the basis of the above divisions.

#### ***4.2.3.1 Channels with a High-Temperature, High-Pressure Boundary***

The CANDU fuel channels are of this type, and typified by the CANDU 6 channel illustrated in Figure 4.4. Pressure-retaining components are the pressure tubes, end fittings, and closure seals. The Zr–2.5%Nb pressure tube (104-mm inside diameter, with a 4-mm wall thickness and 6.1 m length) is produced by extrusion, cold working, and stress relieving. The tube is roll-expanded into AISI type 403 stainless steel end fittings by a procedure that leaves low-tensile residual stresses at the end of the rolled zone. The total length of the fuel channel, including the end fittings, is 10.1 m. The channel is accessed at each end for fuel removal and replacement. New fuel is inserted at the inlet end and used fuel removed at the outlet end, and there are 12 bundles in each channel. (In the initial build of Bruce reactors there are 13 bundles in each channel and fuel was inserted at the outlet end.) The fuel, in the form of 37-element bundles, can be stored in the rotating magazine of the fueling machine before or after removal.

The pressure tube and contents are supported by linear sliding bearings at each end of the reactor. The journal bearings are formed by ring bearings on the end fittings mating with sleeve bearings in the lattice tube. The calandria tube supports the in-core section of the pressure tube through toroidally coiled spacers that accommodate relative axial and diametral movement between the pressure tube and the calandria tube.

Positioning assemblies at each end of the channel locate the channel in the reactor. Typically, the channel is positioned to allow elongation (caused by neutron irradiation) to take place on the full length of the bearings at one end by locking the end fitting at the other end to the positioning assembly. At half-life, the channels are relocated by releasing the channel and pushing it to the inboard extremity of the unused bearing length and locking it to the other positioning assembly. Each end fitting contains (1) a liner tube to prevent the fuel bundles experiencing cross-flow on entering or leaving the fuel channel; (2) a shield plug which supports the fuel at the outlet end and whereby flow is directed into the annulus between the liner tube and the end fitting body (and out through the side port) or from the liner annulus, through the shield plug and into the fuel without causing instability in the fuel; and (3) a closure plug which can be opened by the fueling machine. In the CANDU 6 channel, the seal forms part of a flexible dome that is pressed against a step in the end fitting in order to achieve a pressure face seal.

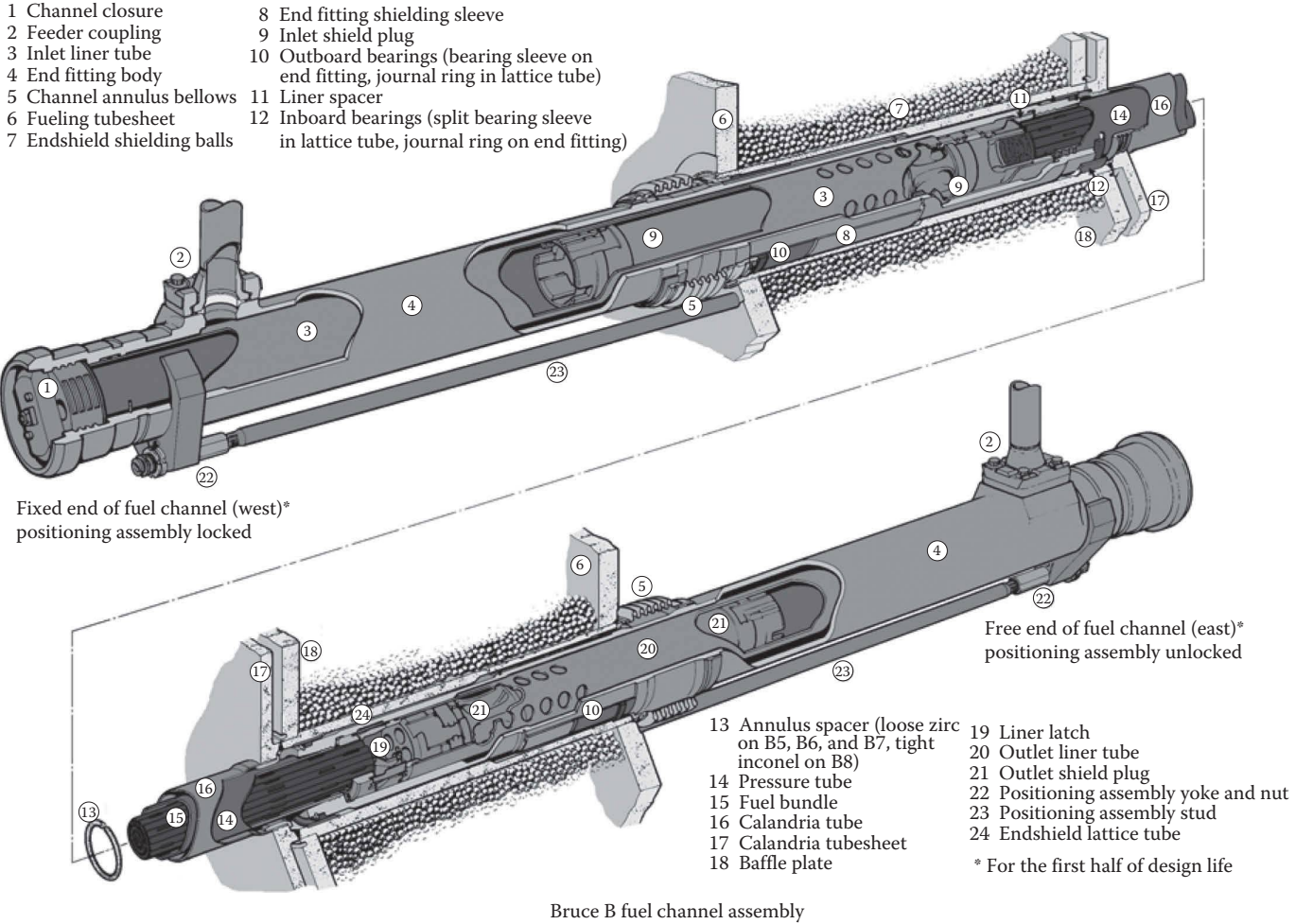


FIGURE 4.4  
 Bruce B fuel channel assembly.

In response to the neutron flux, high temperatures, water environment, and wear, the channels (mostly the pressure tube) change as follows:

- *Dimensions change:* The pressure tubes sag, expand, and elongate. Typically, a CANDU 6 Zr-2.5%Nb pressure tube will expand >4%, elongate by 180 mm, and sag up to 76 mm in 30 years. The calandria tubes sag (and support the pressure tubes) and the pressure tube will sag between spacers but is designed not to make contact with the calandria tubes.
- Pressure tubes pick up hydrogen (as deuterium) from corrosion and crevice reactions. The concentration of hydrogen after 30 years is predicted to be just above the terminal solid solubility at operating temperatures and the presence of hydrides during operation is not considered to influence behavior. The surface oxide resulting from corrosion has no structural effect.
- Mechanical properties of the in-core components change as a result of the fast neutron flux damage. The strength increases and ductility and fracture toughness decrease to shelf levels that are acceptable for service. Recent developments in pressure tube technology have made the pressure tubes more resistant to decreases in fracture toughness caused by irradiation.
- *Pressure tubes wear:* Light scratching by fuel bundle movement can occur. Debris, which can enter the channels from maintenance activities, can become trapped in the fuel and wear the pressure tube through vibration in the flowing water.

Each of these types of change must be monitored by inspection of periodically removed pressure tubes. Debris fretting must be prevented by operating with a *clean* HTS.

The fuel channels of the Pickering, Bruce, and Darlington reactors and the Indian series of reactors are nominally similar but differ in specific features.

#### 4.2.3.1.1 *The SGHWR Fuel Channel*

The pressure tube was of Zircaloy 2. The pressure tube was reduced in diameter at the lower rolled joint where it was rolled into a hub that was, in turn, welded to the stainless inlet piping.

In the upper part of the channel, the pressure tube was rolled into the hub of the upper standpipe which had a side port connected for the coolant outlet, the emergency cooling inlet and, at the top, the closure seal for refueling.

#### 4.2.3.1.2 *The Gentilly-1 Fuel Channel*

The pressure tube in this reactor is a quenched and aged (heat treated) Zr-2.5%Nb and, at 2.41-mm wall thickness, was significantly thinner than the pressure tubes in PHWRs.

An insert was required to attach the thin-walled pressure tube to the end fittings. The calandria tube was separated from the pressure tube by spacers supported on interlocking support rings. The fuel was attached to a central structural tube, which was supported at the bottom and at the top by lower and upper shield plugs, respectively.

#### 4.2.3.1.3 *The Fugen Fuel Channel*

As with the Gentilly-1 pressure tube, the Fugen channel was also made of heat-treated Zr-2.5%Nb, and the wall was only 2.2-mm thick, which also required the use of inserts in forming the rolled joint.

The lower rolled joint has an internal insert to *sandwich* the pressure tube between the insert and the end fitting. However, the upper end fitting sandwiches the pressure tube between an external insert and the end fitting. An upper extension tube connects the channel to the external piping via a reducer. The connection to the inlet feeder is made via a side port and the closure plug at the bottom makes a bore seal with the end fitting extension using a flexed dome component. The Fugen channel has functioned without problems.

#### 4.2.3.1.4 *Cirene Fuel Channel*

The Cirene fuel channel is of similar design to the Gentilly-1 and Fugen channels. The pressure tube is of Zircaloy 2 (106.1-mm inside diameter, 3.15-mm wall thickness).

### 4.2.3.2 *Channels with a High-Temperature, Low-Pressure Boundary*

#### 4.2.3.2.1 *Atucha-1 Fuel Channel*

In the original design of the Atucha-1 channel, the main shroud tube enclosing the fuel assembly was made of Zircaloy 4 and comprised a seam-welded tube (108.2-mm inside diameter and 1.6- or 1.72-mm in wall thickness). A thin (0.1-mm wall thickness) Zircaloy tube, dimpled to maintain separation, surrounded the shroud tube between it and the seam-welded Zircaloy 4 insulation tube (0.4-mm wall thickness). These were attached to austenitic stainless steel end fittings. The bottom end fitting sat in the bottom plenum lattice port allowing a small gap for the circulation of heavy water into the moderator space. The upper end was fastened to the upper plenum.

In the replacement channels of Atucha-1 and proposed for Atucha-2, the Zircaloy 4 isolation tube has been eliminated in favor of a shroud tube and a surrounding insulation tube.

### 4.2.3.3 *Channels with a Low-Temperature, Moderate-Pressure Boundary*

#### 4.2.3.3.1 *The EL4 Fuel Channel*

The pressure boundary tube of the EL4 channel contains a Zircaloy 2 tube rolled into the end shields of the moderator tank. Inside the Zircaloy tube is a stainless steel guide tube, insulated from the Zircaloy 2. The guide tube thus sees the 233°C–475°C temperatures of the CO<sub>2</sub> and carries the fuel assemblies. The seal plugs at the channel ends incorporate a ball valve for fueling machine access.

#### 4.2.3.3.2 *Niederaichbach Fuel Channel*

The Zircaloy 2 pressure tube operated at 100°C and was isolated from the hot CO<sub>2</sub> by a thin foil tube and a stainless steel insulating tube.

#### 4.2.3.3.3 *Lucens Fuel Channel*

In the Lucens fuel channels, the Zircaloy 2 pressure tube was kept to the temperature of the inlet CO<sub>2</sub> gas (225°C) by passing the inlet gas between the carbon matrix fuel and the pressure tube. Low-temperature CO<sub>2</sub> also flowed in the annulus between the pressure tube and the calandria tube.

#### 4.2.3.3.4 *CVTR Fuel Channel*

The fuel channels of the CVTR were made to a U-tube design, each leg containing one fuel assembly. The pressure tubes were made of Zircaloy 2. The fuel contained in the pressure tube was isolated from the wall of the pressure tube by inner and outer circular thermal



baffle tubes, 0.7 and 0.3 mm in wall thickness, respectively. In addition, a hexagonal flow baffle tube, positioned inside the thermal baffles, concentrated the flow through the fuel.

The pressure tube was in contact with the moderator water and heat shielded from the fuel and thus operated in a cold pressurized condition. The pressure tube was rolled into the U-fittings at the bottom of the reactor and into end fittings at the top of the reactor.

The channel tubes were made of aluminum alloy and arranged on a square lattice pitch. The channel tubes were isolated from the fuel assembly by a protective internal magnesium alloy tube which surrounded 150–200 small diameter (4 mm) fuel rods of natural uranium arranged in seven concentric rings around the center rod.

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## 4.3 Heavy Water

### 4.3.1 Purpose of Heavy Water

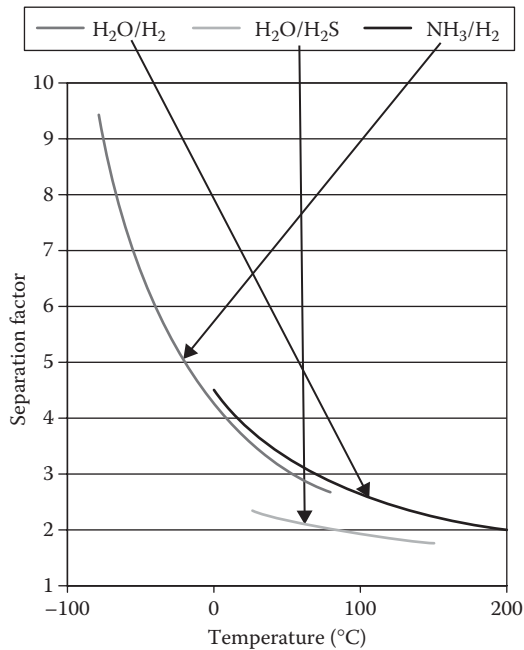
For thermal—as opposed to fast—reactors, neutrons must be slowed down from the high speeds at which they are emitted by fissions, a process known as *moderation*. Normal hydrogen (protium—symbol H as an isotope), because its atomic mass (1) is almost identical to that of a neutron, is the most effective moderator. To achieve high density and because oxygen atoms are almost transparent to neutrons, hydrogen is normally deployed in the form of water. Despite its unequalled performance in slowing neutrons, protium suffers from a serious disadvantage of capturing so many neutrons that reactors using light-water moderation require significant enrichment in  $^{235}\text{U}$  of their uranium fuel. The rare heavier isotope of hydrogen, deuterium (atomic mass = 2, symbol D) is a poorer match to the mass of the neutron and so requires more collisions (and a larger moderator volume) to effect moderation but is almost immune to capturing neutrons. Consequently, a reactor moderated with deuterium in the form of heavy water ( $\text{D}_2\text{O}$ ) can use uranium with natural concentrations of  $^{235}\text{U}$ .

There is, therefore, a choice between using light water and burning fuel that must always be enriched in fissile material or doing a one-time enrichment of heavy water and natural uranium as fuel. Because separating the isotopes of hydrogen is comparatively easy, taking the heavy-water route would be the natural choice *except* that deuterium is rare, occurring only in abundances of around one part in 7000 terrestrially. This has the effect of having to process very large volumes of feed material to produce heavy water and is the main reason behind its relatively high cost, typically \$300/kg.

### 4.3.2 Heavy Water Production

The standard textbook on isotope separation is Benedict et al. (1981) *Nuclear Chemical Engineering*. It provides much more detail than is possible here. Because of the large (2:1) mass ratio of deuterium to protium, the affinities for the two isotopes are quite different in many chemical species. Over the practical operating temperature ranges for the different processes, with water and hydrogen, the equilibrium ratio varies from 3.8 excess in water at 25°C to 2.0 at 200°C. With ammonia and hydrogen, the ratio varies from 6.0 at –40°C to 3.5 at +30°C. With water and hydrogen sulfide, the ratio is around 2 and relatively weakly affected by temperature, ranging from 2.3 at 32°C to 1.8 at 130°C. Figure 4.5 shows the effect of temperature on the separation factors.

However, for a process, one needs a significant difference in equilibrium and adequate rates of reaction. Of the three systems mentioned above, only water and hydrogen sulfide come rapidly to equilibrium. The other systems require catalysts to reach workable rates of

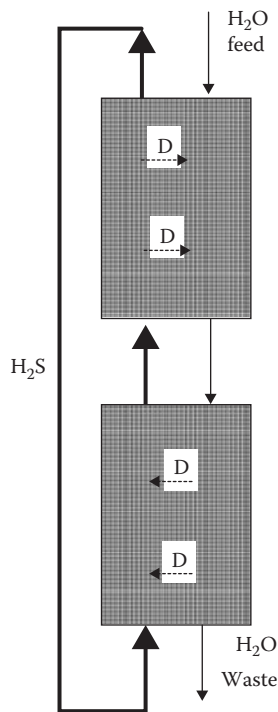


**FIGURE 4.5**  
Equilibrium constants.

exchange. It was in this context that a process based on water and hydrogen sulfide became the near-standard technology for heavy water production even though hydrogen sulfide is a dangerous and highly corrosive substance and the variation of the separation factor is quite weak (2.3–1.8). The workable range of temperatures is limited (to between just under 30°C by formation of a solid gas hydrate and to around 130°C by the vapor pressure of water) where the operating pressure is bounded by gas liquefaction at pressures above 2.2 MPa. The two advantages of this system are that one species is a gas and the other a liquid, and that deuterium can enter the process as water, with its limitless abundance.

In a conceptual, ideal process, water would be continuously converted into hydrogen sulfide and back into water in the arrangement illustrated in Figure 4.6. This is known as a *monothermal* process because deuterium exchange occurs at only one temperature. Between the two conversions, the two species are repeatedly contacted. With an equilibrium ratio around 2.3 at 30°C, the descending water grows steadily richer in deuterium and the hydrogen sulfide is steadily stripped of its deuterium content. At the bottom, water rich in deuterium is converted in hydrogen sulfide at the same concentration, providing an excellent driving force behind the water's enrichment. A large driving force leads to relatively few contact steps in the exchange column. At the top, water at less than half the natural deuterium abundance is produced and the difference from natural deuterium gives the extractive capacity of the process. Unfortunately, there is no practical way of implementing this conceptual process since the conversions of water to hydrogen sulfide and the reverse are too difficult to be economic.

So instead of this monothermal process, a bithermal process (usually known as Girdler sulfide or G-S) was developed. Instead of converting water into hydrogen sulfide, a second *hot* contact tower operating at a higher temperature is included (Figure 4.6). At 130°C, the equilibrium ratio for deuterium is 1.8. By having twice the molar flow of hydrogen sulfide to



**FIGURE 4.6**  
Schematic of a bithermal water–hydrogen sulfide process.

that of water, deuterium can be made to move back from the water into the hydrogen sulfide stream in the hot tower. Deuterium still moves from hydrogen sulfide to water in the cold tower but with lower driving forces because the water must absorb twice as much deuterium as would occur with the monothermal process. The deuterium content of the hydrogen sulfide leaving the top of the *cold* tower is low, enough to allow it to be recycled to the bottom of the hot tower and to initiate extraction of deuterium from the downflowing water stream.

Because the variation in equilibrium ratio is rather small, there is only a small difference in deuterium concentrations between the feedwater entering the top of the cold tower and the water leaving the hot tower. In practice, about 17% of natural water's deuterium content can be extracted, compared with the 50% that would be possible if a monothermal process were practicable. This further raises the already large mass of water that a G-S plant must process to about 37,000-times the product, including allowance for heavy water being 10% heavier than normal water. In consequence, G-S plants are very large (Figure 4.7).

Despite its intrinsic limitations and difficulties, the G-S process was commercialized in the United States and subsequently on a larger scale in Canada and elsewhere. It had provided the preponderance of the 20,000 tons of heavy water produced worldwide to 2007. Production in Canada peaked around 1980 at around 1200 tonne/a from four plants. Since then, G-S production has been phased-out in Canada because ample stockpiles exist and more ACR designs require less heavy water. G-S production does continue in Romania and India.

Because of the G-S process's limitations, alternatives have been developed. The alternative that has been deployed in India and Argentina relies on ammonia–hydrogen exchange. This needs a catalyst and the potassium salt of ammonia,  $KNH_2$ , dissolved in the ammonia is used. Because of the high vapor pressure of ammonia, the exchange process is carried



**FIGURE 4.7**  
G-S plant.

out at around  $-30^{\circ}\text{C}$ . At that temperature, the catalyst has limited activity and its performance must be enhanced by devices that provide large surface area or intense agitation. The process has been configured as a monothermal and a bithermal process, with a hot tower temperature around  $40^{\circ}\text{C}$ , limited by the linked considerations of operating pressure and ammonia vapor pressure. A monothermal process requires cracking of ammonia to hydrogen and nitrogen below the exchange tower and their re-synthesis above the exchange tower. This is economically practical but means that the hydrogen flow is diluted, 25% of the gas being nitrogen. The potassium salt must also be stripped of deuterium and transferred to the re-synthesized ammonia. Configured monothermally or bithermally, plants must depend on processing a large stream of hydrogen or on an exchange step between ammonia and water. In the first instance, even the largest hydrogen production plants are only big enough to produce about 70 tonne/a of heavy water. (The typical scale of G-S plants has been 200–400 tonne/a.) Using ammonia–water exchange avoids this constraint, but the ammonia must be very carefully dried to avoid reaction between it and the potassium salt. Because the available amount of hydrogen is constrained or a fairly demanding exchange step is substituted for hydrogen feed, the process is usually configured to give at least 80% extraction of the available deuterium.

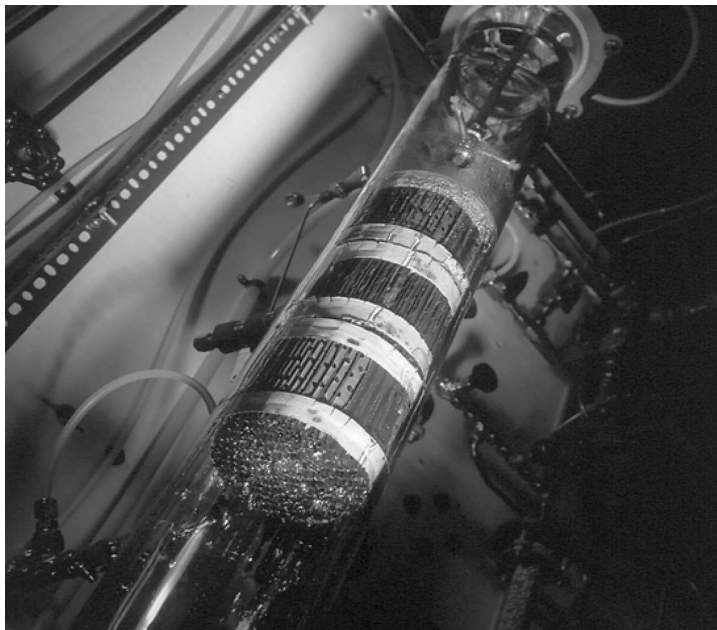
AECL did extensive development of a variant of the ammonia–hydrogen process based on aminomethane ( $\text{CH}_3\text{NH}_2$ ) rather than ammonia. This has better kinetics and a wider envelope of operating temperatures but can only be configured bithermally. This process was superseded by development of processes based on water–hydrogen exchange.

Processes based on hydrogen–water exchange are attractive because their operating temperature range (between  $25^{\circ}\text{C}$  and  $180^{\circ}\text{C}$ ) is easy to accommodate, neither substance is

toxic, and the equilibrium constant is comparable in size and variation with temperature with that for ammonia–hydrogen. The lack of a suitable catalyst was the only obstacle. Platinum was known to catalyze the exchange in the vapor phase, but the low-solubility of hydrogen in liquid water produced almost zero catalytic activity when water was present. This impasse was resolved by the invention of special wetproofed platinum catalysts, devised and extensively developed by AECL. A catalyst of this type is illustrated in Figure 4.8. It is composed of catalyst-coated hydrophobic plates where water vapor and hydrogen are able to exchange deuterium and hydrophilic plates where the water vapor can come to equilibrium in deuterium content with liquid water.

Based on this development, AECL built a three-stage prototype plant at a small hydrogen plant owned by Air Liquide in Hamilton, ON (Figure 4.9). The prototype characterized and proved (1) a monothermal first stage in which the steam–methane reformer did the conversion of deuterium-enriched hydrogen; (2) a second stage of enrichment using a bithermal hydrogen–water process; and (3) a final stage of enrichment using a further monothermal process with water electrolysis converting water into hydrogen. Electrolysis is an energy-intensive process but can easily be made leak-tight and so is particularly suited to handling the high-concentration heavy water product. Because the flows in this final stage are small, electrical energy consumption by electrolysis is not a large consideration. This combination of hydrogen–water processes is now AECL’s preferred technology for heavy water production. It is called combined industrial reforming and catalytic exchange (CIRCE).

A similar application of the monothermal stage with water electrolysis is now the reference design for *upgrading* in-service or recovered heavy water that has been contaminated with ingress of ordinary water. This process is called combined electrolysis and catalytic exchange (CECE). Should large-scale production of hydrogen by electrolysis



**FIGURE 4.8**  
Structured wetproofed catalyst for water–hydrogen exchange.



**FIGURE 4.9**  
Prototype CIRCE plant at Hamilton, Ontario, Canada.

displace large-scale hydrogen production by steam–methane reforming—most plausibly occasioned by constraints on CO<sub>2</sub> emissions and the costs that will be associated with sequestration and storage of carbon dioxide—production of heavy water entirely by CECE would become economic and would be simpler than CIRCE.

### 4.3.3 Tritium

Although deuterium's superiority as a moderator arises from its resistance to absorbing neutrons, neutron absorption does very occasionally occur. This produces the third, atomic mass 3, isotope of hydrogen. This is known as tritium and designated T—only the isotopes of hydrogen have their own accepted chemical symbols. Tritium has a half-life of 12.3 years and decays with a very weak beta emission. In a CANDU moderator, the concentration of tritium rises over many years to equilibrium at around 25 ppm. The moderator is a low-pressure system at about 75°C and can be made very leak-tight. Where heavy water is also used as the reactor coolant (which has been the norm but will not apply when light water is the proposed coolant), providing near-absolute leak-tightness is not possible. So a system of high-performance dryers is installed in zones where leakage is likely to occur. This water is usually mixed with some in-leakage of normal water and so has to be upgraded.

### 4.3.4 Upgrading to Remove Light Water

Upgrading of this recovered water has always been done by water distillation at subatmospheric pressure. This is a very simple process with no moving parts: water is boiled below an exchange column and condensed above the column. The equilibrium constant between liquid and water vapor is, however, very small (ranging from 1.055 at 14 kPa [abs] to 1.035 at 50 kPa [abs])—the practical range used in water distillation. As a consequence, hundreds of contact steps are needed to re-enrich the heavy water and strip deuterium

from the vapor stream so that it can be discarded. However, even though the internal flows within the exchange column must be more than orders of magnitude larger than the feed flow, they remain small in absolute terms and are economically manageable. The columns are rather large, typically around 50-m long and 0.8-m in diameter.

As mentioned previously, water distillation is now considered superseded by the CECE process, which is far more compact and cheaper.

#### 4.3.5 Tritium Extraction

Most operators of CANDU reactors have chosen to apply tritium extraction (also known as *detritiation*) after their reactors have operated for some years, and tritium levels have risen some way toward equilibrium. This is not required for considerations of environmental release but lower levels can simplify reactor maintenance.

For detritiation, variations of water–hydrogen exchange have been used for primary tritium extraction. The CECE process can be used effectively for further tritium enrichment, but above ~300 ppm water becomes sufficiently tritiated as to constitute a radiological hazard if direct skin contact occurs. In the elemental form, tritium is many orders of magnitude less hazardous because the human body neither absorbs nor significantly retains it. Consequently, cryogenic distillation of hydrogen isotopes is employed to produce further enrichment, which can extend to virtually pure tritium. This low-temperature distillation process operates at 22–24 K, but has a good equilibrium ratio (about 1.4).

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## 4.4 HWR Safety

### 4.4.1 Background

The nuclear safety philosophy and the regulatory licensing processes for heavy water pressure tube reactors developed relatively independently of other jurisdictions and, in part, were driven by the unique aspects of HWR design. Early operational experience with research reactors led to the requirements for functional and physical separation of special safety systems from process systems, the requirement for fast-acting SDSs, the requirements for demonstrating high availability and testing of passive safety systems, and the incorporation of an elementary risk-based licensing framework. All of these requirements were encapsulated at a high level in a licensing guidance developed in Canada and referred to as the *Siting Guide* (Hurst and Boyd 1972) which provided a simple and relatively effective licensing framework. Central to the Siting Guide framework were three concepts that in retrospect align well with the concept of the five levels of defense-in-depth scheme articulated by INSAG (IAEA 1996).

First, the process systems should be of high quality to limit the frequency of failures that could lead to accidents and the special safety systems should be highly reliable such that their unavailability is a low-probability condition. This corresponds to level one in the five-level defense-in-depth scheme.

Second, the *safety systems shall be physically and functionally separate from the process systems and from each other*. This ensured clear separation between level two and three defense-in-depth provisions. The process systems, providing level two defense-in-depth, include all systems necessary for control of the plant during normal power operation and shutdown conditions and for equipment and component protection, such as the reactor regulating

system, boiler pressure and feedwater control, shutdown cooling, moderator and end-shield cooling, service water, electrical power, and instrument air. The safety systems, providing level three defense in depth, include the reactor SDS, the ECCS, and containment. Additionally, a requirement was imposed to demonstrate high system reliability through online testing aimed at mitigating against lack of operating experience with any new design.

Third, the concept of risk was introduced in an elementary manner by requiring lower dose consequence limits for more probable failures (single failure of a process system) and applying higher dose consequence limits for less probable events (dual failures involving failure of a process system and coincident unavailability of a special safety system).

#### 4.4.2 Basic Nuclear Safety Functions

Similar to other thermal reactor designs, there are three basic functions that are necessary to mitigate the consequences of fission product releases during a postulated accident. The functions, referred to as the 3 Cs, are control, cool, and contain. *Control* refers to safe reactor shutdown. *Cool* involves the removal of heat—from the fuel produced by the fission process (at power) or by the decay heat after reactor shutdown—and rejection of the heat to a heat sink. *Contain* is simply the physical means to prevent the release of radioactive material to the atmosphere by provision of containment systems.

#### 4.4.3 Reactor Shutdown

The majority of pressure tube HWRs have protective functions in the reactor regulating system—a normally operating process system—that reduce reactor power when required to maintain process conditions in a safe operating range and provide protection of components and equipment. The power reduction can be gradual (power setback) or rapid due to absorber rod drop into the core (stepback).

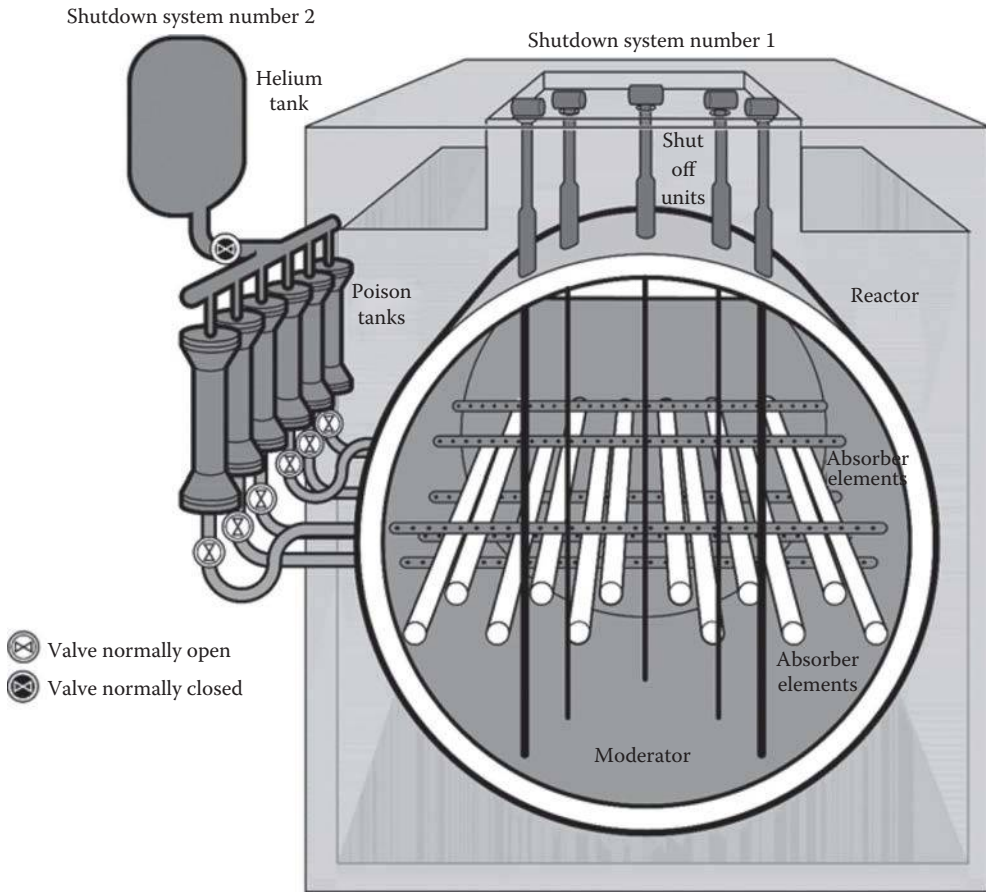
In addition to the reactor regulating system provisions, current HWR designs have two diverse, independent, fast-acting, equally effective and fail-safe safety SDSs, referred to as SDS1 and SDS2 (Figure 4.10).

SDS1 utilizes spring-loaded mechanical shutoff (absorber) rod mechanisms. Upon receipt of a reactor trip signal, an electromagnetic clutch in each mechanism is de-energized, releasing a stainless steel-clad cadmium absorber element that drops into the moderator under gravity, with initial rod acceleration provided by spring thrust. SDS1 is the primary method of quickly shutting down the reactor in an accident.

SDS2 utilizes liquid poison (absorber) injection into the moderator. Upon receipt of a reactor trip signal, fast-acting valves between a high-pressure helium tank and the poison tanks open to pressurize and inject the liquid poison into the moderator. Injection occurs from several perforated horizontal tubes in the calandria, through which gadolinium nitrate solution jets into the moderator. An earlier design, the 220-MWe Indian HWRs, employs a set of vertical empty tubes in the reactor core that can be filled with a liquid poison (lithium pentaborate solution).

Reactor shutdown is initiated by several trip parameters selected to ensure that there are at least two parameters on each SDS to detect any serious malfunction requiring a reactor shutdown. Triplicated sensors and instrumentation channels, independent of those used in regulation, are provided for each of the trip parameters. The sensors and instrumentation channels of the two SDSs are separate and, to the extent practicable, employ diverse components and designs. Typical trip parameters are high reactor power, HTS high and





**FIGURE 4.10**  
Shutdown systems.

low pressure, HTS high-temperature, low-flow in the HTS, low-pressure differential across the reactor core, high containment pressure, normal electric supply failure, and low-level in the steam generators.

Independent triplicated logic is employed in each system, with two out of three coincidence logic for generating the trip signal. Each trip channel is testable on power. In SDS1 and SDS2, loss of electrical power to the SDSs will result in the reactor shutdown mechanisms being deployed to ensure fail-safe operation, either by disengaging the electromagnetic shutoff rod clutches or opening the fast-acting valves to pressurize the liquid poison tanks.

#### 4.4.4 Heat Sinks

The normal heat sink for the fuel in the core is provided by forced circulation of HTS coolant, which transfers the core heat to the secondary coolant in the steam generators, with feedwater supplied to the steam generators by steam generator feedwater pumps. Following shutdown, the core decay heat can normally be removed through two alternate independent and diverse paths: through steam generators, with heat rejected by boiling off

feedwater, or through the shutdown cooling system, with heat rejected to process/service water, which ultimately rejects heat to atmosphere (through cooling towers) or to cooling water from the ultimate heat sink (river, lake, and ocean).

Under reactor shutdown conditions, cooling of the core to remove decay heat can be performed by natural circulation of the HTS coolant through the steam generator tubes (i.e., forced circulation is no longer necessary). On the secondary side, feedwater flow to the steam generators, at a substantially reduced rate (about 4% of normal feed flow), is provided by auxiliary steam generator feed pumps. The power source for these pumps depends on the station design: some are electric, using Class III (diesel generator) power; some stations use steam-driven pumps; and some use direct diesel drive. In the event of failures in the secondary side (either in the feedwater or main steam supply systems), additional safety-related systems such as the steam generator/boiler emergency coolant system or the emergency water system are available to provide separate water supply to the steam generators to maintain the heat sink. The emergency water system is usually a seismically qualified system.

For certain accident conditions, other cooling paths are also provided. During a LOCA, the ECCS is used to refill the core and remove decay heat from the reactor. High-pressure injection is supplied by a system of accumulators containing water and pressurized by nitrogen gas tanks or by high-pressure pumps in some plants. Intermediate pressure injection and long-term recovery and recirculation is provided by pumps (powered by Class III electric supply) with water drawn initially from a tank, and subsequently from a sump (which collects spilled water from the break) via the ECCS heat exchanger. The heat picked up by ECCS water is rejected to process water in ECCS heat exchangers. In current HWRs, the entire sequence is automated, whereas in some older HWRs operator action is required to switch from intermediate pressure injection to recovery mode.

ECCS is accompanied by *crash cool down* of the steam generators, involving blowing off steam to atmosphere through the main steam safety valve. This ensures that the HTS pressure stays below ECCS injection pressure, especially for small LOCA, and also for large LOCA in the long term.

The minimum design objective for the ECCS is to limit the release of fission products from the fuel. While specific acceptance criteria may differ from country to country, typical requirements in this regard are listed below.

- For LOCAs with break size smaller than, and up to, the largest feeder break, there shall be adequate cooling of the core to prevent gross fuel sheath failures. (However, in single channel events, failure of fuel in the affected channel may not be prevented.)
- For LOCAs larger than feeder pipe breaks, fuel failures shall be limited such that the radiological consequences to the public are within limits acceptable to the regulator for this class of event.
- For all LOCAs, the integrity of fuel channels shall be maintained, and fuel geometry shall allow continued coolability of the core by ECCS.
- Adequate long-term cooling capability of the fuel following the LOCA shall be ensured.

A unique feature of pressure tube HWRs is large volumes of heavy water or light water surrounding the fuel channels and the calandria vessel, respectively. These water volumes provide an inherent means for removal of decay heat from the core during BDBA that progresses to severe core damage. The two water sources are the heavy water moderator

surrounding the fuel channels and the light water shielding surrounding the calandria vessel. The moderator provides an effective heat sink to prevent the development of severe core damage during LOCA events with failure of ECCS. The shielding water surrounding the calandria vessel provides an inherent heat sink in the event that moderator system cooling is lost, causing moderator boil-off and subsequent disassembly of fuel channels inside the calandria vessel. Both of these inherent heat sinks are normally in place and require no operator actions under severe accident management guidelines (SAMG) to activate them. They provide important passive means that can stabilize core damage during severe accidents or significant time delay during severe accident progression to allow alternative SAMG candidate actions to be undertaken.

#### 4.4.5 Containment

The chief function of the containment system is to limit the accidental release of radioactivity to the environment to within acceptable limits. The containment system consists of a leak-tight envelope around the reactor and associated nuclear systems and includes a containment isolation system (for fast closure of valves/dampers in lines penetrating the containment), containment atmosphere energy removal (cooldown) systems, and clean-up systems. Hydrogen control is provided in the newer, larger HWRs to cater for thermochemical hydrogen generation from the zircaloy–steam oxidation reaction due to overheated fuel and from long-term hydrogen build-up due to radiolysis after a LOCA.

Current pressure-tube HWR designs employ the following major types of containment:

1. Single unit containment, with a dousing system for pressure suppression, as used in CANDU 6
2. Multiple unit containment with a common vacuum building, as used in the Pickering, Darlington, and Bruce stations
3. Double containment system used in Indian HWRs, incorporating a double envelope, and pressure suppression

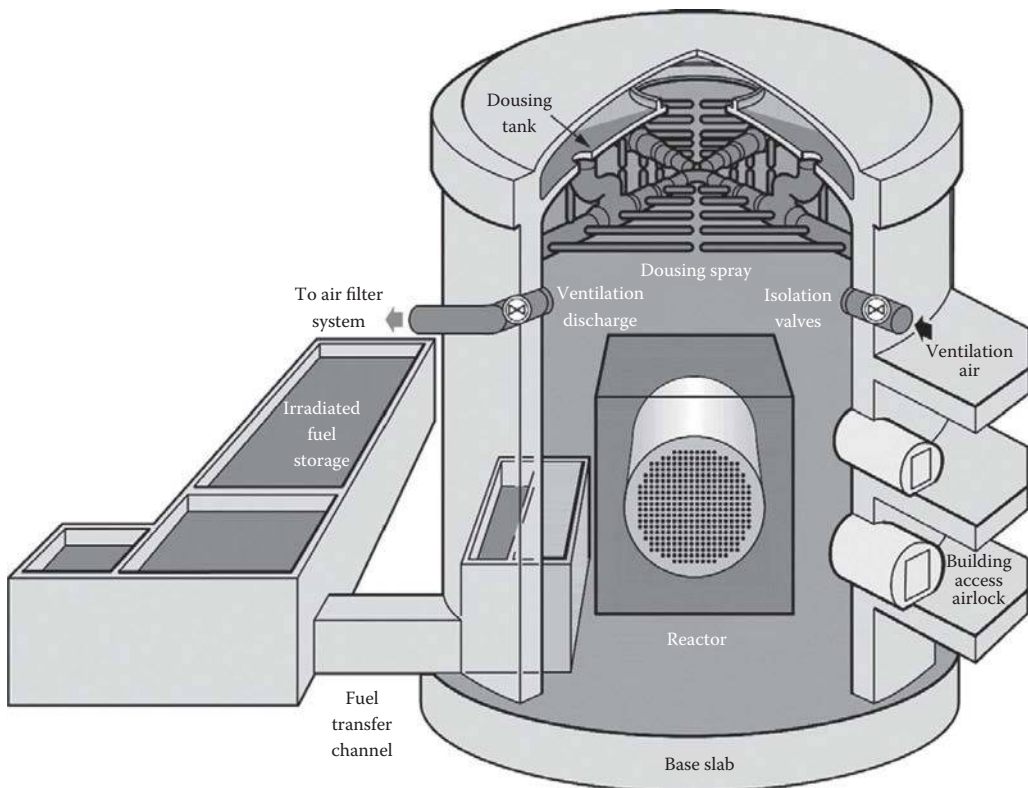
These concepts are described briefly in the following sections. Note that the HWR design does not determine the type of containment; that decision is driven by other factors such as single versus multiple unit philosophy, national regulatory requirements, allowable leak rates, and construction cost.

#### 4.4.6 Single-Unit Containment System

The CANDU 6 single-unit containment consists of a cylindrical pre-stressed, post-tensioned concrete building with a concrete dome (Figure 4.11). The concrete provides strength and shielding; the building is lined with an epoxy coating to improve leak-tightness. Beneath the outer dome there is an inner dome having an opening in the crown. The double dome together with the perimeter wall forms a container, providing storage at an elevated level for water for dousing and emergency core cooling.

On a rise in pressure or a release of radioactivity to the containment, the containment isolation system would close all penetrations open to the outside atmosphere, mainly the containment ventilation system. This is a subsystem of the containment safety system.

A sufficiently large pressure rise (e.g., from a LOCA or steam line break) would trigger the dousing spray system through valves in the dousing spray headers. The purpose of the



**FIGURE 4.11**  
Single unit containment.

dousing spray is to suppress the short-term pressure rise caused by the accident, thus the flow rate is very high. Dousing turns on when the overpressure rises to  $>14$  kPa, and turns off when it falls to  $<7$  kPa, resulting in a cyclic operation for small LOCA. Operation of redundant dousing valves maintains the pressure following a LOCA below the containment design pressure.

#### 4.4.7 Multiunit Containment System

In the multiunit vacuum system, employed in the CANDU stations in Ontario, four to eight reactors, each with its own local containment, are connected by large ducting to a separate, common vacuum building kept, as its name implies, at near zero absolute pressure. Should steam be released from a pipe break in the reactor building, the pressure causes banks of self-actuating valves connecting the vacuum building to the ducting to open. Steam and radioactivity are then sucked along the duct; the steam is condensed by dousing in the vacuum building and soluble fission products such as iodine are washed out. The dousing is passively actuated by the difference in pressure between the main body of the vacuum building and the vacuum chamber; it does not require electrical power or compressed air supplies to operate. This concept, which was developed because of the economics of multiunit sites, has several unique safety characteristics:

- After an accident, the entire containment system is sub-atmospheric for several days; thus leakage is inward, rather than outward. This is true (for less time, of course) even with an impairment in the containment envelope.
- The overpressure period in the reactor building is very short, of the order of a couple of minutes, so the design pressure is reduced and the design leak-rate can be increased relative to single unit containment.
- Even if the vacuum building is not available, the large interconnected volume of the four or eight reactor buildings provides an effective containment.
- Several days after an accident, when the vacuum is gradually depleted, and the containment pressure rises toward atmospheric, an emergency filtered air discharge system (EFADS) is used to control the pressure and ensure the leakage is filtered.

#### 4.4.8 Containment System of Indian HWRs

Current Indian HWRs use a double containment design. The annular space between the primary and secondary containment envelopes is provided with a purging arrangement to maintain a negative pressure in the space. This arrangement will significantly reduce the ground level releases to the environment during accidents where there is a radioactivity release into the primary containment.

#### 4.4.9 Safety Analysis

The basic purposes of safety analysis are to assist in the design of safety-related systems, and then to confirm that the radiation dose limits are met. As such, safety analysis requires predictions of the consequences of hypothetical accidents.

In most cases, the approach to HWR licensing has been performance-based rather than prescriptive. That is, the regulator sets overall requirements on the classes of accidents to be considered, and on the public dose limits as a function of accident class, but leaves it up to the licensee to a large extent to determine how best to meet the requirements and limits. In particular most HWR regulators do not specify the design requirements in great detail, nor do they specify prescriptive assumptions on accident analysis methods.

Safety analysis covers a wide range of initiating event failures and combinations of coincident or subsequent failures of process and safety systems. In addition, mitigating process system actions are normally not credited in demonstrating the effectiveness of the safety systems.

A unique aspect resulting from the application of the Siting Guide is that the design basis analysis has included a class of events called dual failures. A dual failure is defined as the simultaneous failure of a process system and the unavailability of a safety system or subsystem. Safety analysis is therefore performed for the failure of each process system in the plant; then for each such failure combined with the unavailability or impairment of each relevant safety system or subsystem in turn. Examples of major impairments are

- Unavailability of one of the two SDSs (always assumed)
- No emergency coolant injection
- No containment isolation
- Failure of dousing
- Deflated airlock door seals

- Failure of vault coolers
- Partial or total loss of vacuum (vacuum containments)

Dual failures in HWR are in many respects equivalent to severe accidents considered for other reactor types.

#### 4.4.10 Safety Analysis Scope

The general scope of safety analysis for HWRs, covering the accident categories considered, safety barriers challenged, and the technical disciplines involved are summarized in Table 4.2. Failures in safety support systems (such as instrument air) are addressed in the PSA.

HWR accident analysis practice has been to use physically realistic models of the system behavior, with conservatism incorporated in assumptions on input parameters and operator response. This requires relatively detailed models in the technical disciplines identified above.

#### 4.4.11 LOCA

The LOCA imposes the most severe challenge to all three safety systems (shutdown, emergency core cooling, and containment) and sets many of their design requirements (Luxat 2003). The LOCA are categorized according to the magnitude of the pipe rupture and the resultant process systems response.

**TABLE 4.2**

Scope of HWR Safety Analysis

Accident Category	Barriers Challenged	Technical Discipline
Loss of regulation	Fuel sheath	Reactor physics
	HT system boundary	System thermal hydraulics
Loss of reactivity control	Fuel sheath	Reactor physics
	HT system boundary	System thermal hydraulics
Loss of HT flow	Fuel sheath	System thermal hydraulics
	HT system boundary	Reactor physics
Loss of HT coolant (small LOCA and single channel events)	Fuel sheath	Reactor physics
	Containment	System thermal hydraulics
		Containment thermal hydraulics
		Fuel and fuel channel thermal–mechanical behavior
		Fission product release and transport
Loss of HT coolant (large LOCA)	Fuel pellet	Reactor physics
	Fuel sheath	System thermal hydraulics
	Containment	Moderator thermal hydraulics
		Containment thermal hydraulics
		Fuel and fuel channel thermal–mechanical behavior
Feedwater system failure	Fuel sheath	System thermal hydraulics
	HT system boundary	Containment thermal hydraulics
Steam supply system failure	Fuel sheath	System thermal hydraulics
	HT system boundary	Containment thermal hydraulics
		Fission product release and transport

A very small LOCA (or leak) is defined as having a break discharge flow rate that can be handled by the heavy water makeup system without the need for any safety system intervention. A small break LOCA is defined as a pipe break that cannot be compensated by the heavy water makeup system and extends multiple feeder pipe ruptures such that the reactor regulating system, without credit for stepback action, is capable of limiting any power excursion. A large break LOCA is defined as a pipe break beyond the range of breaks in multiple feeder pipes which give rise to uncompensated coolant void reactivity and a resultant power excursion.

For large and small LOCAs, one of the SDSs, the emergency core cooling and containment are all required and are initiated automatically or, at the lower end of the range, by the operator. For very small LOCAs, shutdown may be manual or automatic; emergency core cooling and containment are not required, and might not be initiated automatically.

#### **4.4.11.1 Small Break LOCA**

Because of the total length of feeders and pressure tubes, a small break is about 100 times more probable than a large break. Clearly this range is analyzed for both economic and safety considerations.

The requirement for ECCS for small breaks is to limit or prevent fuel damage, mainly for economic reasons. For breaks up to an equivalent area of the severance of several feeder pipes, the pumps are much more influential in determining channel flow than the break. Thus, the flow is always forward or recirculating, although as steam quality builds up with time, the pump head decreases and the resistance of the circuit increases; so the magnitude of the flow falls with time.

The first requirement is to shut down the reactor before fuel sheaths experience prolonged dryout at high power. Prevention of dryout is sufficient but not necessary to prevent fuel sheath failure. Shutdown occurs on process parameter signals such as low-pressurizer level, low-storage tank level, low-flow, low-pressure, low-core pressure drop, or high-pressure within containment. For a feeder-size break, shutdown is initiated within the first 3 or 4 min. As the circuit continues to empty after shutdown, the flow in the headers or channels eventually falls low enough that the coolant phases separate. This could result in steam cooling of some of the upper fuel elements in a channel or of some of the channels connected to the mid-plane of the header. Prolonged stratification can lead to sheath damage in the order of a few minutes, so this defines the time at which ECC must become effective. Once refill has occurred, the pumps maintain recirculating flow, and this pattern continues into the long-term until the pumps are tripped. Pump trip has been automated on certain HWRs in the longer term, after ECC refill, to avoid pump cavitation once the circuit has refilled with cold water.

The break is a major heat sink for decay heat. Breaks greater than the cross-sectional area of a feeder pipe can remove all the decay heat from one HTS loop. The steam generators, however, if not cooled, can hold-up HTS pressure; thus, the steam generators must be cooled down fast enough to ensure that ECC injection can proceed. Crash cool down takes the steam generators from normal operating pressure to close to atmospheric in about 15 min, and this allows continued ECC makeup flow.

Special cases of a small break LOCA are those involving failures in single fuel channels. Such events include pressure tube ruptures, feeder stagnation break, channel flow blockage, in-core LOCA involving failure of pressure tube and calandria tube, and failures of fuel channel end fittings. The in-core LOCA introduces additional phenomena such as interaction of the broken channel with neighboring channels and the

reactivity mechanism guide tubes. All of these failures can damage fuel: the first by mechanical damage following the pressure tube rupture; the second and third by overheating due to reduced flow; the fourth by mechanical damage following the channel rupture and ejection of the fuel into the calandria vessel; and the last by ejection of the fuel into the calandria vault, followed by mechanical damage and oxidation in the vault atmosphere.

A severe flow blockage >90% of the channel flow area is required to cause pressure tube failure due to overheating. A single channel event leading to channel failure also requires analysis of the pressure transient within the calandria, to show that the calandria vessel itself remains intact, that the SDS devices within it can still perform their function, and the break does not propagate by causing failures of other reactor channels.

For a steam generator tube failure (categorized as a leak) it is first necessary to identify the failure by detecting the leakage using a D<sub>2</sub>O-in-light water detection system. Because a single steam generator tube failure is within the capability of the D<sub>2</sub>O makeup system, fuel cooling is not at risk. The operator will shut down the reactor and depressurize the HTS, thereby stopping the leak to the secondary side. The HTS can then be drained to below the level of access to the lower steam generator head so that the tube can be isolated (plugged). Cool down and fuel cooling is achieved by using the shutdown cooling system. The shutdown cooling system can remove decay heat at full system pressure, so the operator is not dependent on the secondary side for depressurization/cool down.

#### 4.4.11.2 Large Break LOCA

The range of HTS behavior is encompassed by large pipe ruptures at three locations: at the core inlet (inlet header), at the core outlet (outlet header), and upstream of a main heat transport pump (pump suction pipe). Breaks at these locations affect the two core passes of a loop in different ways. The core pass upstream of the break (upstream and downstream directions are defined relative to the normal flow direction) always has flow that is accelerated toward the break. The fuel cooling tends to be increased and emergency core coolant refill is rapid in the flow direction toward the break. The core pass downstream of the break has its flow reduced by the flow that is diverted out of the break and is therefore more likely to experience degraded fuel cooling.

A double-ended guillotine rupture of an inlet header reverses the flow in the downstream core pass. On the other hand, a small break at the inlet header will maintain the flow in the normal flow direction. Therefore, it is possible to select a break size that leads to a period of sustained very low-flow in the downstream core pass. This low-flow arises from a balance between the break flow and the flow delivered by the upstream pump. Such breaks, referred to as *critical breaks*, tend to more limiting with respect to cooling of the fuel and fuel channels than other large breaks and are therefore analyzed in detail. After a short period of about  $\leq 30$  s of very low-channel flows in the affected pass, voiding at the pump suction degrades the pump head causing channel flows in the downstream pass to reverse toward the break.

Flows in the long-term are determined by the balance between the break and the pumps (which may be tripped at some point in the accident). At the lower end of the large break spectrum, ECCS refill flows and long-term flows will be in the forward direction. If the break is larger, ECCS refill will be in the reverse direction and this flow direction will persist into the long-term. Flow for intermediate breaks may reverse when the pumps are tripped.



Breaks at the outlet end of the core (outlet header break) will cause increased flow in the upstream pass and reduced flow in the downstream pass. Large outlet header breaks may be able to reverse the flow in the downstream core pass. For the largest outlet header break the voiding of the downstream core pass is slower than for inlet breaks because the path from the break to the core is longer and the resistance is higher. Thus, when sustained low-flow does occur there is less stored heat in the fuel. Fuel temperature increases during sustained low-flow are lower than for the inlet header case; however, ROH breaks are limiting for sheath strain failures because the sheath temperature is high when the coolant pressure is low. Smaller outlet header breaks allow continued forward flow: ECCS refill is in the forward direction and a long-term recirculating flow pattern will occur. At first the flow goes backward through the downstream pump but as the circuit depressurizes the pump acts more like a check valve. Injection water into the inlet header of the downstream core pass is therefore prevented from going through the pump to the break and instead is forced in the forward direction through the core pass. Refill of both core passes is in the forward direction. The long-term flow pattern for the largest reactor outlet header break is recirculating until the pumps are tripped; then flows are directed in each pass toward the break. Pump suction breaks are hydraulically similar to reactor outlet header breaks.

#### 4.4.11.3 Analysis Methods

As shown in Table 4.2, large break LOCA events involve the most physical phenomena and, therefore, require the most extensive analysis methods and tools. Typically, 3D reactor space-time kinetics physics calculation of the power transient is coupled with a system thermal hydraulics code to predict the response of the heat transport circuit, individual channel thermal-hydraulic behavior, and the transient power distribution in the fuel. Detailed analysis of fuel channel behavior is required to characterize fuel heat-up, thermochemical heat generation and hydrogen production, and possible pressure tube deformation by thermal creep strain mechanisms. Pressure tubes can deform into contact with the calandria tubes, in which case the heat transfer from the outside of the calandria tube is of interest. This analysis requires a calculation of moderator circulation and local temperatures, which are obtained from computational fluid dynamics (CFD) codes. A further level of analysis detail provides estimates of fuel sheath temperatures, fuel failures, and fission product releases. These are inputs to containment, thermal-hydraulic, and related fission product transport calculations to determine how much activity leaks outside containment. Finally, the dispersion and dilution of this material before it reaches the public is evaluated by an atmospheric dispersion/public dose calculation. The public dose is the end point of the calculation.

Traditionally, a *conservative* approach to safety analysis has been employed. In this approach, pessimistic assumptions, bounding input data and even conservative physical models are used to obtain a pessimistic bounding analysis. This approach has been required to a greater or lesser extent by most regulators and for most reactor types. HWRs also followed this approach for licensing analysis, although the physical models used were realistic.

The advantage of the approach was that the answer was known to be pessimistic; and in some cases the safety analysis could be simplified by using bounding rather than realistic assumptions (as long as the results were still acceptable). There are a number of disadvantages to the conservative analysis, namely

- The margin between the expected behavior and the conservative predicted behavior is unknown (how *conservative* the answer is).
- There is a tendency over time for more and more conservatisms to be added in, in an unsystematic fashion.
- As the conservative prediction gets close to the regulatory acceptance limit, regulators become uncomfortable at the apparent lack of margin, despite the conservatisms.
- The predictions of the computer codes can yield physical conditions in areas where validation is impossible (e.g., very high fuel temperatures).

Recently, however, limited use of *best estimate plus uncertainty analysis* methods has been undertaken. This is consistent with the international trend toward use of such methods. In this approach, more physically realistic models, assumptions, and plant data are used to yield analysis predictions that are more representative of expected behavior. This requires a corresponding detailed analysis of the uncertainties in the analysis and their effect on the calculated consequences. Typically, the probability of meeting a specific numerical safety criterion, such as a fuel centerline temperature limit, is evaluated together with the confidence limit that results from the uncertainty distributions associated with governing analysis parameters. The *best estimate plus uncertainties* approach addresses many of the problematic issues associated with conservative bounding analysis by

- Quantifying the margin to acceptance criteria
- Allowing rational combination of uncertainties
- Evaluating *cliff-edge* effects
- Highlighting parameters that are important to safety
- Focusing safety Research and Development on areas of true importance
- Providing the basis for more realistic compliance monitoring

#### 4.4.12 Severe Accidents

Severe accidents may be defined simply as accidents for which heat removal from the reactor fuel is insufficient to remove the heat generated for a sufficiently prolonged period of time such that damage occurs to fuel or structures within the reactor core. Despite the existence of engineered safety systems, the possibility exists (albeit at a very low level of likelihood) that an accident can progress beyond the acceptable design basis envelope and develop into a severe accident. The severity of such accidents can be characterized by the nature and extent of core damage that occurs during the progression of an accident.

The progression of events to an accident with severe fuel or core damage in an HWR involves several broad stages in which thermal-hydraulic behavior of the reactor fuel, fuel channels, HTS, and a number of key process systems govern the rate at which severely degraded cooling conditions develop and the extent of resultant damage to the reactor core (Blahnik et al. 1993; Luxat 2007; Rogers et al. 1995). As indicated previously, the moderator can provide an inherent passive heat sink in the event of a LOCA with failure of the ECCS. In this case, if moderator cooling can be assured then the reactor damage state is limited to severe fuel damage with fuel channels remaining intact. Should moderator system cooling be lost and moderator boil-off occur, then the fuel channels will disassemble in the calandria vessel forming a debris bed, including the formation of a molten corium

pool within a solid debris crust. If shield cooling water can be maintained on the outside of the calandria vessel such that adequate heat transfer from the vessel wall can be assured, then the core debris can be retained in the vessel (in-vessel retention) and further progression of the accident is terminated (Luxat 2007; Muzumdar et al. 1998). However, if in-vessel retention cannot be assured (Luxat and Luxat 2007), then ex-vessel debris coolability issues and the adequacy of containment heat sinks have to be addressed to demonstrate that containment integrity is not impaired.

The unique inherent and passive heat sink design features of HWRs result in severe accidents that are expected to progress with ample opportunity for operator actions to stabilize the plant and mitigate the consequences.

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## 4.5 Beyond the Next Generation CANDU: CANDU X Concepts

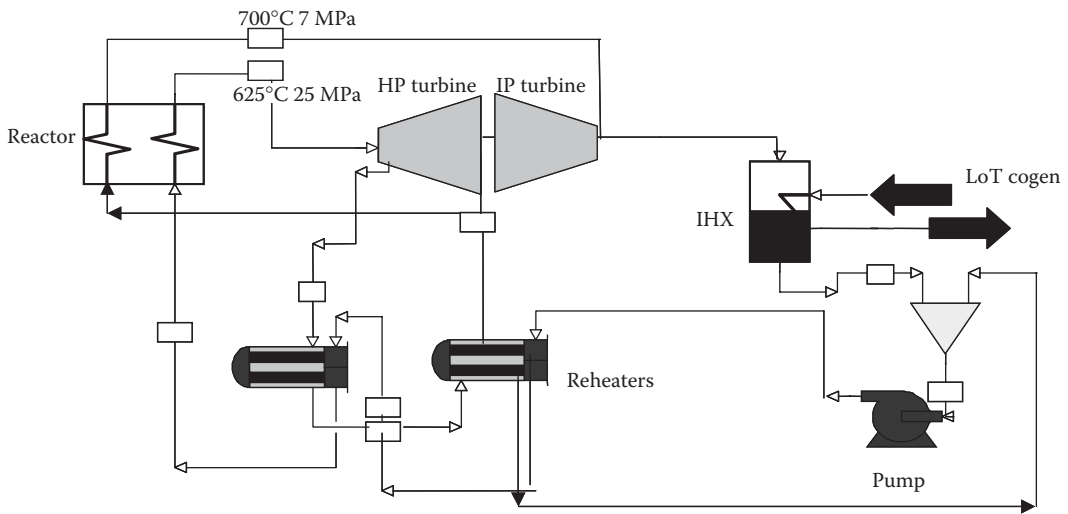
### 4.5.1 Introduction

In AECL's view, the next generation of pressure-tube HWRs encompasses an evolutionary set of technologies that lead ultimately to the *CANDU X*. The *CANDU X* is not a specific reactor design but embodies concepts extrapolated from the current knowledge base that we believe can be achieved in our development programs over the next 25 years. Therefore, *CANDU X* is a changing set of technologies and targets, with the overall goal of a further 50% cost reduction beyond the best current technology at any point in time (Spinks et al. 2002).

One element of AECL's development program is to continue to improve plant economics by increasing thermal efficiencies. This will require development of materials and systems that can withstand the higher temperatures and pressures that this improvement entails. The first step in this long-term evolution is a reactor utilizing supercritical water (SCW) as the HTS coolant. This will require HTS components that operate at about 430°C and 25 MPa and will boost the thermal efficiency to about 40%. The ultimate goal is to improve the efficiency to about  $\geq 50\%$  by reheating the coolant to 625°C at 5–7 MPa and operating the HTS under mixed supercritical and subcritical channel conditions. Such a system could make use of existing small direct cycle turbines with single reheat, currently used in thermal power plants, located inside the containment building to generate electric power, and a steam generator with an external turbine/generator outside containment to produce additional power (Figure 4.12).

Pressure-tube reactors like CANDU are very amenable to using SCW. SCW coolant cycles result in substantial coolant density variations (particularly if the water temperature crosses the critical temperature in the core). There can be a density change (by as much as a factor of 7) through the core, complicating flux gradients and flux shaping requirements. These complications are less important in pressure-tube type reactors for two reasons. First, because the moderator is located in the calandria vessel and is separated from the coolant, the coolant has relatively less effect on the neutronics. Second, depending on orientation, the channel flows can be bidirectionally interlaced (opposite flow direction in adjacent channels) or can use re-entrant flow paths, so the density gradients are balanced and a more axially uniform flux profile is achievable.

Another major reason why pressure-tube reactors are suitable for SCW coolant is their ability to adapt the pressure boundary to accommodate much higher pressures. At the



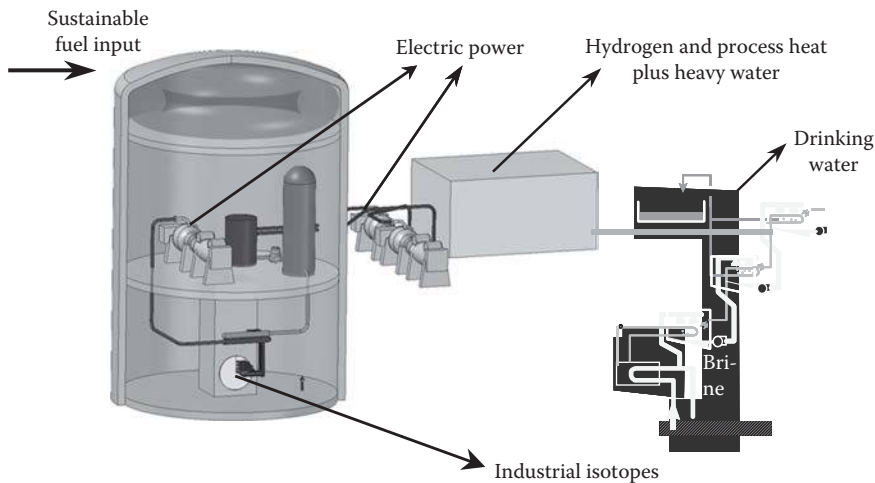
**FIGURE 4.12**  
CANDU CWR (schematic).

25-MPa pressures required for SCW coolant, there would be challenging requirements for developing a large-pressure vessel. For pressure-tube reactors, it will be far easier to meet the requirements by evolving the design of the fuel channel. A new, insulated fuel channel concept is being developed in which the high-pressure boundary is kept at a relatively low temperature where the material strength is higher. This fuel channel concept will experience lower creep strains during aging (even with SCW coolant) than the current fuel channel design. Pressure-retaining components can be tested directly at full-scale, which greatly facilitates development of the SCW coolant technology for the CANDU system.

Heat transfer under supercritical conditions has already been widely investigated (Piroo et al. 2002) and provides a literature survey. The IAEA is coordinating an international cooperative research program on heat transfer behavior and thermohydraulic code testing for SCWRs, which is intended to lead to issue of a TECDOC report.

Figure 4.13 shows the CANDU X concept co-producing hydrogen, process heat, and distilled water alongside electricity. Hydrogen can be produced by conventional low-temperature electrolysis, but thermochemical and high-temperature electrolytic processes are being developed. If these high temperature approaches prove to be economically superior, heat could be provided directly from the SCWR or, if higher temperatures are needed, by the use of re-entrant superheater tubes located in the reactor's periphery. Distilled water—either using heat to enhance reverse osmosis or for a distillation-based approach—requires very modest temperatures. This application is expected to become more widespread with rising pressures on water supply in many parts of the world and can easily be applied with any reactor type. However, HWRs offer the novel possibility of using the moderator—in which about 5% of reactor heat is deposited—as the heat source.

Generation of electricity with supercritical steam is existing technology from fossil-fired plants. The distinctive challenges in developing the CANDU SCWR concern the materials that will be exposed to the high temperatures of the supercritical steam and the intense neutron flux in the reactor's core. With the pressure tubes shielded by the insulator from high temperature, materials robust to the effects of very high-temperature and neutron flux are only required for the fuel cladding and the insulator.



**FIGURE 4.13**  
CANDU X concept co-producing hydrogen, process heat, and distilled water alongside electricity.

Several variants of SCWR concepts have been proposed and are reviewed in Duffey and Piro (2006), and the major pressure tube variants are listed in Table 4.3 from Canada and Russia.

#### 4.5.2 Passive Safety: Eliminating Core Melt

A key element of future reactor designs is improved safety. The unique channel layout for pressure-tube designs uses the moderator as a backup heat sink for emergency heat removal and provides a *walk away* safety argument that requires no active systems to activate or be operated (Vasic and Khartabil 2005). By testing passive moderator heat rejection systems, the most promising design is a passive, flashing-driven moderator cooling system. One of the main objectives of CANDU SCWR is to optimize the advanced fuel channel design to ensure that the passive moderator-cooling loop can remove reactor decay heat in the unlikely event of a LOCA combined with LOECC, that is, normal cooling is lost and emergency heat removal systems do not activate.

Using the results of code simulations, it is possible to optimize the performance of the insulated fuel channel under decay heat generation conditions and variable thermal-physical characteristics of the insulating layer. This study shows that the advanced channel design is promising and can prevent overheating of the fuel even during very severe accident scenarios. The results show that advanced fuel channel design combined with the passive moderator heat rejection and selection of the fuel-clad design have the potential to avoid core melting and to reduce significantly or even eliminate the possibility of core damage.

##### 4.5.2.1 Beyond Uranium

Development of reactor technology to allow burning of a range of fissile fuels is another important avenue of HWR development.

TABLE 4.3

Typical Pressure Tube SCWR Parameters

Parameters	Unit	SCWR CANDU	ChUWR	ChUWFR	KP-SKD
Country	–	Canada	Russia	Russia	Russia
Organization	–	AECL		RDIPE (НИКИЭТ)	
Reactor type	–	PT	PT	PT	PT
Spectrum	–	Thermal	Thermal	Fast	Thermal
Power thermal	MW	2540	2730	2800	1960
Electrical	MW	1140	1200	1200	850
Linear max/ave	kW/m		38/27		69/34.5
Thermal efficiency	%	45	44	43 (48)	42
Pressure	MPa	25	24.5	25	25
$T_{in}$ coolant	°C	350	270	400	270
$T_{out}$ coolant	°C	625	545	550	545
Flow rate	kg/s	1320	1020		922
Core height	m		6	3.5	5
Diameter	m	~4	11.8	11.4	6.45
Fuel	–	UO <sub>2</sub> /Th	UC	MOX	UO <sub>2</sub>
Enrichment	% wt.	4	4.4		6
Cladding material	–	Ni alloy	Stainless steel	Stainless steel	Stainless steel
# of fuel assemblies		300	1693	1585	653
# of rods/FA		43	10	18	18
$D_{rod}/\delta_w$	mm/mm	11.5 and	12/1	12.8	10/1
Pitch	mm	13.5			
$T_{max}$ cladding	°C	<850	630	650	700
Moderator	–	D <sub>2</sub> O	Graphite		D <sub>2</sub> O

Source: Duffey, R. et al., Supercritical water-cooled pressure channel nuclear reactors: Review and status, *Proceedings of GLOBAL*, Paper No. 020, Tsukuba, Japan, October 9–13, 2005. With permission.

A huge, long-term economic opportunity is opening up in the nuclear energy field with growing appreciation that nuclear fission is likely essential to any global program to stabilize atmospheric levels of greenhouse gases (GHGs). Alternative technologies—wind, solar, tidal, geothermal, and hydrocarbon combustion with CO<sub>2</sub> sequestration—all produce electricity more expensively than nuclear. Some are intrinsically intermittent; some are still in early development. In this situation and because the need to curtail GHG emissions is urgent, extensive nuclear deployment seems essential. Objective assessments suggest that nuclear deployment of 5,000–10,000 new reactors—10–20-times the present number—will be needed to stabilize GHG concentrations by ~2050. This is in addition to the envisaged massive investments in renewable technologies such as wind and solar and to extensive conservation and efficiency measures (Miller et al. 2006). Deployment on this scale will be difficult to sustain if it is based on today's technology, which is entirely dependent on uranium and predominantly uses once-through fuel cycles, cycles that typically extract only 0.6%–0.7% of the energy available in the uranium resource. Higher reactor temperatures can help, but the issue is eliminated only if the industry develops fuel recycle and/or utilizes thorium.

Worldwide, there is renewed interest in nuclear power as a result of concerns not only about climate change but also air pollution, energy security, and the cost and availability

of fossil fuels. Nuclear power program decisions will be increasingly based on political, strategic, and economic considerations involving the complete nuclear fuel cycle, including resource utilization, radioactive waste disposal, proliferation resistance, and supply assurances. The global nuclear industry needs to address each of these issues. The industry's long-term growth and its capacity for substantial GHG abatement will depend on following a path that addresses these issues by developing advanced fuel cycles and reactor designs optimized for such fuel cycles. The overall direction of the global power reactor development program should be refocused to provide a greater emphasis on integrated and complementary reactor and fuel cycle development, including the development and deployment of fuel enrichment and reprocessing technology and services.

Technical and policy developments in the field of energy that are linked and overlapping are detailed below.

- In the global energy context, the inexorably growing demand, increasing costs of alternative energy sources, and concerns about security of energy supply and environmental emissions of carbon dioxide and other GHGs are all driving the need for more extensive deployment of nuclear energy worldwide. In March 2016, the World Nuclear Association reported that 440 reactors with a total capacity of 384 GW were supplying 11.5% of the world's electricity; 65 (69 GW) were under construction; 173 (182 GW) were planned; and a further 337 (379 GW) were proposed. These large increases include several countries announcing plans to build and deploy nuclear reactors for the first time (e.g., Turkey, Egypt, and Chile). Deployments are occurring for several reasons, including GHG abatement and enhanced security of energy supply.
- In the business context, the international trends to more effective uranium utilization, closed fuel cycles with reprocessing and recycle of spent fuel, and more effective and efficient management of spent fuel and reduction of eventual wastes are becoming obvious. These trends require major exporters of nuclear reactors and uranium fuel with international commitments to develop an effective international presence and new technical processes to keep technology relevant and competitive (e.g., as evidenced by the Global Nuclear Energy Partnership efforts of the United States).
- In the strategic economic context, it is necessary to develop a longer-term view that will synergistically benefit the economy, the global environment, and the furtherance of interests at large. Such a view will provide a sound technological and scientific framework for the future and must also address the collateral issues such as nuclear nonproliferation more effectively and realistically (e.g., the Iran, Iraq, Israel, Pakistan, India, and North Korea positions).
- Technically, this will open up an important opportunity to transform *waste* disposal into *fuel* management. By recycling fissile and fertile material and burning actinides, the true waste that remains—fission fragments—represents a few percent or less of the fuel discharged from current reactor types. These fission fragments will decay almost entirely to stable isotopes within a few hundreds of years, and their secure disposal should be a far less emotive source of objection than has been the case with unprocessed spent fuel, which retains significant levels of activity for as much as 250,000 years, albeit at levels comparable with uranium ore.

### 4.5.3 Two Views of Fuel Cycles: DFC or AFC

Based on the natural evolution of using enriched fuel from weapon-based  $^{235}\text{U}$  enrichment technology, present thermal reactors use uranium as the main fuel supply with some recycling of Pu mixed-oxide (MOX) fuel. The cycle is essentially a once-through system, with fuel irradiated to about 40,000 MWd/t, and then stored until cooled and ready for Pu separation, or kept in interim storage buildings (e.g., Zwiileg Facility in Switzerland) until ready for sending to the underground repositories planned in many countries. As an order of magnitude, an operating 1-GW(e) LWR today requires mining of 180 t/a of uranium (House of Representatives Standing Committee 2006). (To produce uranium enriched to slightly over 3%  $^{235}\text{U}$ , five times as much depleted uranium at 0.24%  $^{235}\text{U}$  is produced as depleted uranium.)

So with >384 GW in operation today (predominantly supplied from conventional uranium mines) present world demand is ~70,000 t/a. We can provide an upper bound estimate of demand for 5000 GW of new reactors needing ~one million t/a by 2050. Today's estimates of proven uranium reserves at a cost of <\$130/kg is about six million tons (IAEA and OECD-NEA 2005). Even allowing that exploration will likely lead to a doubling or tripling of the resource estimate to, say, 20 MtU, just 2000 reactors operating for 60 years would use all the world's cheapest uranium with present fuel cycles technology.

Just the present and planned 650 reactors could be kept going for another 150 years, but that falls far short of the scope for reactor deployment. This is not a cause for alarm, there is plenty of uranium, and more uranium reserves will be found but at higher prices. Moreover, aggressively adopting recycling and increased fuel utilization with existing reactor types might allow up to 1500 reactors.

So there emerges at least two views of fuel cycles, which we may summarize as shown below.

1. *Traditional demonstrated fuel cycle (DFC) view.* For those already with access to or reserves of uranium, such as the United States, France, and Canada, the uranium fuel cycle is an already demonstrated fuel cycle (DFC) and is fine while uranium is cheap and assuredly available.

There is always more uranium to find, even though the cycle is known to be unsustainable (as per the above calculations) because most current reactors (LWRs and HWRs) are very inefficient fuel users.

In DFC's unhurried view, *when* uranium becomes too scarce and/or expensive, one can switch to technologies able to breed more fissile material than they consume. This could employ fast reactors, accelerator-driven breeders, and/or Pu recycle, even if it is more expensive and requires a different reactor technology. Given its greater complexity and higher costs, transition to one or more of these advanced technologies is still decades away. In the meantime, existing thermal reactors will continue to discharge spent fuel, which can be stored retrievably and considered more as a future resource for recycling than waste.

2. *New alternate fuel cycle (AFC) view.* Without introduction of radically different reactor types, countries without access to large uranium reserves and needing energy supply surety can initiate a turn much sooner to a new AFC that will ensure sustainable supply and smaller waste streams. Preferably, this should be a more intrinsically nonproliferation cycle, with no significant Pu generation, thus not requiring all of today's policing and international stress. As well as not



requiring the introduction of a new reactor technology, this perspective should acknowledge the constraints on U-enrichment ownership and deployment as a proliferation concern while still allowing vastly expanded reactor builds.

Burning thorium rather than uranium is the most obvious AFC. Thorium is about three times more plentiful globally than uranium and with careful fuel design and recycling, an HWR gives a near breeding cycle. So, it is more sustainable with much lower waste amounts and storage needs (as little as 10%). Switching to thorium would enable more reactor deployment using today's reactor technologies and help stabilize fuel cost and supply. This approach avoids having to introduce many fast reactors. In the long term, full benefit from thorium requires extracting and recycling the  $^{233}\text{U}$ . LWRs cannot achieve a near-self-sufficient thorium cycle.

This AFC opportunity is real and potentially could totally alter the global fuel cycle and the reactor deployment opportunities, and India (Kukodkar and Simha 2006) has already chosen to develop it as a national priority. Such AC concepts are in fact not new; what is new is the concept that an alternate *sustainable and closable* fuel cycle may enable greater benefits from nuclear energy deployment worldwide.

#### **4.5.3.1 Link between Fuel Natural Resources and Reactor Technology**

There is a link between the choice of fuel cycle and the optimal reactor, qualified by noting that we would not be pursuing a *perfect* cycle but a practical and at the same time economic one. The conventional answer has always been that we need to move to a *breeding* fast reactor, with a higher energy neutron spectrum that produces more plutonium, from neutron capture in  $^{238}\text{U}$ , than it burns to produce power, and hence can be used to generate its own fuel. The fast-spectrum reactor is quite a complicated technology, and more expensive generally, and is one commonly adopted optimization solution. But there are others.

We must *optimize nuclear technology*: reduce the volume of waste drastically by re-use/burn of discharged fuel, and ensure sustainability and maximization of resource utilization. This means that the core design, fuel design, and safety must be harmonized with the waste streams and fuel supply.

In addition, we must *agree on definitions of sustainable and closable global fuel cycle*, recognizing the inexorable international pressure to restrict the number of countries who would be *allowed* to enrich and reprocess fuel.

This means we must examine fuel cycles that use different processing, different separation systems for isotopes, and also place different and smaller demands on enrichment facilities.

The response to these challenges involves some new thinking, new R&D, and a new fuel development strategy using alternate cycles that include recycling *and* thorium fuels.

#### **4.5.3.2 Global Realities and Directions: Emphasis on Nuclear Energy and Fuel Cycles**

An increasing emphasis on global energy supplies is inevitable. World nuclear use will grow as energy demand, economic needs, environment issues, and supply security concerns grow. As part of the effort to address concerns over potential climate change, massive switching to noncarbon sources is needed and should be anticipated.

LWRs will provide much of the anticipated expansion, but the co-existence of HWRs adds flexibility and the capacity to extend the fuel resource. In the short-term, although expectations of nuclear expansion led to a near 3-fold price increase for uranium after 2002, this has stimulated a wave of exploration that is already defining large additions to the known reserve. However, exploration will not address the supply issue indefinitely. Deeper into the expanded deployment of nuclear power, extracting more of the available energy in uranium and adding thorium will become important and the flexibility and superior efficiency inherent in HWRs will become increasingly attractive, ultimately providing the best gateway into utilizing the world's large thorium resource.

---

## Glossary

<b>AFC</b>	Alternate Fuel Cycle
<b>BLW</b>	Boiling Light Water
<b>CANDU</b>	CANada Deuterium Uranium
<b>CANFLEX</b>	CANDU FLEXible fueling—an advanced fuel bundle design
<b>CANLUB</b>	Graphite-lubricated CANDU fuel
<b>CNSC</b>	Canadian Nuclear Safety Commission
<b>DFC</b>	Demonstrated Fuel Cycle
<b>DUPIC</b>	Direct Use of Spent PWR Fuel in CANDU
<b>ECCS</b>	Emergency Core Cooling System
<b>HTS</b>	Heat Transport System
<b>HWM</b>	Heavy Water Moderated
<b>HWR</b>	Heavy Water Reactor
<b>IAEA</b>	International Atomic Energy Agency
<b>LOCA</b>	Loss of Coolant Accident
<b>LOECC</b>	Loss of Emergency Core Cooling
<b>LVRF</b>	Low Void-Reactivity Fuel
<b>MOX</b>	Mixed (uranium and plutonium) Oxide fuel
<b>MWe</b>	Megawatts electrical power
<b>MWth</b>	Megawatts thermal power
<b>NPD</b>	Nuclear Power Demonstration reactor
<b>NRU</b>	Nuclear Reactor Universal (an AECL test reactor located at Chalk River, ON)
<b>OREOX</b>	Oxidation and REDuction of Oxide fuels (reprocess technology for spent reactor fuel)
<b>OTT</b>	Once-Through Thorium fueling cycle
<b>pD</b>	Hydrogen-ion concentration in heavy water, analogous to pH in ordinary water
<b>PHWR</b>	Pressurized Heavy Water Reactor
<b>PIE</b>	Post-Irradiation Examination
<b>RU</b>	Recycled Uranium fuel
<b>SEU</b>	Slightly Enriched Uranium (enriched in <sup>235</sup> U)
<b>SGMB</b>	Sol-Gel Microsphere Pelletization
<b>SSET</b>	Self-Sufficient Equilibrium Thorium Cycle

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# 5

## *High-Temperature Gas-Cooled Thermal Reactors*

Chris Ellis and Arkal Shenoy

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### 5.1 History of High-Temperature Gas Cooled Reactors

The high-temperature gas-cooled reactors (HTGRs) concept evolved from early air-cooled and CO<sub>2</sub>-cooled reactors. The use of helium in lieu of air or CO<sub>2</sub> as the coolant, in combination with a graphite moderator, offered enhanced neutronic and thermal efficiencies. The combination of helium cooling and graphite moderator makes possible production of high-temperature nuclear heat, and hence the name HTGR.

To date, seven HTGR plants have been built and operated (Table 5.1). The first was the 20 MW(t) Dragon test reactor in the United Kingdom. Dragon was followed by construction of two relatively low power plants, the 115 MW(t) Peach Bottom I (PB-1) in the United States and the 49 MW(t) Arbeitsgemeinschaft Versuchsreaktor (AVR) in Germany. PB-1 and AVR demonstrated electricity generation from HTGR nuclear heat using the Rankine (steam) cycle. These two plants were followed by the construction of two mid-size steam cycle plants, the 842 MW(t) Fort St. Vrain (FSV) plant in the United States and the 750 MW(t) thorium high-temperature nuclear reactor (THTR) plant in Germany. In addition to demonstrating the use of helium coolant (with outlet temperatures as high as 950°C) and graphite moderator, these early plants also demonstrated coated particle fuel, a fuel form that employs ceramic coatings for containment of fission products at high temperature, which is

**TABLE 5.1**  
 HTGR Plants Constructed and Operated

Feature	Dragon	Peach Bottom	AVR	Fort St. Vrain	THTR	HTTR	HTR-10
Location	UK	USA	Germany	USA	Germany	Japan	China
Power (MW(t)/Mwe)	20/-	115/40	46/15	842/330	750/300	30/-	10/-
Fuel elements	Cylindrical	Cylindrical	Spherical	Hexagonal	Spherical	Hexagonal	Spherical
He temp (In/Out°C)	350/750	377/750	270/950	400/775	270/750	395/950	300/900
He press (Bar)	20	22.5	11	48	40	40	20
PWR density (MW/m <sup>3</sup> )	14	8.3	2.3	6.3	6	2.5	2
Fuel coating	TRISO <sup>a</sup>	BISO <sup>b</sup>	BISO <sup>b</sup>	TRISO <sup>a</sup>	BISO <sup>b</sup>	TRISO <sup>a</sup>	TRISO <sup>a</sup>
Fuel kernel	Carbide	Carbide	Oxide	Carbide	Oxide	Oxide	Oxide
Fuel enrichment	LEU <sup>c</sup> /HEU <sup>d</sup>	HEU <sup>d</sup>	HEU <sup>d</sup>	HEU <sup>d</sup>	HEU <sup>d</sup>	LEU <sup>c</sup>	LEU <sup>c</sup>
Reactor vessel	Steel	Steel	Steel	PCRVE <sup>e</sup>	PCRVE <sup>e</sup>	Steel	Steel
Operation years	1965–1975	1967–1974	1968–1988	1979–1989	1985–1989	1998–	1998–

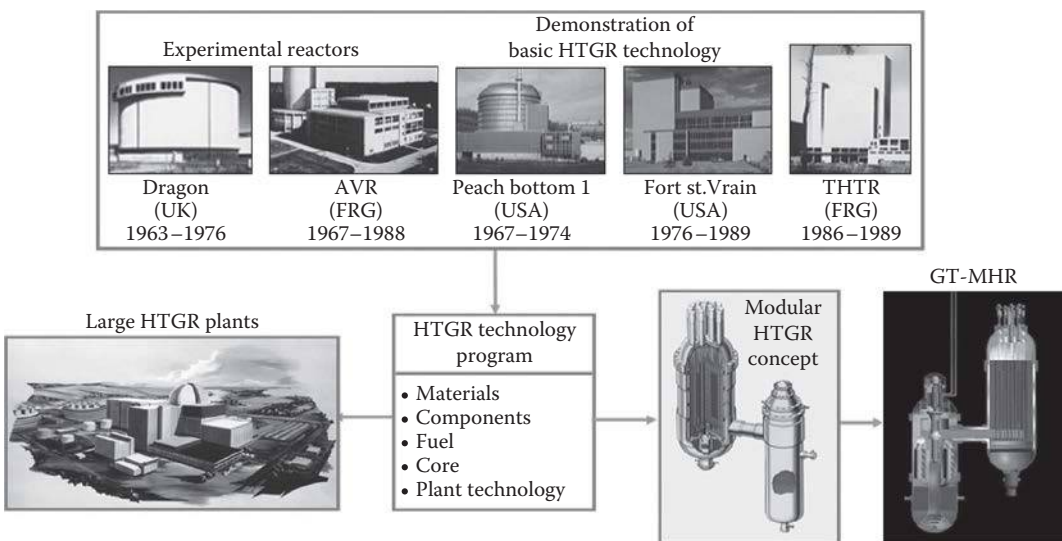
<sup>a</sup> TRISO refers to a fuel coating system that uses three types of coatings, low-density pyrolytic carbon, high-density pyrolytic carbon, and silicon carbide.  
<sup>b</sup> BISO refers to a fuel coating system that uses two types of coatings, low-density pyrolytic carbon and high-density pyrolytic carbon.  
<sup>c</sup> LEU means low-enriched uranium (<20% U<sup>235</sup>).  
<sup>d</sup> HEU means high-enriched uranium (>20% U<sup>235</sup>).  
<sup>e</sup> PCRVE means prestressed concrete reactor vessel.

a key feature of HTGRs. Figure 5.1 displays pictures of most of the HTGR plants and which elements of the HTGR technology program influenced modern plants.

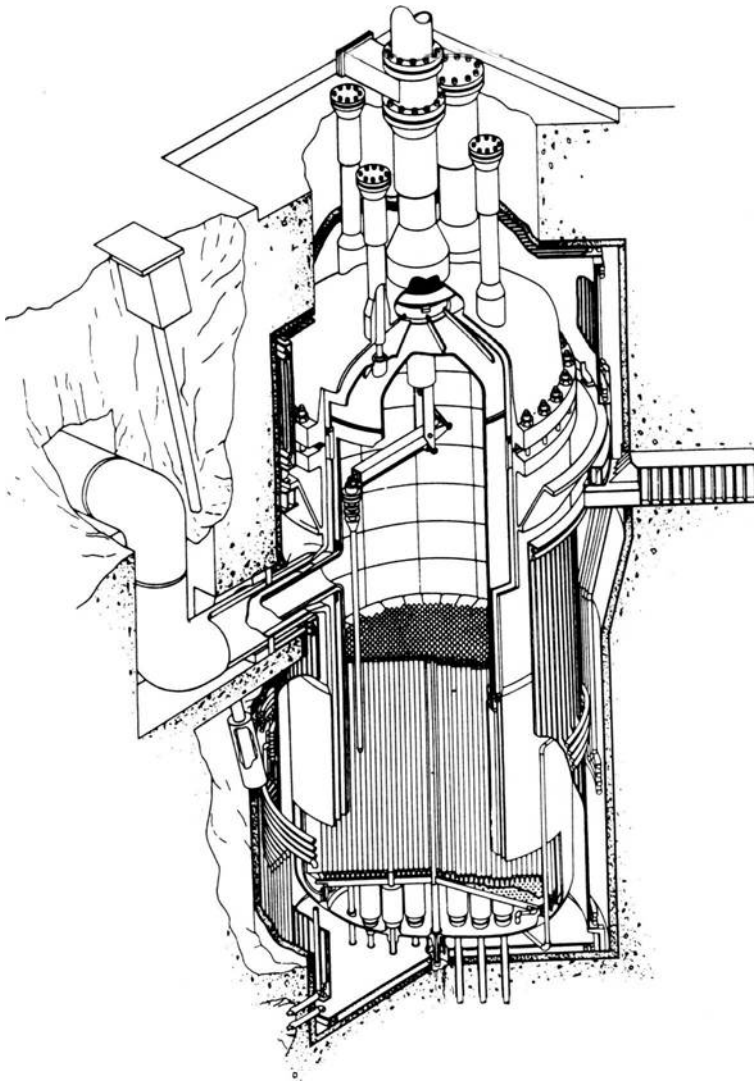
### 5.1.1 Description of PB-1

The PB-1 active core is a cylinder, 2.8 m high, containing 804 fuel elements, 36 control rods, and 19 shutdown rods (Figure 5.2) (Melese and Katz 1984). The fuel elements, 89 mm in diameter, are vertically oriented in a closely packed triangular array with helium flowing up between the elements. The bottom and top graphite reflector sections are an integral part of the fuel element, which has a total height of 3.66 m including the fuel element end fittings. The side reflector, ~60 cm thick, consists of an inner ring of hexagonal graphite elements surrounded by a segmented graphite ring, with a 4 m outer diameter. Helium coolant at 345°C enters the reactor vessel from the outer annuli in the concentric ducts in each of the two loops. It cools the vessel walls and the reflector before flowing up through the core and leaving through the inner concentric ducts at 725°C. The steel reactor pressure vessel, 4.2 m in diameter and 11 m high, is designed for 385°C and 3.1 MPa (31 atm), the actual helium pressure being 2.4 MPa.

The fuel elements are solid and semi-homogeneous with graphite serving as the moderator, cladding, fuel matrix, and structure. They consist of an upper reflector section, a fuel-bearing section, an internal fission product trap, and a bottom reflector. A low permeability graphite sleeve, ~3 m long, is joined to the upper reflector at one end and to a bottom connector fitting at the other end. It contains the fuel bodies consisting of annular fuel compacts, 44 × 69 mm and 76 mm long, stacked on a graphite spine. Uranium and thorium carbide particles, coated with pyrolytic carbon (PyC), are uniformly dispersed in the fuel compacts. The total fuel loading is 236 kg of 93.5% enriched uranium and 1450 kg of thorium. A small fraction of the helium coolant enters the fuel element through a porous plug in the upper reflector piece and flows downward between the fuel compacts and the graphite sleeve. After sweeping fission product gases from the active core zone, the purge



**FIGURE 5.1**  
Broad global foundation of helium reactor technology.



**FIGURE 5.2**  
Isometric view of the peach bottom reactor.

gas flows through the internal traps and then through a purge line to external traps. The fuel elements are designed to stay in the reactor for 3 years and are batch-loaded with the reactor shutdown, consistent with US light water reactor (LWR) conditions. Helium leaving the reactor flows through the two steam generators (one per loop) before being returned to the reactor by horizontal single-stage 3200 rpm electrically driven centrifugal compressors (1.85 MW each). Vertical shell-and-tube forced recirculation steam generators are used, each section of which is constructed of a bank of U-tubes. Each steam generator shell, ~2.4 m in diameter by 9 m high, is cooled by cold helium leaving the economizer. The secondary reactor containment is a vertical, cylindrical-shaped steel shell, 30.5 m in diameter and 49.5 m high, designed for an internal overpressure of 0.055 MPa (0.55 atm) at 65°C.



### 5.1.2 FSV Description

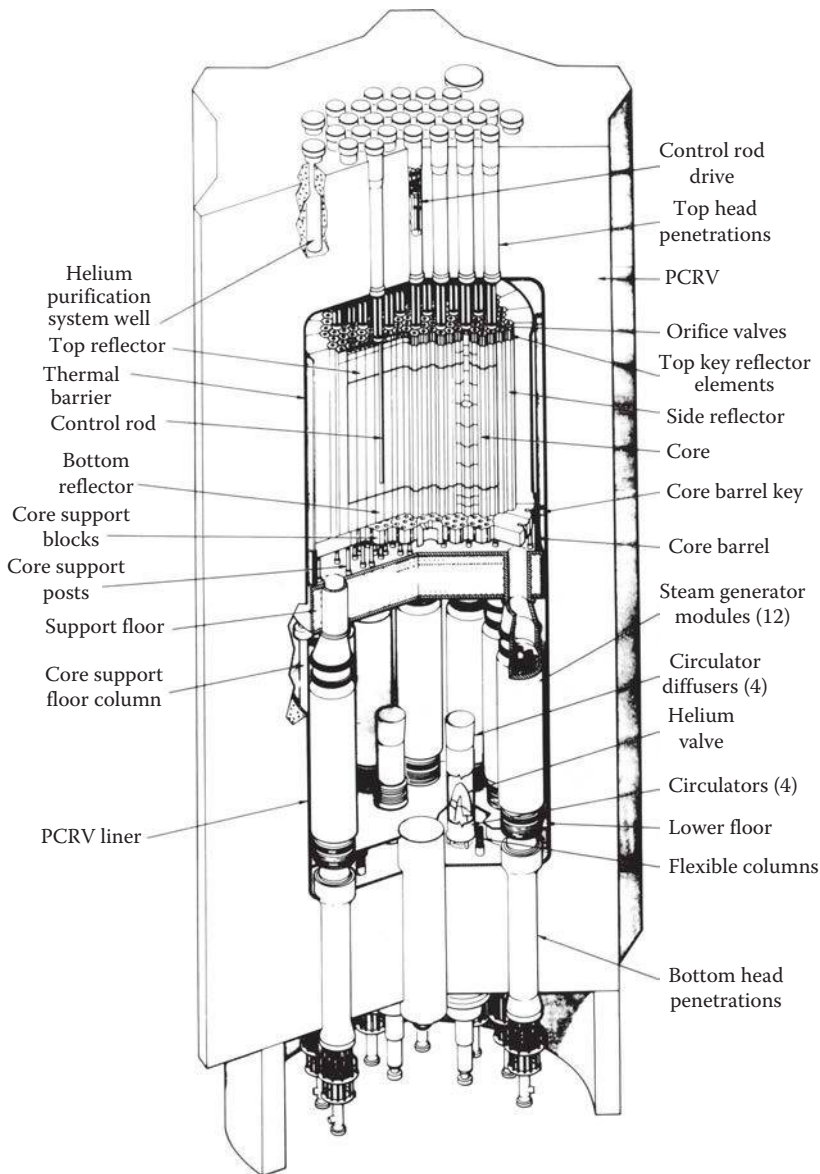
The FSV reactor was designed to produce 842 MW(t) and 330 MW(e), and had many design features similar to the gas-turbine modular helium reactor (GT-MHR) discussed in Sections 5.4 and 5.5, for example, graphite moderation; helium coolant; and very similar designs for fuel particles, fuel elements, and control rods (Baxter et al. 1994). The fuel compacts, which were inserted into machined blind holes in the fuel element, were composed of TRISO-coated fuel particles in a carbonaceous matrix. The TRISO coatings on the fuel particles had been shown to be a highly impervious barrier to radionuclide release in irradiation tests (IAEA 1997). Coolant holes, slightly larger in diameter than the fuel holes, were drilled in parallel through the block to allow the helium to be circulated through the fuel element coolant holes and remove the heat generated in the fuel. The FSV reactor core is composed of 247 columns of fuel elements, with six fuel elements stacked in each column. Axial reflector blocks are also located above and below the core. The core columns were grouped into 37 refueling regions with the flow in each region controlled by an adjustable inlet flow control valve at the top of the core to maintain a fixed core outlet temperature as power changed due to fuel burnup. The 37 refueling regions contain five or seven columns. About one-sixth of the 37 regions were refueled each reactor year. The elements in the central column of each of the 37 refueling regions contained two holes for insertion of control rod pairs and one hole for insertion of reserve shutdown pellets. The control rods consisted of pairs of metal-clad boronated graphite control rods and were operated by electric drives and cable drums. The reserve shutdown pellets were boronated graphite cylinders with spherical ends which could be dropped into the core to provide an independent and diverse reactor shutdown system.

Figure 5.3 is a cut-away view of the FSV reactor core in a prestressed concrete reactor vessel (PCRV), with control rods inserted into the top of the core. The PCRV acted as a pressure vessel, containment, and biological shield. The bottom head had 12 penetrations for the steam generator modules, four penetrations for the helium circulators, and a large central opening for access. A 3/4 in.-thick carbon steel liner anchored to the concrete provided a helium-tight membrane. Two independent systems of water-cooled tubes welded to the concrete side of the liner and kaowool fibrous insulation of the reactor side of the liner limited the temperatures in both the liner and the PCRV.

The primary coolant circuit was wholly contained within the PCRV with the core and reflectors located in the upper part of the cavity and the steam generators and circulators located in the lower part. The helium coolant flowed downward through the reactor core and was then directed into the reheater, superheater, evaporator, and economizer sections of the 12 steam generators. From the steam generators, the helium entered the four circulators and was pumped up around the outside of the core support floor and the core barrel before entering the plenum above the core. The superheated and reheated steam was converted to electricity in a conventional steam cycle power conversion turbine-generator system.

### 5.1.3 Steam Cycle/Variable Cogeneration HTGR Plant Description (Conceptual Design, Circa 1985)

Based on experience obtained from Peach Bottom, FSV, and international programs, the effort in the United States in 1983 was concentrated on the design of a 2240 MW(t) four-loop HTGR steam cycle/variable cogeneration system (Melese and Katz 1984). It would have a maximum electrical output of 820 MW, in the all-electric mode, or a minimum of 231 MW(e) while providing 631 kg/s of high-quality steam (5.9 MPa/538°C). These conditions, obtained with a maximum helium temperature of 690°C, would lead to 38% net efficiency in the all-electric version. Such use of a topping steam cycle for electricity production and



**FIGURE 5.3**  
Isometric view of FSV PCRV, core and primary system.

of reduced pressure steam for process heat applications has the dual advantage of resource conservation and of savings in electricity and/or steam costs. There is a large market potential for steam in the United States; the main problems appear to be institutional. Several modifications have been included in the design of the 2240 MW(t) HTGR compared with the FSV system: a multicavity PCRV rather than a single cavity; a nonreheat steam cycle instead of nuclear reheat; electric motor-driven circulators compared to steam drives; a core auxiliary heat removal system; a reactor secondary containment building; a reduced outlet helium temperature ( $690^{\circ}\text{C}$  vs.  $775^{\circ}\text{C}$ ); and a flexible fuel cycle, that is, 20%–93% uranium

enrichment. Those changes were expected to improve plant performance and reliability, to simplify plant operation and maintenance, and to satisfy projected licensing and regulatory requirements. Participants in the US Department of Energy-funded program included utilities (Gas-Cooled Reactor Associates), a national laboratory (Oak Ridge), and industry (GA Technologies and General Electric) with, as main subcontractors, Bechtel Power, Combustion Engineering, and United Engineers and Constructors. Research and development was also performed on advanced HTGR systems, such as process heat or steam reforming applications, or direct-cycle design for cogeneration applications.

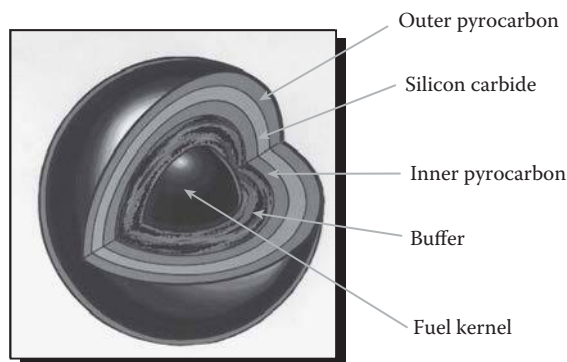
At the time, renewed interest arose in the United States as well as in other countries in small or modular HTGR systems. To obtain acceptable economics compared to fossil-fired plants, the goals of small HTGR power plant designs were simplification of the overall system, reduction of construction time, standardized design, and maximum inherent safety with passive systems. Modular systems could be built on a phased basis, thus relieving the initial investment risk. Improved reliability with multiple units should facilitate applications to process heat. These modular, and more modern, prismatic HTGR designs shall be discussed in Section 5.3.

---

## 5.2 HTGR-Type Comparison and Contrast

### 5.2.1 HTGR-Type Similarities

All HTGR designs utilize a refractory-coated particle fuel. These particles are identified as BISO- or TRISO-coated particle fuel, which consists of a spherical kernel of fissile and/or fertile fuel material (as appropriate for the application), encapsulated in multiple layers of refractory coatings. The multiple coating layers form a miniature, highly corrosion-resistant pressure vessel and an essentially impermeable barrier to the release of gaseous and metallic fission products. The BISO type is no longer in use within currently operating HTGRs, as seen in Table 5.1, mainly due to its inferior performance and fission product retention compared with the TRISO type (Hanson 2004). The BISO is therefore just a historical note, and in Sections 5.2 through 5.6 all fuel particles shall refer to the TRISO type, as shown in Figure 5.4. The overall diameter of standard TRISO-coated particles can vary between 650 to 850  $\mu\text{m}$ , depending upon the burnup goal and type of fuel utilized (fissile vs. fertile).



**FIGURE 5.4**  
TRISO-coated fuel particle.

The fuel kernel may be of oxide, carbide, or oxycarbide in composition. For high burnup applications, an oxycarbide kernel is preferred to enhance performance (Hanson 2004). The carbide component of the kernel undergoes oxidation to get excess oxygen released during fission. If the carbide component were not present, excess oxygen would react with carbon in the buffer to form carbon monoxide. High levels of carbon monoxide can lead to failure of the coating system by overpressurization and kernel migration.

The buffer layer is deposited over the kernel and consists of low-density, porous pyrocarbon.

The buffer attenuates fission fragments that recoil from the kernel and provides sufficient void space to accommodate gases, including gaseous fission products and carbon monoxide. The buffer also acts as a sacrificial layer to accommodate potential kernel migration and swelling and isolates the kernel from load-bearing layers of the coating system.

The high-density inner pyrolytic carbon (IPyC) layer protects the kernel and buffer from chemical attack by chlorine compounds, which are generated as byproducts during deposition of the silicon carbide (SiC) layer. The IPyC layer also provides a surface for deposition of the SiC layer and delays transport of radionuclides to the SiC layer. The IPyC layer shrinks with the accumulation of fast neutron fluence, which helps to maintain the SiC layer in compression, provided the bond between the IPyC and SiC layers remains strong and continuous during irradiation.

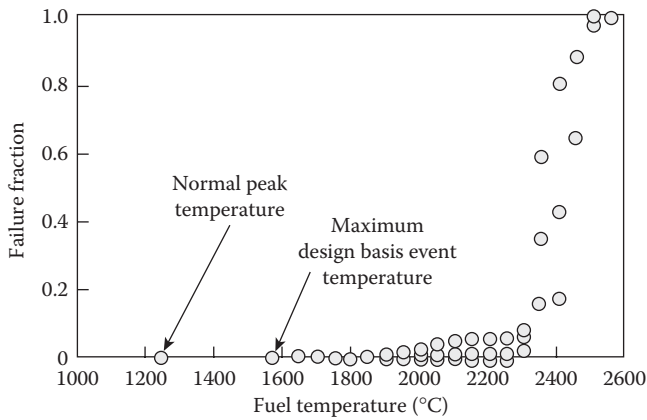
The SiC layer is deposited under conditions that produce a high-density, high-strength coating with a fine-grain microstructure. This layer provides the primary structural support to accommodate stresses generated by internal gas pressure and irradiation-induced dimensional changes of the pyrocarbon layers. The SiC layer provides an impermeable barrier to gaseous, volatile, and most metallic fission products during normal operation and hypothetical accidents. Dimensional changes of the SiC are very small during irradiation, and it is considered to be dimensionally stable.

The high-density outer pyrolytic carbon (OPyC) layer protects the SiC layer from mechanical damage that may occur during fabrication of fuel compacts and fuel elements, and provides a bonding surface for the compact matrix. The OPyC layer also shrinks during irradiation, which helps to maintain the SiC layer in compression. The OPyC layer prevents the release of gaseous fission products if both the IPyC and SiC layers are defective or fail in service.

The TRISO coatings provide a high-temperature, high-integrity structure for retention of fission products to very high burnups. The coatings do not start to thermally degrade until temperatures approaching 2000°C are reached (Figure 5.5). For example, typically for a reactor outlet coolant temperature of 850°C, normal operating fuel kernel temperatures do not exceed about 1250°C, and worst-case accident temperatures are maintained below 1600°C. Extensive tests in the United States, Europe, and Japan have demonstrated the performance potential of this fuel, but tests still need to be done to demonstrate that it satisfies Generation IV performance requirements or normal operating and accident conditions (Hanson 2004).

### 5.2.2 HTGR-Type Differences

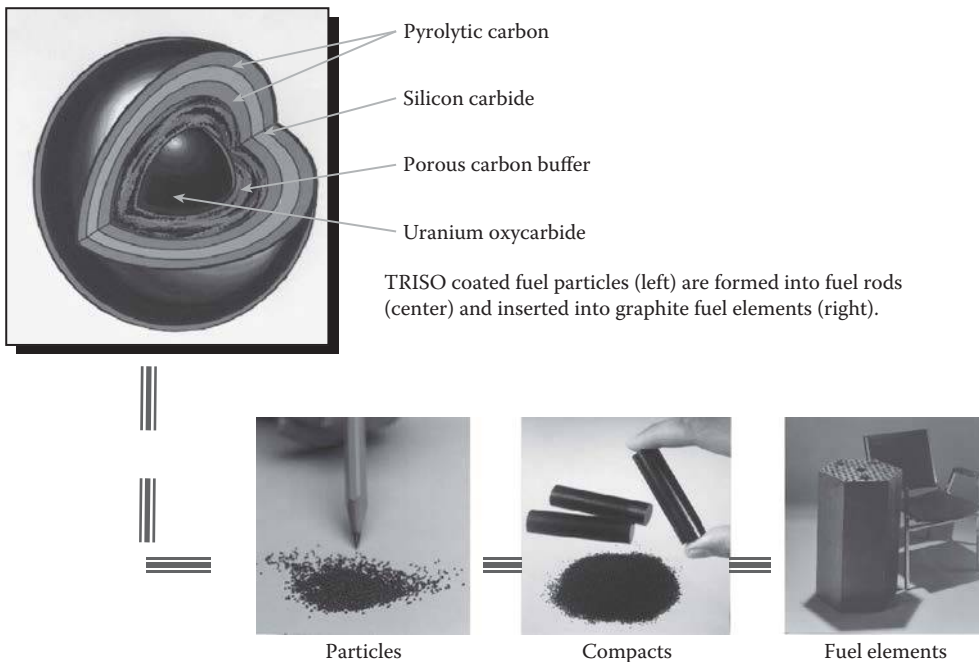
The most important HTGR design distinction is spherical fuel elements versus hexagonal/cylindrical fuel elements. As shown in Table 5.1, the AVR and THTR HTGRs in Germany utilize spherical fuel elements, known as a *pebble-bed reactor*. The remaining HTGRs in Table 5.1 utilize hexagonal/cylindrical fuel elements, known as a *prismatic block reactor*. As previously mentioned, both of these HTGR design concepts use TRISO-coated fuel particles, but the fuel particles are contained in fuel elements having quite different



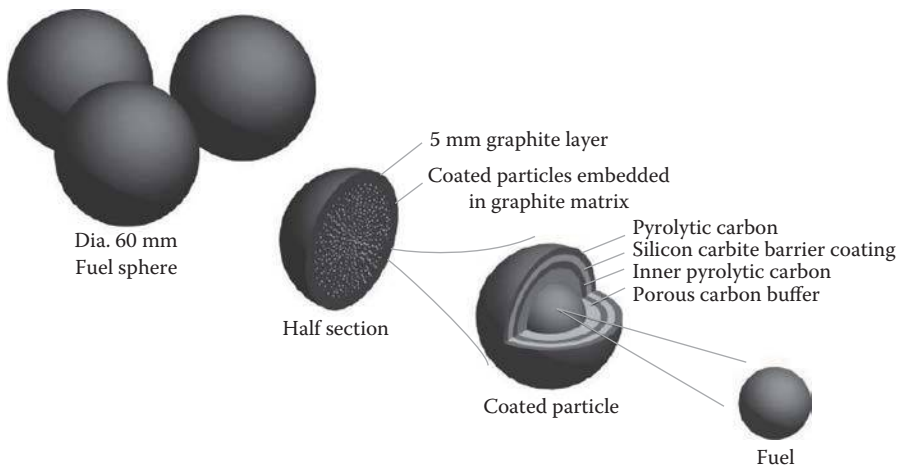
**FIGURE 5.5**  
TRISO-coated particle fuel temperature capability.

configurations, as described below. Generally, prismatic block reactor designs are capable of a higher power density.

In a prismatic block reactor, the TRISO-coated fuel particles are mixed with a carbonaceous matrix and bonded into cylindrical fuel compacts normally 12.5 mm outer diameter × 50 mm long and loaded into fuel holes in hexagonal-shaped graphite fuel blocks that are about 80 cm in height and 36 cm across flats (Figure 5.6). The fuel is cooled by helium that flows downward through vertical coolant channels in the graphite blocks.



**FIGURE 5.6**  
TRISO-coated particle fuel arrangement in hexagonal fuel elements.



**FIGURE 5.7**  
TRISO-coated particle fuel arrangement in spherical fuel elements.

Spent fuel blocks are removed and replaced with fresh fuel blocks during periodic refueling outages.

In a pebble-bed reactor, the fuel particles are contained in billiard ball-sized spherical fuel elements (i.e., pebbles), as shown in Figure 5.7 (PBMR 2005). The fuel is cooled by helium flowing downward through a close-packed bed of the spherical fuel elements. These pebbles are removed continuously from the core during reactor operation, measured for radionuclide content, and returned to the core or replaced with a fresh fuel element depending on the amount of fuel depletion. With this continuous online refueling approach, there is no need for refueling outages.

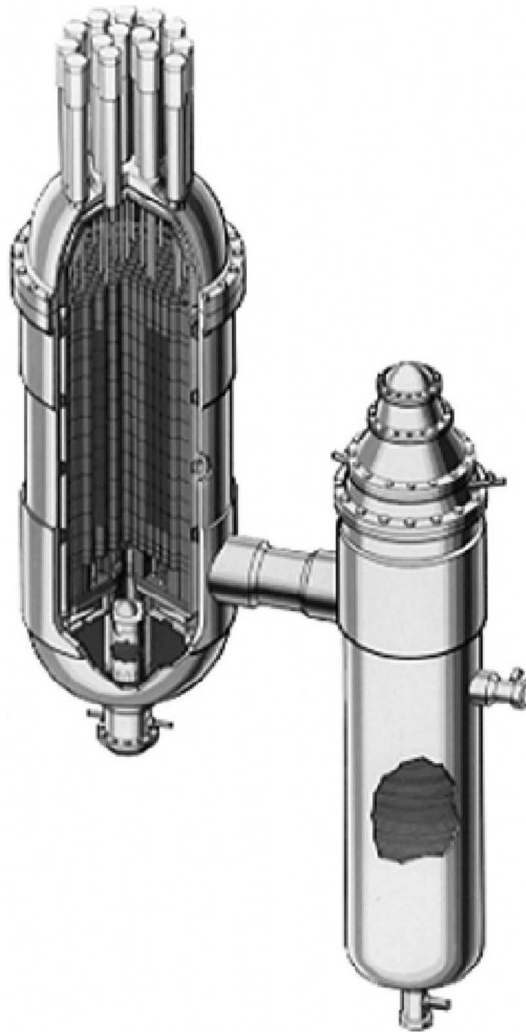
### 5.3 HTGR Design Evolution

Past HTGR designs were first challenged in 1984, when the US Congress asked the HTGR industry to investigate the potential for using their technology to develop a *simpler, safer* nuclear power plant design. This goal of developing a passively safe HTGR plant that was also economically competitive has since stayed with the HTGR industry. In addition to this, and more recently, the Generation IV Forum further challenged HTGR designs to enhance proliferation-resistance and reduce spent fuel inventory. HTGR designers of both types (pebble bed and prismatic block) have responded with a modular reactor approach described in more detail below. As a result, modular HTGR designs started to be considered in the United States in the late 1980s with a thermal rating of 250 MW(t) and a net plant efficiency of ~37% with 4 MPa helium at a top temperature of 690°C. As seen in Table 5.1, because there is limited worldwide experience in HTGR operation, its design evolution and follow-up to construction is even more limited. The 10 MW(t) China HTR-10 (pebble bed) and 30 MW(t) Japan HTTR (prismatic block) reactors are the only operational HTGRs in the world. Although these are only test reactors, having relatively small core thermal power which is completely discharged, they provide critical performance data to help validate a key HTGR evolutionary design change: TRISO fuel particle performance

under high burnup and high coolant outlet temperatures. Several modular (prismatic only) plant design descriptions follow in chronological order.

### 5.3.1 MHTGR Steam Cycle Plant Description (Conceptual Design, Circa 1990)

The reference modular high-temperature gas reactor (MHTGR) steam cycle plant consisted of four identical 350 MW(t) reactor modules with a net electrical output of approximately 550 MW(e) (see Figure 5.8) (Williams et al. 1994). Each module is housed in a vertical cylindrical concrete silo embedded underground. Each silo serves as an independent vented containment structure. The four reactor structures form part of the nuclear island (NI) along with other structures which house systems for helium purification; shutdown cooling; hot cell maintenance; power conditioning; and heating, ventilating, and air conditioning. A storage array cooled by natural circulating air is provided to accommodate



**FIGURE 5.8**  
MHTGR steam cycle plant vessels.

on-site storage of spent fuel in an adjacent reactor service area. The four reactor structures and the reactor service area are covered by a common enclosure which allows sharing of auxiliary cranes and fuel handling equipment.

The MHTGR energy conversion area (ECA), or turbine island, is nonsafety-related and is separated from the NI so that conventional, fossil-fired equipment and standards can be used in its construction and operation. It is located adjacent to the NI so the main steam and feedwater connections between the turbine building and the individual reactor structures will be as short and direct as possible. The reference energy conversion design incorporates two 275 MW(e) nonreheat turbine generator sets, each connected to a pair of reactors. Four stages of feedwater heating are used to optimize the turbine cycle.

The core incorporated a graded low-enriched uranium (LEU) and thorium (LEU/Th) fuel cycle with an equilibrium cycle in which fuel exposure reaches 964 effective full power days (EFPDs) (3.3 calendar years at 80% equivalent availability), with one-half of the active core being replaced every 482 EFPDs (1.65 calendar years at 80% equivalent availability).

### 5.3.2 PS/C-MHR Plant Description (Conceptual Design, Circa 1995)

The process steam/cogeneration modular helium reactor (PS/SC-MHR) was designed to meet the rigorous requirements established by the Nuclear Regulatory Commission (NRC) and the electric utility-user industry for a second-generation power source in the late 1990s (Shenoy 1995). The plant was expected to be equally attractive for deployment and operation in the United States, other major industrialized nations, and the *developing* nations of the world.

At the time, the most economic PS/C-MHR plant configuration included an arrangement of several identical modular reactor units, each located in a single reactor building. The plant was divided into two major areas: the NI, containing several reactor modules, an ECA, containing turbine generators and other balance-of-plant equipment. Each reactor module was designed to be connected independently to steam turbine in or other steam utilizing systems.

The reactor module components are contained within three steel pressure vessels; the reactor vessel, a steam generator vessel, and connecting cross-vessel. The uninsulated steel reactor pressure vessel is approximately the same size as that of a large boiling water reactor (BWR) and contains the core, reflector, and associated supports. The annular reactor core and the surrounding graphite reflectors are supported on a steel core support plate at the lower end of the reactor vessel. Top-mounted penetrations house the control-rod drive mechanisms and the hoppers containing boron carbide pellets for reserve shutdown.

The heat transport system (HTS) provides heat transfer during normal operation or under normal shutdown operation using high-pressure, compressor-driven helium that is heated as it flows down through the core. The coolant flows through the coaxial hot duct inside the cross-vessel and downward over the once-through helical bundle steam generator. Helium then flows upward, in an annulus, between the steam generator vessel and a shroud leading to the main circulator inlet. The main circulator is a helium-submerged, electric motor-driven, two-stage axial compressor with active magnetic bearings. The circulator discharges helium through the annulus of the cross-vessel and hot duct and then upward past the reactor vessel walls to the top plenum over the core.

Major cogeneration applications are highly energy intensive and diverse, including such processes as those associated with heavy oil recovery, tar sands oil recovery, coal liquefaction, h-coal liquefaction, coal gasification, steel mill and aluminum mill processes. Several process heat applications were also considered in the design too, which are discussed in more detail later.



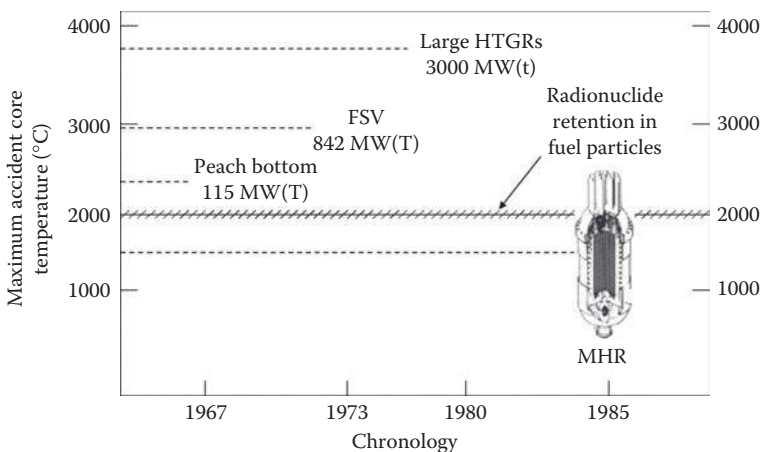
### 5.3.3 GT-MHR Plant Description (Preliminary Design, Circa 2000)

Like most nuclear power plants up to that time, HTGR plants had been designed with reactor core length-to-diameter (L/D) ratios of about 1 for neutron economy. Detailed evaluations showed that low power density HTGR cores with L/Ds of 2 or 3, or more, were effective for rejecting decay heat passively. In the long slender, low power density HTGR cores, it was found that decay heat could be transferred passively by natural means (conduction, convection, and thermal radiation) to a steel reactor vessel wall and then thermally radiated (passively) from the vessel wall to surrounding reactor cavity walls for conduction to a naturally circulating cooling system or to ground itself (Labar et al. 2003).

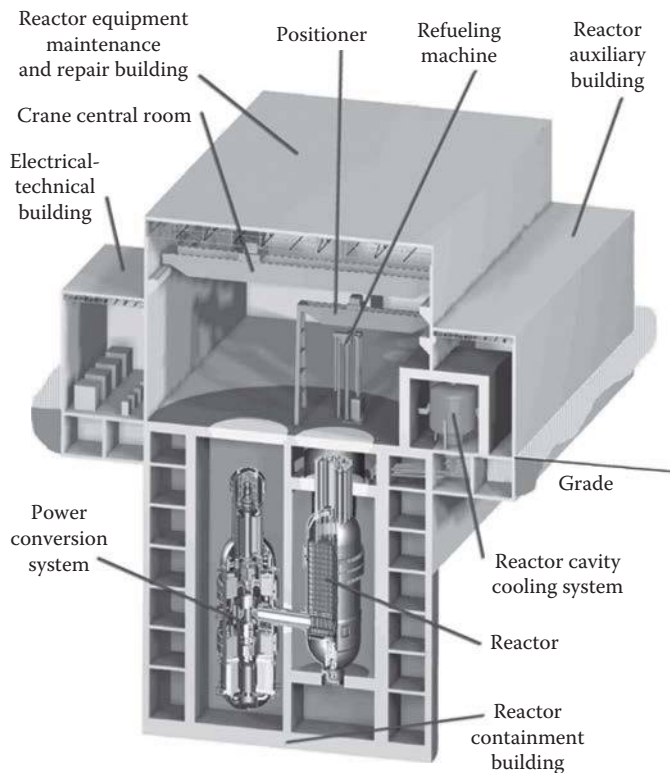
To maintain the coated particle fuel temperatures below damage limits during passive decay heat removal, the core physical size had to be limited, and the maximum reactor power capacity was found to be about 200 MW(t) for a solid cylindrical core geometry. However, a 200 MW(t) power plant was not projected to be economically competitive. This led to the development of an annular core concept to enable larger cores and, therefore, higher reactor powers. The first MHTGR designed with an annular core had a power of 350 MW(t). When coupled with a steam cycle power conversion system (PCS), the plant had a net thermal efficiency of 38% and was economically competitive (marginally) at that time (late 1980s). To improve economics while maintaining passive safety, the core power was subsequently raised to 450 MW(t) and then to the current reference core power of 600 MW(t). The resultant modular HTGR design, now known as the modular helium reactor (MHR), represents a fundamental change in reactor design and safety philosophy, and is shown in Figure 5.9.

The latest evolution made for the purpose of economics has been replacement of the Rankine steam cycle PCS with a high-efficiency Brayton (gas turbine) cycle PCS to boost the thermal conversion efficiency to ~48%. The coupling of the MHR with the gas turbine cycle forms the GT-MHR. The GT-MHR retains all of the MHR passive safety characteristics but is projected to have more attractive economics than any other generation alternative (Shenoy 1996). The organization behind the MHR and GT-MHR designs is General Atomics (GA).

The GT-MHR, seen in Figure 5.10, couples a gas-cooled MHR, contained in one pressure vessel, with a high-efficiency Brayton cycle gas turbine PCS contained in an adjacent



**FIGURE 5.9**  
HTGR prismatic block evolution into the MHR.



**FIGURE 5.10**  
GT-MHR module.

pressure vessel. The reactor and power conversion vessels are interconnected with a short cross-vessel and are located in a below-grade concrete silo. The below-grade silo arrangement provides high resistance to sabotage—a requirement in a post 9/11 world. The GT-MHR share the same Gen-IV goals relating to safety, economics, environmental impact, and proliferation resistance (USDOE 2002), summarized as follows:

*Safety:* The safety design objective is to provide the capability to reject core decay heat relying only on passive (natural) means of heat transfer (conduction, convection, and radiation) without the use of any active safety systems.

*Economics:* The economics design objective is a busbar generation cost (20 year levelized) less than the least cost generation alternative.

*Environmental impacts:* The environmental impact design objectives, relative to the impacts of LWRs, are

- Reduced thermal discharge
- Reduced heavy metal wastes
- Reduced risk of repository spent fuel radionuclide migration to the biosphere

*Proliferation resistance:* The proliferation resistance design objective is a plant and fuel system that has high resistance to sabotage and to diversion of either weapons-usable special nuclear materials or radioactive materials.

The safety design objective is achieved through a combination of inherent safety characteristics and design selections that take maximum advantage of the inherent characteristics. The inherent characteristics and design selections include

- Helium coolant, which is single phase, inert, has only minute reactivity effects and does not become radioactive
- Graphite core, which provides high heat capacity, slow thermal response, and structural stability to very high temperatures
- Refractory coated particle fuel, which retains fission products at temperatures much higher than normal operation and postulated accident conditions
- Negative temperature coefficient of reactivity, which inherently shuts down the core above normal operating temperatures
- A low power density core

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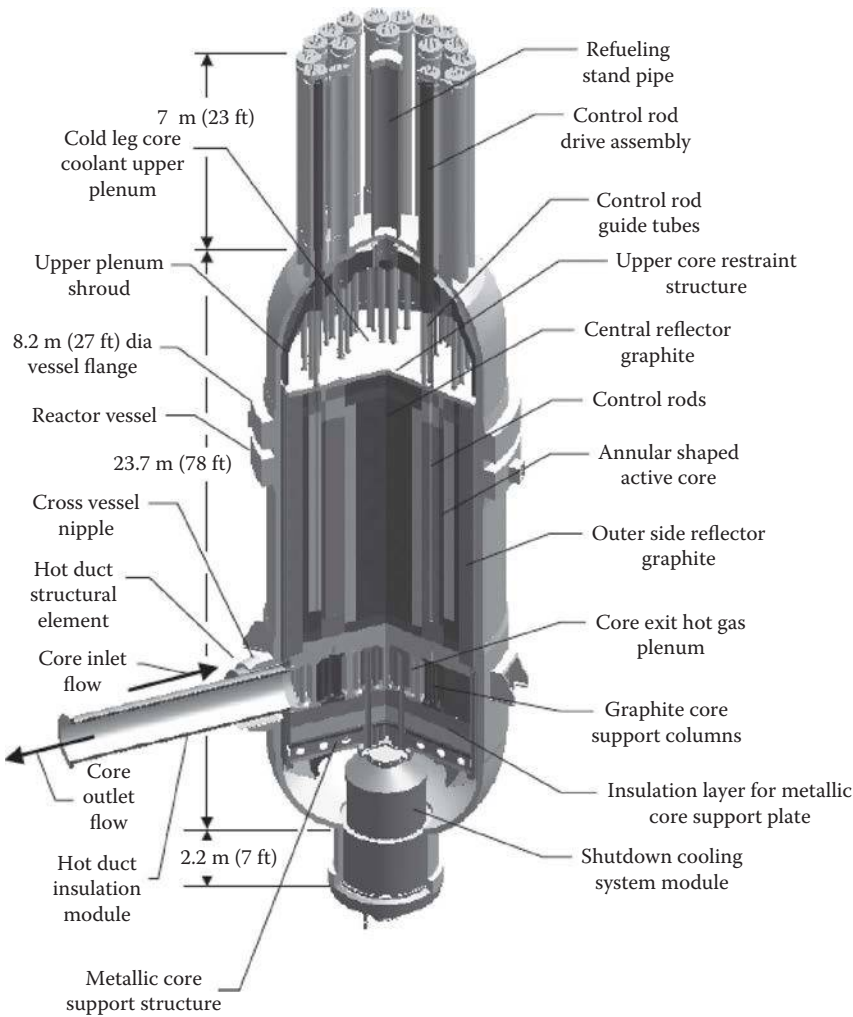
## 5.4 GT-MHR Design

### 5.4.1 GT-MHR Reactor System

Figure 5.11 shows a cross-sectional view of the GT-MHR reactor system, which includes the reactor core, the neutron control system, and other equipment within the reactor vessel. The core design consists of an array of hexagonal fuel elements surrounded by identically sized solid graphite reflector elements vertically supported at the bottom by a core support grid plate structure and laterally supported by a core barrel. The fuel elements are stacked 10 fuel elements high in an annular arrangement of 102 columns (Figure 5.12) to form the active core. The core is enclosed in a steel reactor pressure vessel. Control rod mechanisms are located in the reactor vessel top head, and a shutdown cooling system (SCS) provided for maintenance purposes only is contained in the bottom head.

The mixed mean helium outlet temperature is 850°C. The hot outlet helium flows from the reactor core to the PCS through a hot duct located in the center of the cross-vessel; helium is cooled to 490°C in the PCS and returns to the reactor through the annulus formed between the cross-vessel outer shell and the central hot duct. The cooled helium flows up to an inlet plenum at the top of the core through the annulus between the reactor vessel and the core barrel. From the top inlet plenum, the helium is heated by flowing downward through coolant channels in the fuel elements, collected in a bottom outlet plenum, and guided into the cross-vessel hot duct. All the core components exposed to the heated helium are either graphite or thermally insulated from exposure to the high temperature helium. Graphite has high strength, does not readily combust, and has dimensional stability to very high temperatures (~2300°C).

Because of the accident at Chernobyl in 1986, the role of graphite in reactor safety has received increased attention. However, the consequences of the Chernobyl accident were caused by massive fuel failure and not by graphite oxidation that occurred during the accident. Decay heat from the nuclear fuel was sufficient to maintain relatively high graphite temperatures for an extended period of time, causing the graphite to radiate the *red glow* that was observed during the accident. High-purity, nuclear-grade graphite reacts very slowly with oxygen and would be classified as noncombustible by conventional standards.



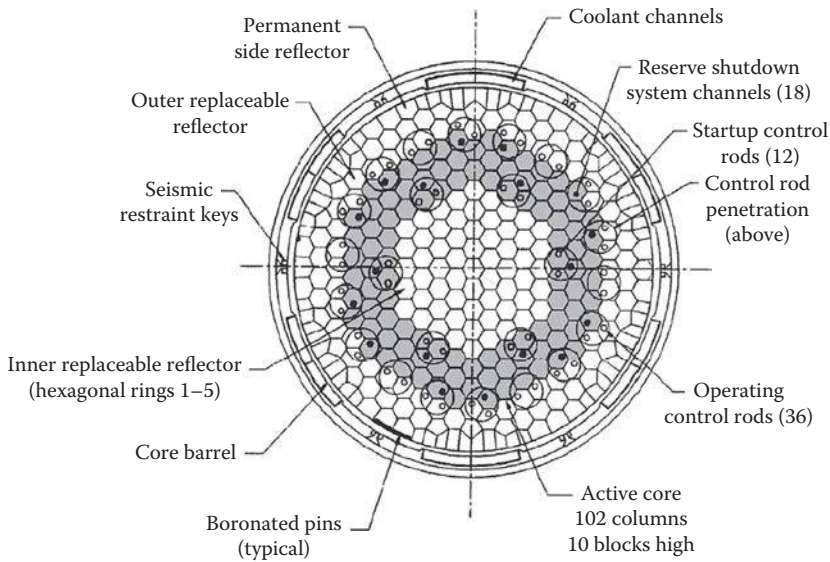
**FIGURE 5.11**  
GT-MHR reactor system.

In fact, graphite powder is a class D fire extinguishing material for combustible metals, including zirconium. For the GT-MHR (and the PBMR for that matter), the oxidation resistance and heat capacity of graphite serves to mitigate, not exacerbate, the radiological consequences of a hypothetical severe accident that allows air into the reactor vessel.

Tables 5.2 and 5.3 show an outline of GT-MHR nominal plant design parameters and GT-MHR-coated particle fuel design parameters, respectively.

#### 5.4.2 GT-MHR PCS

The GT-MHR direct Brayton cycle (gas turbine) PCS contains a gas turbine, an electric generator, and gas compressors located on a common, ~29 m long vertically orientated shaft supported by magnetic bearings. The PCS also includes recuperator, precooler, and intercooler heat exchangers. Heated helium flows (Figure 5.13) directly from the MHR into



**FIGURE 5.12**  
GT-MHR annular core cross section at vessel mid-plane.

**TABLE 5.2**

GT-MHR Nominal Plant Design Parameters

<i>MHR System</i>	
Power rating (MW(t))	600
Core inlet/outlet temperatures (°C)	491/850
Peak fuel temperature—normal operation (°C)	1250
Peak fuel temperature—accident conditions (°C)	<1600
Helium mass flow rate (kg/s)	320
Core inlet/outlet pressures (MPa)	7.07/7.02
<i>Power Conversion System</i>	
Helium mass flow rate (kg/s)	320
Turbine inlet/outlet temperatures (°C)	848/511
Turbine inlet/outlet pressures (MPa)	7.01/2.64
Recuperator hot side inlet/outlet (°C)	511/125
Recuperator cold side inlet/outlet (°C)	105/491
Net plant efficiency (%)	48
Net electrical output, 1 module (MW(e))	286

a gas turbine to drive the generator and gas compressors. From the turbine exhaust, the helium flows through the hot side of the recuperator, through the precooler, and then passes through low- and high-pressure compressors with intercooling. From the high-pressure compressor outlet, the helium flows through the cold, high-pressure side of the recuperator where it is heated for return to the reactor.

The use of the direct Brayton cycle to produce electricity results in a net plant efficiency of approximately 48% as shown in Figure 5.14. This efficiency is ~50% higher than that in current LWR nuclear power plants.

**TABLE 5.3**  
GT-MHR-Coated Particle Design Parameters

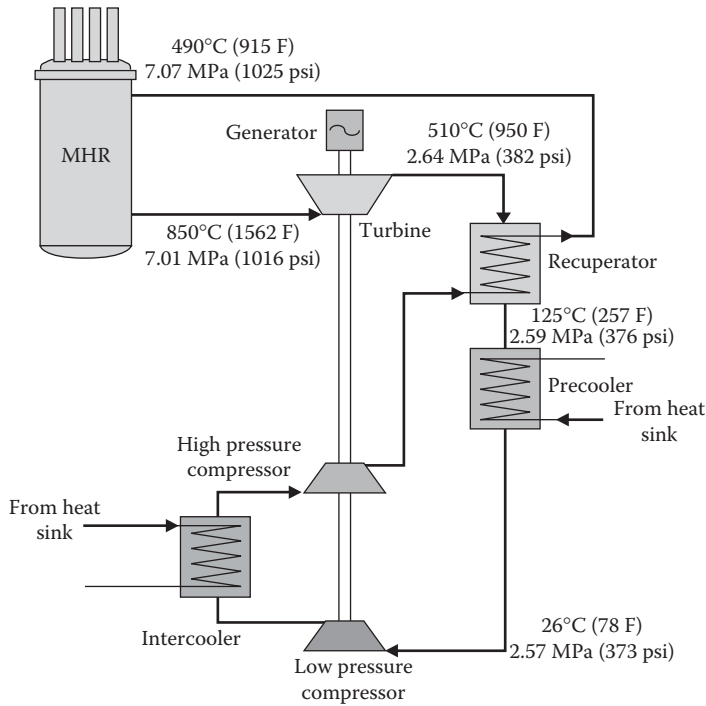
	Fissile Particle	Fertile Particle
Composition	UC <sub>0.5</sub> O <sub>1.5</sub>	UC <sub>0.5</sub> O <sub>1.5</sub>
Uranium enrichment, %	19.8	0.7 (Natural Uranium)
<i>Dimensions (μm)</i>		
Kernel diameter	350	500
Buffer thickness	100	65
IPyC thickness	35	35
SiC thickness	35	35
OPyC thickness	40	40
Particle diameter	770	850
<i>Material Densities (g/cm<sup>3</sup>)</i>		
Kernel	10.5	10.5
Buffer	1.0	1.0
IPyC	1.87	1.87
SiC	3.2	3.2
OPyC	1.83	1.83
<i>Elemental Content Per Particle (μg)</i>		
Carbon	305.7	379.9
Oxygen	25.7	61.6
Silicon	104.5	133.2
Uranium	254.1	610.2
Total particle mass (μg)	690.0	1184.9
Design burnup (% FIMA) <sup>a</sup>	26	7

<sup>a</sup> FIMA is an acronym for fissions per initial metal atom.

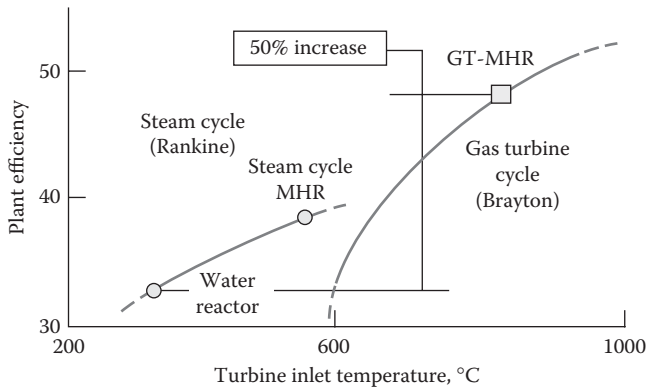
The GT-MHR gas turbine PCS has been made possible by key technology developments during the last several years in large aircraft and industrial gas turbines; large active magnetic bearings; compact, highly effective gas-to-gas heat exchangers; and high strength, high temperature steel alloy vessels. The selection of (1) the direct cycle PCS and (2) integrated vertical shaft PCS arrangement was made on the basis of achieving optimum economics from consideration of several alternatives. There are several alternative high-efficiency Brayton cycle PCS and arrangements that could be used. Some of these would require less development effort but would have higher capital cost and electricity generation cost.

#### 5.4.3 GT-MHR Heat Removal System

The GT-MHR has two active, diverse active heat removal systems, the PCS and the SCS that can be used for the removal of decay heat. In the event that neither of these active systems is available, an independent passive means is provided for the removal of core decay heat. This is the reactor cavity cooling system (RCCS) that surrounds the reactor vessel (Figure 5.15). For passive removal of decay heat, the core power density and the annular core configuration have been designed such that the decay heat can be removed by conduction to the pressure vessel (Figure 5.16) and transferred by



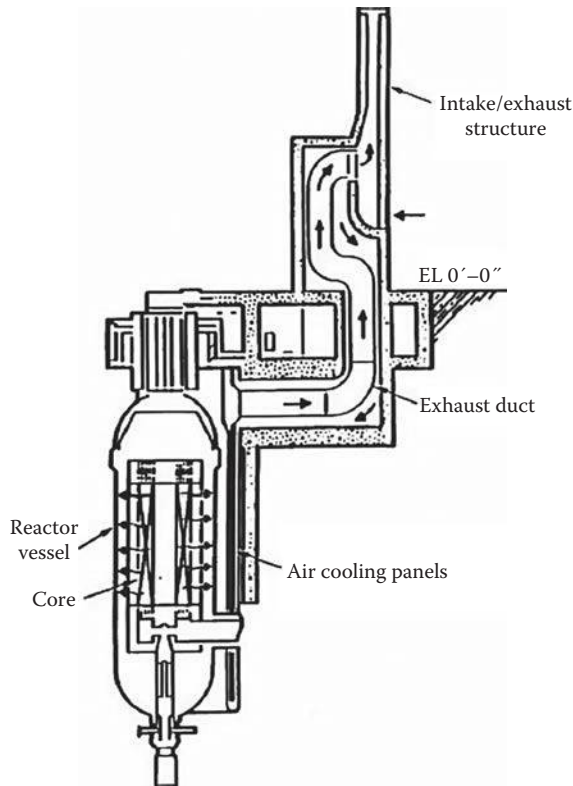
**FIGURE 5.13**  
GT-MHR coolant flow schematic.



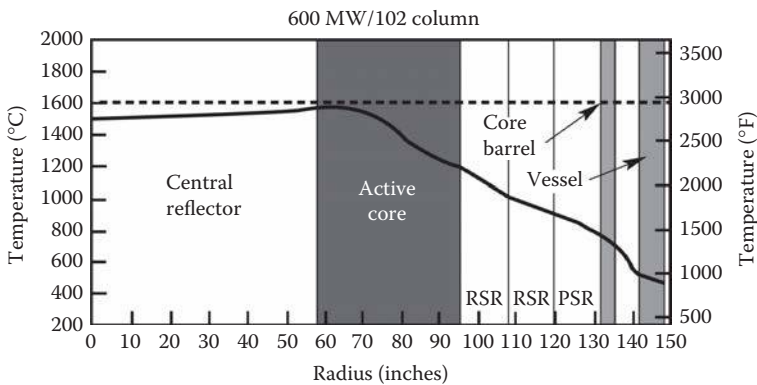
**FIGURE 5.14**  
Comparison of thermal efficiencies.

radiation from the vessel to the natural circulation RCCS without exceeding the fuel particle temperature limit (Figure 5.17).

Even if the RCCS is assumed to fail, passive heat conduction from the core, thermal radiation from the vessel, and conduction into the silo walls and surrounding earth (Figure 5.18) is sufficient to maintain peak core temperatures to below the design limit. As a result, radionuclides are retained within the refractory-coated fuel particles without the need for active systems or operator action. These safety characteristics and design



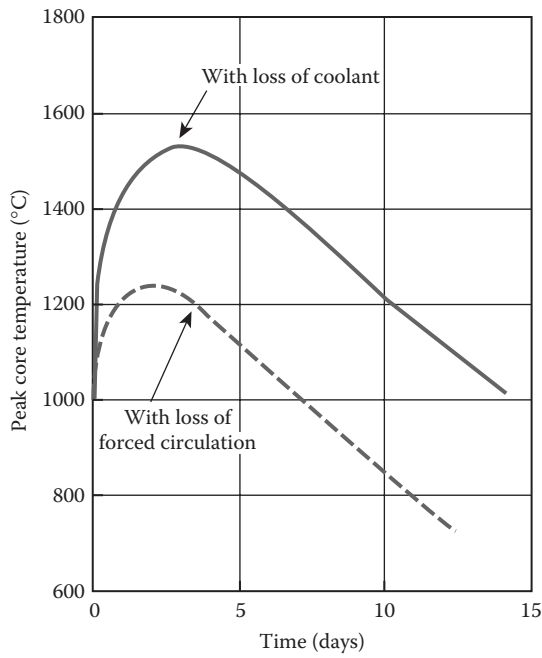
**FIGURE 5.15**  
GT-MHR passive reactor cavity cooling system.



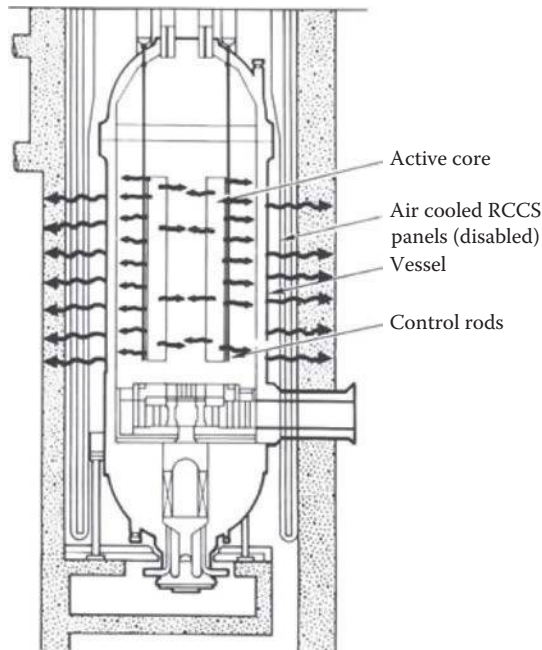
**FIGURE 5.16**  
GT-MHR radial temperature gradient during after-heat rejection to RCCS.

features result in a reactor that can withstand loss of coolant circulation, or even loss of coolant inventory, and maintain fuel temperatures below damage limits (i.e., the system is meltdown proof). The core graphite heat capacity is sufficiently large that any heat up, or cooldown, takes place very slowly. A substantial time (of the order of days vs. minutes for other reactors) is available to take corrective actions to mitigate abnormal events and to restore the reactor to normal operations.





**FIGURE 5.17**  
GT-MHR core heat-up temperatures with passive after heat rejection.



**FIGURE 5.18**  
GT-MHR passive radiation and conduction of after heat to silo containment.

#### 5.4.4 GT-MHR Environmental Characteristics

The GT-MHR has significant environmental impact advantages relative to LWR plants (Table 5.4) between a four-module GT-MHR plant and a large PWR. The thermal discharge (waste heat) from the GT-MHR is significantly less than the PWR plant because of its greater thermal efficiency. If this waste heat is discharged using conventional power plant water heat rejection systems, the GT-MHR requires <60% of the water coolant per unit of electricity produced. Alternatively, because of its significantly lesser waste heat, the GT-MHR waste heat can be rejected directly to the atmosphere using air-cooled heat rejection systems such that no water coolant resources are needed. Because of this capability, the use of the GT-MHR in arid regions is possible.

The GT-MHR produces less heavy metal radioactive waste per unit energy produced because of the plant's high thermal efficiency and high fuel burnup. Similarly, the GT-MHR produces less total plutonium and Pu-239 (materials of proliferation concern) per unit of energy produced.

The deep-burn capability and high radionuclide containment integrity of TRISO particles offer potential for improvements in nuclear spent fuel management. A high degree of degradation of plutonium and other long-life fissile actinides can be achieved by the deep-burn capability. Nuclear design analyses of the MHR deep-burn concept indicate that, in one pass through the reactor, virtually complete destruction can be accomplished of weapons-usable materials (Pu-239), and up to 90% of all transuranic waste, including near total destruction of neptunium-237 (the most mobile actinide in a repository environment) and its precursor, americium-241. The resultant particles contain significantly reduced quantities of long-life radionuclides and very degraded fissile materials that can then be placed in a geologic repository with high assurance that the residual products have insufficient interest for intentional retrieval and will not migrate into the biosphere by natural processes before decay renders them benign.

**TABLE 5.4**

Resource Consumption and Environmental Impact Comparison

Plant Parameters	Large PWR	GT-MHR
Thermal power (MW(t))	3914	4 × 600
Electric power (MWe)	1385	1145
60 year power generation (GWY)	72.3	59.8
<i>Thermal Discharge</i>		
Heat rejection (GWt/GWe)	1.8	1.1
Cooling water req'd (10 <sup>4</sup> Acre-Ft/GWY)	2.4	1.4
<i>Equilibrium Fuel Cycle</i>		
Heavy metal loading (MT/GWt)	26.8	7.5
U Enrichment (%)	4.2	15.5 (Avg)
SWU Demand (103 kg-SWU/GWY)	135	221
U <sub>3</sub> O <sub>8</sub> Consumption (MT/GWY)	181	246
Full power days per cycle	432	460
<i>Spent Fuel Discharge</i>		
Discharged heavy metal (MT/GWY)	21.4	5.4
Discharged Pu (kg/GWY)	235	109
Discharged Pu <sup>239</sup> (kg/GWY)	171	43

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## 5.5 GT-MHR Fuel Cycles

### 5.5.1 GT-MHR Uranium Fuel Cycle

A commercialization option of the GT-MHR started development at GA in 1993 to produce electricity at competitive generation costs and is a promising candidate for near-term commercial deployment in the United States. Two different types of fuel TRISO particles are used for power profiling purposes: 19.9% low-enriched (LEU) particles and natural uranium (NU) particles. The design used a once-through fuel cycle, refueling half of the core at every reload interval (Shenoy 1996).

### 5.5.2 GT-MHR Plutonium Fuel Cycle

When fueled with weapon-grade plutonium (94% enriched Pu-239), the GT-MHR can provide the capability to consume >90% of the initially charged Pu-239, and >65% of the initially charged total plutonium, in a single pass through the reactor. This option is referred to as a *plutonium consumption MHR* (PC-MHR) and was under development from 2000 to 2013 in a joint United States–Russian Federation program to provide capacity for disposition of surplus weapons plutonium. The design was also a once-through fuel cycle type; however, only one-third of the core is replaced during refueling.

### 5.5.3 GT-MHR Thorium Fuel Cycles

#### 5.5.3.1 GT-MHR HEU/Th Fuel Cycle

This fuel cycle is based upon FSV-type fuel, which operated from 1976 through 1989. Fuel composition consists again of two separate TRISO particles, 93% highly enriched uranium (HEU) particles and fertile Th-232 particles to achieve maximum U-233 conversion ratios and therefore limit the amount of plutonium produced. Although HEU-fueled reactors would not be considered for commercial use in the United States, the interest here is historical in nature. This design also uses a once-through fuel cycle, refueling half of the core at every reload interval.

#### 5.5.3.2 GT-MHR LEU/Th (Single Particle) Fuel Cycle

This fuel cycle concept was initially conceived at GA in 1977 and promoted as a *nonproliferation* design option because fissile and fertile fuels co-exist in the same TRISO fuel particle. This design effectively denatures the U-233 produced from the fertile Th-232 fuel by mixing it with nonfissile plutonium nuclides generated from the 19.9% LEU. A significant quantity of Pu-238 is also produced so that the plutonium would also generate a considerable amount of decay heat, thereby making the depleted fuel less attractive as bomb material. This design would also use a once-through fuel cycle, refueling half of the core at every reload interval.

#### 5.5.3.3 GT-MHR LEU/Th (Dual Particle) Fuel Cycle

This fuel cycle was considered as a method for achieving much longer fuel cycle lengths and extended burnup due to an expected higher conversion ratio from the thorium

breeding. By separating the 10.9% LEU TRISO particles from the Th-232 TRISO particles, the MHR can simulate the fertile *blanket effect* utilized in fast breeder reactors. Because the bred U-233 would not be denatured here, this may likely be a closed fuel cycle to recycle the bred fissile uranium.

#### 5.5.4 GT-MHR Mixed Actinide Fuel Cycles

##### 5.5.4.1 Deep-Burn MHR (DB-MHR)

This fuel cycle's sole fuel source uses reprocessed transuranic waste discharged from LWRs. The fissile plutonium (obtained after a AFCI-UREX or other similar process) becomes the main driver fuel for this cycle. Fortunately, the core neutron spectrum allows for significant neutron capture in the resonance region by several minor actinides mixed in with the plutonium. As a result, this design provides its own negative reactivity control without the need for burnable poisons. Over 96% of the initial Pu-239, including over 60% of the initial actinide nuclides, can be destroyed in this cycle.

##### 5.5.4.2 Self-Cleaning MHR (SC-MHR)

This fuel cycle combined discharged and recycled TRU waste from an LEU or mixed LEU/Thorium (LEU/Th) fuel cycle with the fresh fuel. The mixed-core fuel is composed of 80% fresh fuel and 20% discharged and recycled TRU waste. This is therefore essentially a closed-cycle LEU or LEU/Th fuel cycle. Through recycling-bred fissile and minor actinide nuclides from a cycle discharge, very high actinide destruction is possible, approximately >80%.

Table 5.5 shows some MHR parameters of all presented fuel cycle options (Ellis and Baxter 2004).

**TABLE 5.5**

MHR Parameters of All Fuel Cycle Options

MHR Fuel Cycle Option	LEU/NU	LEU/Th	PC-MHR	DB-MHR	SC-MHR
Number of TRISO particle types	2	2	1	2	2
Fuel description	UC <sub>0.5</sub> O <sub>1.5</sub>	UC <sub>0.5</sub> O <sub>1.5</sub> /ThO <sub>2</sub>	PuO <sub>1.7</sub>	TRU oxides	LEU/Th TRU oxides
Reactor thermal power (MW(t))	600	600	600	600	600
Fuel cycle length (EFPD)	477	950 <sup>a</sup>	317	540	540 <sup>a</sup>
Net thermal efficiency (%)	47.7	47.7	47.7	47.7	47.7
Average EC uranium enrichment (%)	15.5	19.9	0	0	19.9
Average EC plutonium enrichment (%)	0	0	94	60	60
EC uranium loading (kg)	2262	1552 <sup>a</sup>	0	0	1242 <sup>a</sup>
EC thorium loading (kg)	0	710 <sup>a</sup>	0	0	568 <sup>a</sup>
EC plutonium loading (kg)	0	0	369	237	47 <sup>a</sup>

Note: EC, equilibrium cycle; EFPD, effective full power days.

<sup>a</sup> Best estimate.

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## 5.6 MHR Next-Generation Potential Applications

### 5.6.1 Nonelectric Applications

#### 5.6.1.1 Alumina Plant

Aluminum refining uses two major energy-intensive processes:

1. Aluminum oxide or alumina is obtained from bauxite via the Bayer chemical process. This process uses a significant amount of steam to react with bauxite and for mechanical drive. It also requires electric power.
2. Alumina is reduced to aluminum by electrolysis. This process requires large amounts of electric power.

Most existing commercial aluminum plants use energy from natural gas power plants. Hydroelectric power supplies a very small fraction of the total aluminum electric power requirements. An MHR could be utilized for producing alumina from bauxite. For the size alumina plant considered, a two module 600 MW(t) PS/C-MHR can supply 100% of the process steam and electrical power requirements and produce surplus electrical power and/or process steam, which can be used for other process users or electrical power production. Presently, the bauxite ore is reduced to alumina in plant geographically separated from the electrolysis plant. However, with the integration of  $2 \times 600$  MW(t) PS/C-MHR units in a commercial alumina plant, the excess electric power available, 233 MW(e), could be used for alumina electrolysis. It has been shown that the steam and electrical energy requirements for a typical commercial alumina plant processing 726,680 t (800,000 t) per year of alumina ( $\text{Al}_2\text{O}_3$ ) can be satisfied by a two module PS/C-MHR (Shenoy 1995).

#### 5.6.1.2 Coal Gasification

Several countries are interested in developing plants producing gaseous synthetic fuels derived from coal, based on their national objective to reduce foreign oil imports and to use or export the abundant coal. Exxon catalytic coal gasification (ECCG) is a gasification process developed in the United States. Initially, coal gasification plants are expected to obtain thermal power requirements from fossil sources (coal or product liquid and gaseous fuel from the synfuel plant) and to obtain electric power partly from in-plant cogeneration and partly from local utilities. Most processes are estimated to consume 25%–30% of the feed coal to satisfy the plant energy needs. The ECCG process uses alkali metal salts as a gasification catalyst with a novel processing sequence. Although no net heat is required for the gasification reaction, heat input is required for drying and preheating the feed coal, gasifier heat losses, and catalyst recovery operations. Mechanical drives and plant electrical power also have energy input requirements. An MHR could provide thermal and electrical energy for the ECCG process to benefit worldwide interests by conserving fossil fuel and reducing environmental impact.

#### 5.6.1.3 Coal Liquefaction

The solvent-refined coal (SRC-II) process is an advanced process developed by Gulf Mineral Resources Ltd. to produce a clean, nonpolluting liquid fuel from high sulfur bituminous

coals. Coupling of two module 600 MW(e) PS/C-MHR to the SRC-II process could commercially process 24,300 t (26,800 t) of feed coal per stream day, producing primarily fuel oil and secondary fuel gases (Shenoy 1995).

In the SRC-II process, the process steam is generated by direct gas-fired boilers and the process heating by direct gas firing. The fuels utilized are hydrocarbon-rich gas, or CO-rich gas, and purified syngas (i.e., no feed coal is used for fuel). It was shown that a  $2 \times 600$  MW(t) PS/C-MHR can supply these thermal requirements principally by substituting for the fuel gases previously employed (Shenoy 1995). The displaced gases, which are treated already, may then be marketed.

The 538°C (1000°F) steam supply of the PS/C-MHR provides all system thermal energy requirements in the form of process steam generation, steam superheating, and slurry heating. However, slurry heating by steam will entail the development of a new heat exchanger design. The  $2 \times 600$  MW(t) PS/C-MHR does not generate all the required electrical energy, and a deficit of  $-38$  MW(e) results.

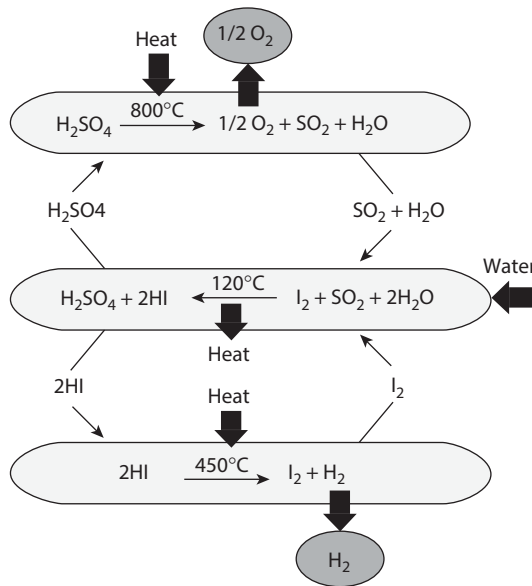
#### 5.6.1.4 H-Coal Liquefaction

In countries with large coal reserves, a strong interest exists to develop and commercialize plants producing liquid and gaseous synthetic fuels derived from coal because of their national objective to reduce foreign oil imports or to export liquid coal. The H-coal liquefaction is one process which can be used to convert coal into liquid fuel. The H-coal process has several advantages over other processes, including an isothermal reactor bed, hydrogenation of the coal with a direct, continuously replaceable catalyst (i.e., no dependence on catalytic effects of coal ash), and the absence of quench injections (which would be required with a series of fixed beds).

#### 5.6.1.5 Hydrogen Production

A significant *hydrogen economy* is predicted to limit dependence on petroleum and reduce pollution and greenhouse gas emissions. Hydrogen is an environmentally attractive fuel but contemporary hydrogen production is primarily based on fossil fuels. The United States produces  $\sim 11$  million tons of hydrogen a year by steam reformation of methane for use in refineries and chemical industries, and the use is growing by  $\sim 10\%$  per year. This is the thermal energy equivalent of 48 GW(t) and consumes about 5% of our natural gas usage. Use of hydrogen for all the transportation energy needs in the United States would require a factor of 18 more hydrogen than currently used. Clearly, new sources of hydrogen will be needed. Nuclear energy can be one of the sources.

Hydrogen can be produced from nuclear energy by several means. Electricity from nuclear power can separate water into hydrogen and oxygen by electrolysis (Richards 2006b). The net efficiency is the product of the efficiency of the reactor in producing electricity times the efficiency of the electrolysis cell, which, at the high pressure needed for distribution and utilization, is about 75%–80%. If a GT-MHR with 48% electrical efficiency is used to produce the electricity, the net efficiency of hydrogen production could be about 36%–38%. Electrolysis at high temperature, providing some of the energy directly as heat, promises efficiencies of about 50% at 900°C. Thermochemical water-splitting processes similarly offer the promise of heat-to-hydrogen efficiencies of  $\sim 50\%$  at high temperatures. Thermochemical water splitting is the conversion of water into hydrogen and oxygen by a series of thermally driven chemical reactions that could use nuclear energy as the heat source.



**FIGURE 5.19**  
Sulfur iodine thermochemical water-splitting cycle.

The sulfur iodine (S-I) thermochemical water-splitting cycle has been determined to be best suited for coupling to a nuclear reactor (Richards 2006a). The S-I cycle (Figure 5.19) consists of three chemical reactions, which sum to the dissociation of water. Only water and high temperature process heat are input to the cycle and only hydrogen, oxygen, and low temperature heat are output. All the chemical reagents are regenerated and recycled. There are no effluents. An intermediate helium heat transfer loop would be used between the primary coolant loop and the hydrogen production system. With an outlet temperature of  $850^\circ\text{C}$ , a maximum temperature of  $825^\circ\text{C}$  is estimated for the process heat to the process, which yields 43% efficiency. At a reactor outlet temperature of  $950^\circ\text{C}$  and a  $50^\circ\text{C}$  temperature drop across an intermediate heat exchanger, an efficiency of 52% is estimated.

#### 5.6.1.6 Steel Mill

The US steel industry is very large and consumes large quantities of energy. It uses 35% of this energy in the form of electricity, fuel oil, or natural gas; the balance is coal. Therefore, the supply of the noncoal energy by an MHR can conserve scarce fossil fuel resources.

A  $2 \times 600 \text{ MW(t)}$  PS/C-MHR plant can satisfy the energy requirements for a typical commercial steel mill to produce  $6.5 \times 10^6 \text{ t}$  ( $7.2 \times 10^6 \text{ t}$ ) (liquid) of steel per year (Shenoy 1995). The surplus energy, which may be generated either as steam at  $5.0 \text{ MPa}$  ( $725 \text{ psia}$ ) and  $365^\circ\text{C}$  ( $689^\circ\text{F}$ ) at  $125 \text{ kg/s}$  ( $106 \text{ l/h}$ ) or  $\sim$  electric power [ $-100 \text{ MW(e)}$ ], can be exported outside the plant. Depending on the steel mill location, steam could be supplied to neighboring industries or, alternatively, the electric power can be sold to a utility.

#### 5.6.1.7 Synthetic Fuels

Increasing world demand for oil, and the perception that conventional (i.e., liquid) oil reserves have peaked, has caused oil prices to sky rocket and has renewed interest in

unconventional oil reserves in the form of oil shale and tar sands, which are estimated to hold several times more oil than the current liquid reserves. It has also spurred interest in synthetic fuel production and in better methods for recovery of heavy oil from operating wells where production is dropping. Economic oil recovery from any of these areas requires high-temperature gas or high-temperature steam. An HTGR is uniquely suited for these applications because it can produce the high-temperature gas and the high-temperature steam at the conditions required for these processes in an environmentally acceptable manner (i.e., without burning natural gas). It can also co-produce electricity for oil field and on-site uses, and the hydrogen needed to convert the hydrogen deficient, heavy crude, into a refinable, syncrude product.

GA's objective for the synthetic fuels program is to develop pre-conceptual MHR designs for each of these applications, including good cost estimates and construction schedules, which can be used to obtain government agency (US Department of Energy [DOE], DoD), or oil company funding for detailed design studies leading to the construction of a demonstration MHR plant. The PS/C-MHR version of the reactor will be used because it is based on Peach Bottom and FSV experience, has been reviewed at the preliminary safety information document (PSID) level by the NRC, has realistic cost estimates, and could be put on line in a short time frame without a large technology development program.

#### **5.6.1.8 Research/Test Reactors**

In 2006, the University of Texas of the Permian Basin (UTPB) made a partnership with GA, The University of Texas System, and with the participation of local city and county governments as well as with the collaboration of other academic, industrial, and government laboratories, and proposed to construct and operate a high-temperature teaching and test reactor (HT<sup>3</sup>R) as a multifaceted energy research facility. Its proposed location is near the UTPB campus in Andrews County, Texas.

The mission of the HT<sup>3</sup>R is to be a research and test facility that can support the education and training of the next-generation of nuclear scientists and engineers, as well as the performance of high-temperature R&D on materials and processes for the economic production of electricity, hydrogen, synthetic hydrocarbon fuels, and desalinated water. This primary mission will be supported by facilities for research, development, and pilot-scale testing programs including a radiation laboratory, high temperature materials and process laboratory, and an energy transfer laboratory. The HT<sup>3</sup>R will be the cornerstone for a new UTPB research and development *Center of Excellence* that will investigate new frontiers in the applications of high-temperature materials, processes, plus nuclear science and engineering R&D. The HT<sup>3</sup>R will be an HTGR with passively safe design features. The HTGR is also a leading candidate for the development of Generation IV reactors meant to provide significant improvement over existing power reactors with regards to safety, economics, proliferation-resistant fuel cycles, and flexibility of applications. Outlet temperatures of 850°C to >950°C will lead to a variety of applications with the potential for significantly higher thermal efficiencies.

The proposed design of the HT<sup>3</sup>R and its associated facilities are synergistic with the proposed next-generation nuclear plant (NGNP) authorized by Congress for deployment at the Idaho National Laboratory as well as the Global Nuclear Energy Partnership (GNEP) that has been proposed by the US President. With the planned physical and operating characteristics of the HT<sup>3</sup>R being very similar to the proposed commercial-scale HTGR plants, the HT<sup>3</sup>R can significantly benefit the NGNP development by reducing key identified risks.



### 5.6.2 Proliferation Resistance Applications

The GT-MHR has very high proliferation resistance due to low fissile fuel volume fractions and the refractory characteristics of the TRISO fuel particle coating system that forms a containment from which it is difficult to retrieve fissile materials.

GT-MHR fresh fuel and spent fuel have higher resistance to diversion and proliferation than the fuel for any other reactor option. The GT-MHR fresh fuel has high proliferation resistance because the fuel is very diluted by the fuel element graphite (low fuel volume fraction). GT-MHR spent fuel has the self-protecting, proliferation resistance characteristics of other spent fuel (high radiation fields and spent fuel mass and volume). However, GT-MHR spent fuel has higher proliferation resistance than any other power reactor fuel because of the reasons given below.

- The quantity of fissile material (plutonium and uranium) per GT-MHR spent fuel element is low (50 times more volume of spent GT-MHR fuel elements would have to be diverted than spent LWR fuel elements to obtain the same quantity of Pu-239).
- The GT-MHR spent fuel plutonium content, the material of most proliferation concern, is exceedingly low in quantity per spent fuel block and quality because of high fuel burnup. The discharged plutonium isotopic mixture is degraded well beyond LWR spent fuel, making it particularly unattractive for use in weapons.
- No process has yet been developed to separate the residual fissionable material from GT-MHR spent fuel. While development of such a process is entirely feasible (and potentially desirable sometime in the future), there is no existing readily available process technology such as for spent LWR fuel. Until such time as when the technology becomes readily available, the lack of the technology provides proliferation resistance.

The TRISO fuel particle coating system, which provides containment of fission products under reactor operating conditions, also provides an excellent barrier for containment of the radionuclides for storage and geologic disposal of spent fuel. Experimental studies have shown that the corrosion rates of the TRISO coatings are very low under both dry and wet conditions. The coatings are ideal for a multiple-barrier, waste management system. The measured corrosion rates indicate the TRISO coating system should maintain its integrity for a million years or more in a geologic repository environment.

### 5.6.3 Sustainability Applications

GA is currently performing parametric studies on the LEU/Th (dual particle) fuel cycle in hopes of utilizing the thorium breeding to significantly expand the fuel cycle length while requiring less fuel ore.

### 5.6.4 Implementation Requirements

There are two primary advantages to HTGR technology: the inherent safety of a core that does not require any coolant to prevent fuel damage in an accident and the high efficiency that comes from the high temperature. An extended station blackout event like Fukushima would not be a serious casualty for an HTGR. The DOE is conducting a fuel qualification program with very promising initial results. So from an engineering standpoint these

features, linked with the wide range of applications beyond electricity, would indicate a promising future for HTGR implementation.

However, the HTGR's Achilles' heel is its low power density—at best about one-tenth of today's water reactors. This results in a core that is significantly more expensive due to its size and due to the complexity of TRISO particle-based fuel fabrication relative to today's reactors of a similar power level (revenue generation capability). Recent interest in the NGNP project to build an HTGR in the United States has been stalled primarily due to the competition from inexpensive US natural gas.

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# 6

## *Integrated Fast Reactor: PRISM*

Maria Pfeffer, Scott Pfeffer, Eric Loewen, Brett Dooies, and Brian Triplett

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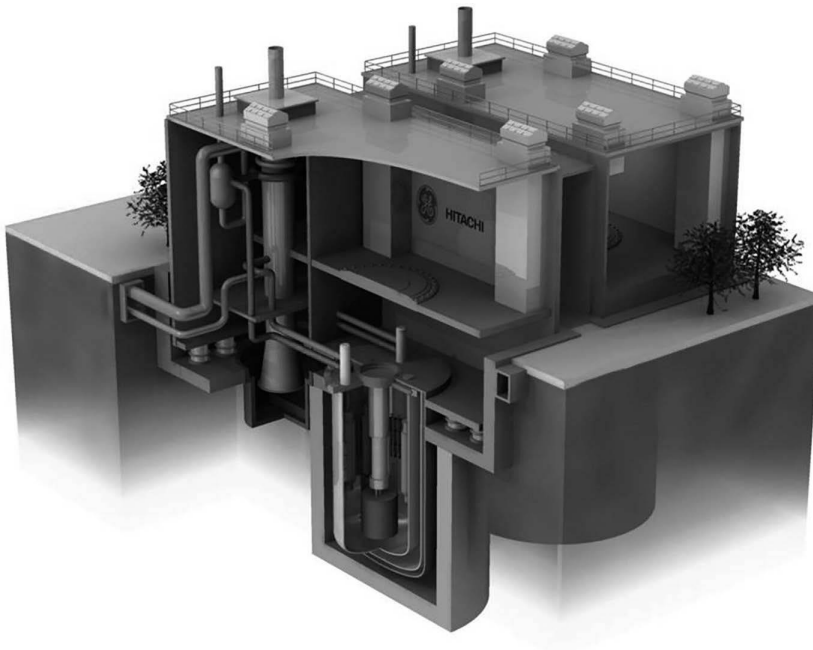
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## 6.1 PRISM

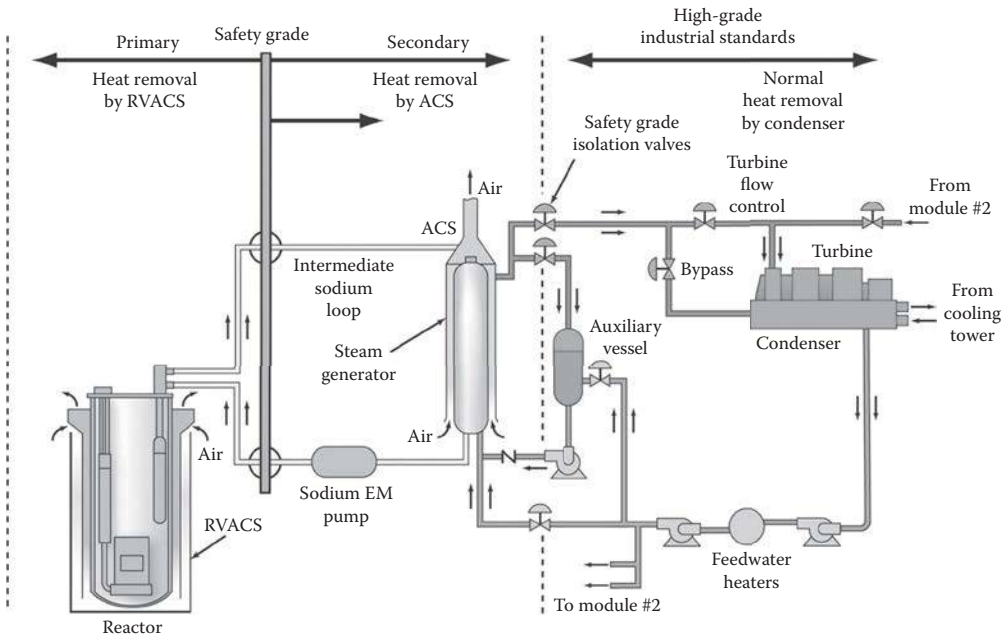
The Power Reactor Inherently Safe Module (PRISM), designed by GE Hitachi Nuclear Energy (GEH), is a small, modular, sodium-cooled fast reactor (Figure 6.1). The PRISM core is located in a pool-type containment vessel and is fueled with metallic fuel. Each PRISM module produces 311 MWe of electricity. PRISM is inherently safe due to its negative power reactivity feedback, large in-vessel coolant inventory, passive heat removal systems, below ground siting, and atmospheric reactor vessel operating pressure. PRISM is able to fission a variety of nuclear fuel sources including electrometallurgically recycled used nuclear fuel (UNF) from light water reactors (LWRs) as well as legacy separated materials such as plutonium. The PRISM design is often coupled with an electrometallurgical used fuel recycling process.

PRISM employs passive safety, digital instrumentation and control, and modular fabrication techniques to expedite plant construction [1–4]. PRISM has a rated thermal power of 840 MW and an electrical output of 311 MWe. Each PRISM module has an intermediate sodium loop that exchanges heat between the primary sodium coolant from the core with water/steam in a sodium–water steam generator (SG). The steam from the sodium–water SG feeds a conventional steam turbine. A diagram of the PRISM nuclear steam supply system (NSSS) is shown in Figure 6.2.

Two PRISM reactor modules are paired to form one power block. A PRISM power block is shown in Figure 6.1. The power block supplies steam for one 622 MWe turbine generator. Plant electrical output can be tailored to local needs by the modular addition of power blocks.



**FIGURE 6.1**  
PRISM.



**FIGURE 6.2**  
PRISM nuclear steam supply system.

## 6.2 General Electric’s History of Sodium-Cooled Fast Reactors

The concept of a liquid metal-cooled reactor dates back to the genesis of nuclear energy. The first nuclear reactor to generate electricity was the liquid sodium–potassium-cooled fast reactor Experimental Breeder Reactor-I (EBR-I) [5]. EBR-I’s successor, the sodium-cooled fast reactor (SFR) EBR-II operated successfully for over 30 years, producing 20 MW of electricity via a sodium-steam power cycle [6].

General Electric (GE) started work on sodium-cooled power reactor systems in 1946 with the establishment of the Knolls Atomic Power Laboratory. That effort was successfully applied to the United States’ second nuclear submarine, USS *Seawolf* (SSN575), which operated at sea for 2 years. The submarine’s reactor, S2G, was designed and built by GE.

Spending a significant amount of effort in active scientific, developmental, engineering, construction, and reactor operations, GE developed vast expertise in the area of SFRs. In 1961, after the initially favorable results in the GE LWR fuel program, GE proposed that a small test reactor be built to explore the Doppler effect during operation. This reactor used an oxide fuel to compare GE’s calculation to the performance of a relatively large ceramic fuel fast reactor core. The reactor, built in Arkansas and called the southwest experimental fast oxide reactor (SEFOR), had three simple goals: gather enough technical information from SEFOR to progress to the next major step in the commercial development of fast reactors; develop an organization with the necessary experience to accomplish the next major steps in the field of sodium-cooled reactors; and determine if the GE assessment of the SFR was justified through first-hand experience with the design, construction, and operation of a liquid metal-cooled fast reactor built on a commercial basis.

In the early 1980s, the liquid metal fast breeder reactor (LMR) program focused on deployment of the Clinch River Breeder Reactor (CRBR) in Tennessee. The program encountered difficulties because of cost escalations and schedule delays. The LMR program faced challenges because uranium was not becoming scarce and prohibitively expensive as had been predicted earlier.

While the CRBR project was being debated, a small group at GE's advanced reactors program pursued a technology other than large loop sodium reactors. At the time, the 1000 MWt CRBR was envisioned as the stepping-stone to 3000 MWt commercial plants—the increased scale was believed to be necessary to be economically competitive with the large LWRs. GE questioned the economics of large fast reactors and conducted internal work based on an alternative small modular approach. This small reactor, with rated power in the range of 400–1000 MWt, could provide stair-step plant power levels by adding reactor modules at a site to reach economic, power generation goals. This was the genesis of GE's PRISM.

In August of 1981, representatives from Argonne National Laboratory (ANL)'s Special Project Office visited the GE advanced reactor team. GE shared the idea of the relatively small PRISM reactor vessel, which could be transported to a refueling center about every 18 months. ANL shared their in-core refueling machine process for the EBR-II. It became apparent that rather than moving an entire reactor, technology was available to move just the fuel. From this synergistic meeting between GE and the national laboratory, the concept of PRISM matured.

When Congress terminated the CRBR project in 1983, the Department of Energy (DOE) began the advanced liquid metal reactor (ALMR) program. The goal of the ALMR program was to increase the efficiency of uranium usage by breeding plutonium and create the condition wherein transuranic (TRU) isotopes would never leave the site. The ALMR was designed to allow any TRU isotope to be consumed as fuel and is the forerunner to the Global Nuclear Energy Partnership (GNEP), and what is now the International Framework for Nuclear Energy Cooperation (IFNEC).

GE joined the ALMR program with its two key elements: reactor design and fuel cycle development. GE led industry partners to refine the conceptual design of the PRISM reactor. The national laboratories, led principally by ANL, tackled the fuel cycle development and waste characterization.

The ALMR program was funded from 1984 to 1994. Two products emerged: the advanced conceptual PRISM reactor design and the proliferation resistant pyroprocess for spent fuel recycling. Following the discontinuation of DOE's ALMR program, GE continued to develop PRISM.

In 2006, GEH revitalized the PRISM engineering efforts as part of the DOE's GNEP [7]. Under GNEP, GEH proposed the Advanced Recycling Center (ARC) as a commercial solution for nuclear fuel recycling. The ARC is comprised of the nuclear fuel recycling center (NFRC) and PRISM. Together, these two technologies are used to produce electricity from recycled UNF. The ARC consists of six PRISM reactor modules, each producing a net 311 MW of electricity. Two PRISM reactor modules are paired together in a power block to supply a 622 MW turbine generator. The total ARC electrical output is 1866 MW. The fuel for the six PRISM reactors is supplied by a single NFRC, which is able to process LWR UNF, PRISM UNF, and legacy fissile material into PRISM fuel.

At end of life, LWR UNF is comprised of approximately 95% uranium, 1% TRUs, and 4% fission products [8]. Many of these TRU isotopes have long half-lives that can create long-term engineering challenges for geologic disposal [9]. The integral fast reactor (IFR)

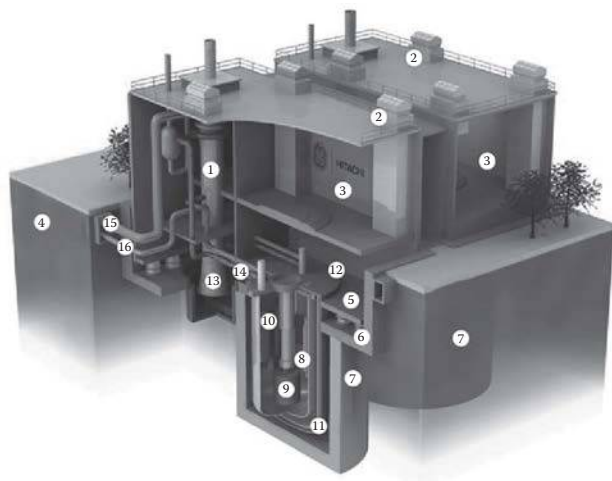
concept proposes the recycling of the 96% of the fissionable material remaining in LWR UNF (uranium and TRUs) [10]. In this process, the uranium and TRU material is continually recycled via electrometallurgical processing (pyroprocessing) in a metal-fueled SFR. The ARC is the commercialization of this IFR concept. PRISM within the ARC is designed to generate electricity by fully recycling the reusable fissionable material in UNF [11].

### 6.3 PRISM Technology

PRISM is a fast neutron reactor plant design with passive reactor shutdown, passive shutdown heat removal, and passive reactor cavity cooling. PRISM supports a sustainable and flexible fuel cycle to consume TRU elements within the fuel as it generates electricity. The essence of the reactor technology is a reactor core housed within a 316 stainless steel reactor vessel. Liquid sodium is circulated within the reactor vessel and through the reactor core by four electromagnetic (EM) pumps suspended from the reactor closure head. Two intermediate heat exchangers (IHX) inside the reactor vessel remove heat for electricity generation.

The PRISM reactors are positioned in a power block with two reactors side by side supporting a single steam turbine generator set. The reactor plant is divided into two areas: the nuclear island (reactors through SGs) and the turbine island to generate electricity. The nuclear island consists of two reactors in separate containments, plus SGs, and shared services, in a single, seismically isolated, partially underground building as depicted in the cutaway view of a PRISM nuclear island shown in Figure 6.3. Each reactor heats an intermediate coolant loop, sending heat to an SG. Steam from the SGs is combined and sent to the turbine generator that produces electricity. Figure 6.2 shows the overall PRISM power

1. Steam generator
2. Reactor vessel auxiliary cooling system
3. Refueling enclosure building
4. Steam tunnel to turbine
5. Reactor protection system modules
6. Seismic isolation bearing
7. Reactor module
8. Primary electromagnetic pump
9. Reactor core
10. Intermediate heat exchangers
11. Lower containment vessel
12. Upper containment building
13. Sodium dump tank
14. Intermediate heat transfer system
15. Steam outlet piping
16. Feedwater return piping



**FIGURE 6.3**  
PRISM nuclear Island.

**TABLE 6.1**

PRISM Plant Parameters

<i>Overall Plant</i>	
Electrical output	1866 MW
Thermal efficiency	37%
Number of power blocks (per plant)	3
Number of reactor modules (per plant)	6
<i>Power Block</i>	
Number of reactor modules (per block)	2
Net electrical output	622 MW
Number of steam generators	2
Steam generator type	Helical coil
Steam cycle	Superheated steam
Turbine type	3600 rpm, tandem compound, four flow
Turbine throttle conditions	14.7 MPa (2135 psia)/452°C (846°F)
Main steam flow	2738 Mg/h (6.024 × 10 <sup>6</sup> lb/h)
Feedwater temperature	216°C (420°F)
<i>Reactor Module</i>	
Thermal power	840 MW
Primary sodium inlet/outlet temperature	360°C (680°F)/499°C (930°F)
Primary sodium flow rate	5.4 m <sup>3</sup> /s (86,000 gpm)
Intermediate sodium inlet/outlet temperature	326°C (619°F)/477°C (890°F)
Intermediate sodium flow rate	5.1 m <sup>3</sup> /s (80,180 gpm)

train that converts TRUs into electricity. A summary overview of the nominal PRISM plant parameters is provided in Table 6.1.

PRISM can provide the energy to generate electricity while burning spent fuel from operating LWRs as well as future LWRs. Because of its relatively small size and its inherently safe encapsulated design, PRISM can, in large part, be factory built and transported to the site.

PRISM is often coupled with an integrated, commercial-scale, recycling facility capable of processing LWR and fast reactor UNF and fabricating PRISM TRU fuel for the co-located PRISM power blocks.

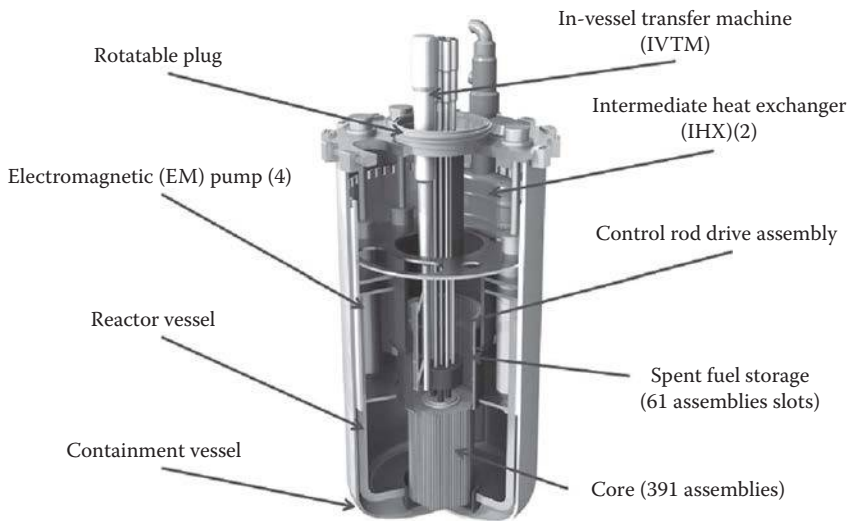
## 6.4 PRISM Design

A PRISM module, pictured in Figure 6.4, consists of the reactor vessel, reactor closure, containment vessel, internal structures, internal components, reactor module supports, and the reactor core.

### 6.4.1 Nuclear Heat Supply System

Nuclear heat is removed from the reactor core by the primary heat transport system (PHTS). The PHTS is entirely contained within the reactor vessel and is composed of the





**FIGURE 6.4**  
PRISM module.

reactor core, the hot pool, the tube-side of the IHX, the cold pool, the EM pumps, the pump discharge piping, and the core inlet plenum. Within each reactor module, four EM pumps circulate the primary sodium through two IHX. The primary coolant flow path is shown in Figure 6.5.

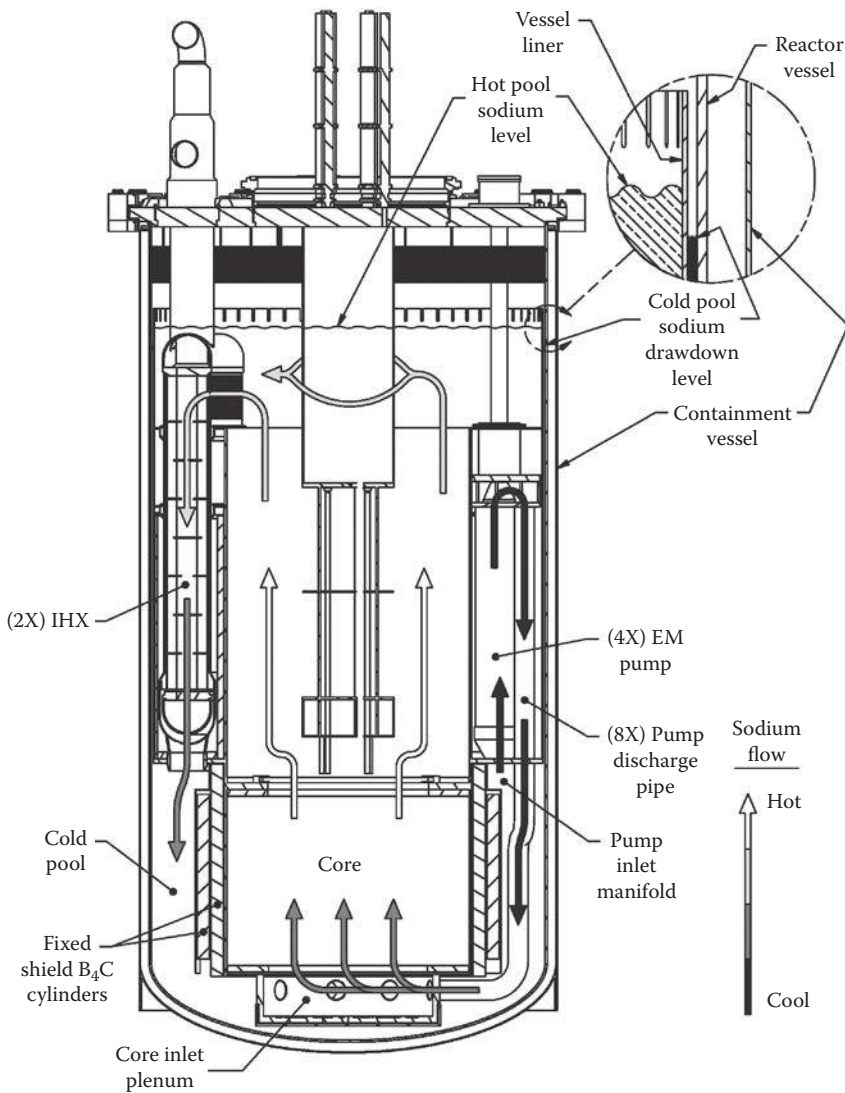
### 6.4.2 Containment

The PRISM containment system uses three successive barriers—fuel cladding, primary coolant boundary (reactor vessel cutaway view shown in Figure 6.4), and a containment boundary that surrounds the reactor vessel—to provide defense-in-depth from postulated releases from the reactor vessel. The function of the PRISM containment is to protect the public from any accidental release of radioactivity from the reactor. The containment consists of the containment vessel surrounding the reactor vessel and a lower containment over the reactor closure. The containment boundary is a steel-lined concrete upper structure that encloses the reactor module as shown in Figure 6.4.

The containment vessel has no penetrations and is designed to be leak tight. It contains the reactor vessel and internals and is sealed with a reactor closure. The inside of the reactor vessel is filled with liquid sodium and a helium cover gas. The helium cover gas is at approximately atmospheric pressure at normal power conditions. A 20 cm annulus between the containment vessel and the reactor vessel is filled with argon at a pressure slightly above the reactor cover gas pressure.

The PRISM reactor is positioned underground in a concrete silo—a fourth containment boundary (Figure 6.3). In the unlikely event of a containment breach, the sodium complies with the natural law of gravity and is contained in the silo. The silo is sized to retain all of the primary sodium while keeping the core, stored spent fuel, and heat exchanger inlets covered with sodium.

The reactor closure is a single plate with a rotatable plug and penetrations for the reactor equipment, primary sodium cleanup, and cover gas service lines. Primary sodium cleanup is accomplished during reactor shutdown by a single cold trap system, shared by each

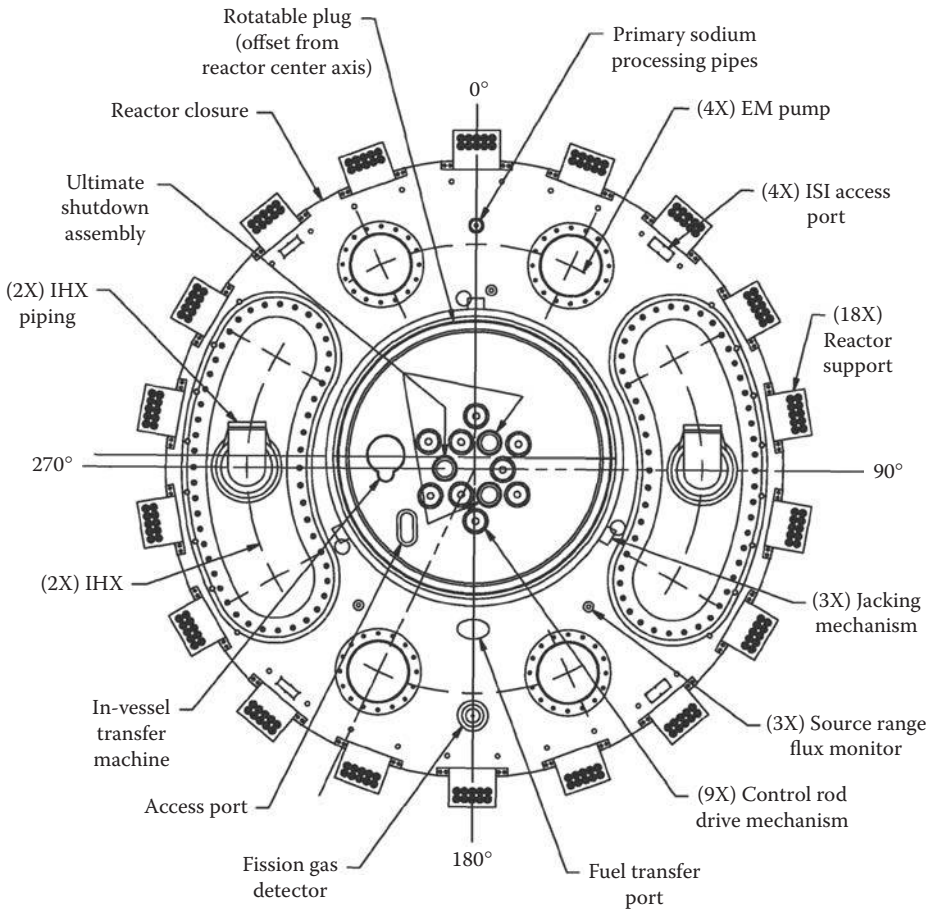


**FIGURE 6.5**  
PRISM primary heat transport system.

power block. The closure penetrations are shown in Figure 6.6. The reactor vessel is butt-welded to a lip on the underside of the closure. The containment vessel is bolted to the closure and sealed by welding. The reactor module is supported entirely from the closure by hold-down brackets. The closure-containment vessel connection and support detail is shown in Figure 6.7.

### 6.4.3 Reactor Internals

The reactor vessel houses the reactor core, which is the heat source for steam generation. The vessel contains this heat, transfers the heat to the IHX, and serves as one of the fission product barriers during normal operation.

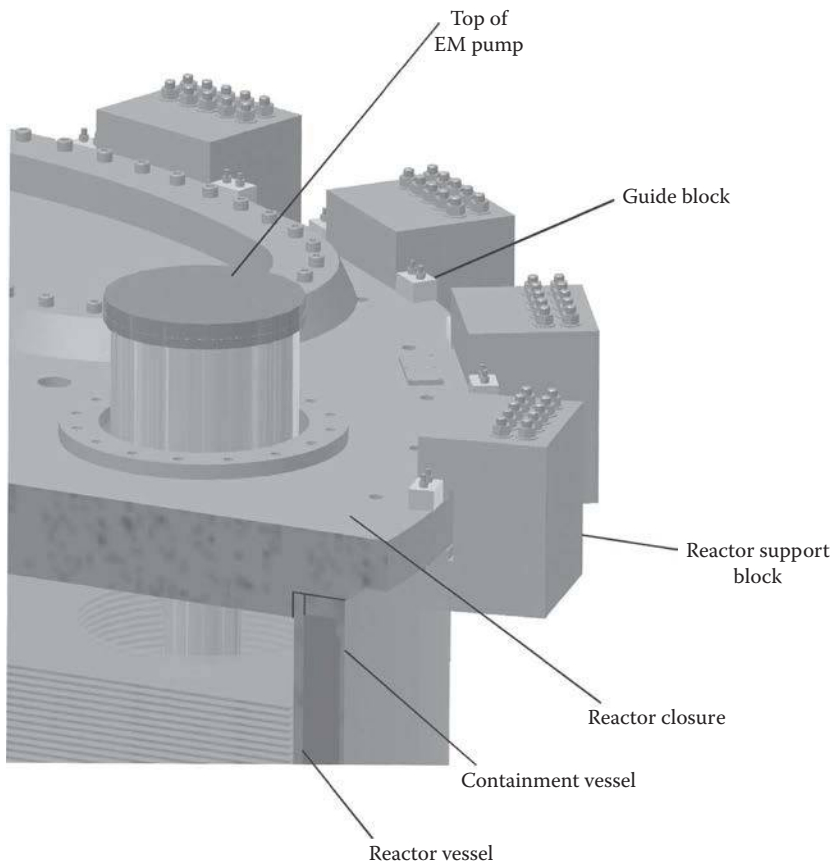


**FIGURE 6.6**  
PRISM reactor closure penetrations.

The reactor core support consists of a redundant beam structure on the bottom and the sides of the reactor vessel. A core barrel and support cylinder extends from the inlet plenum to above the core to contain the core and storage racks for temporary storage of off-loaded fuel assemblies. The internal core structure is shown in Figure 6.8.

The reactor vessel design is based on proven historic SFR technology. The most important new feature of the PRISM reactor vessel and internals is that the reactor vessel has no penetrations (below the reactor closure head). This reactor vessel nozzle configuration precludes any large pipe ruptures at or below the elevation of the core. It is a key factor of the PRISM safety systems to keep the core completely and continuously flooded for the entire spectrum of design basis events/accidents. The reactor vessel is filled with liquid sodium and a helium cover gas.

The reactor vessel is the container and support structure for the reactor core, the primary sodium, and the internal structures. These components provide shielding, flow passages, thermal barriers, sealing and restraint surfaces, and load transfer paths. The reactor vessel performs its support and containment functions during the temperature, pressure, and load variations which occur during its operating lifetime. The reactor



**FIGURE 6.7**  
Closure-containment vessel connection and support.

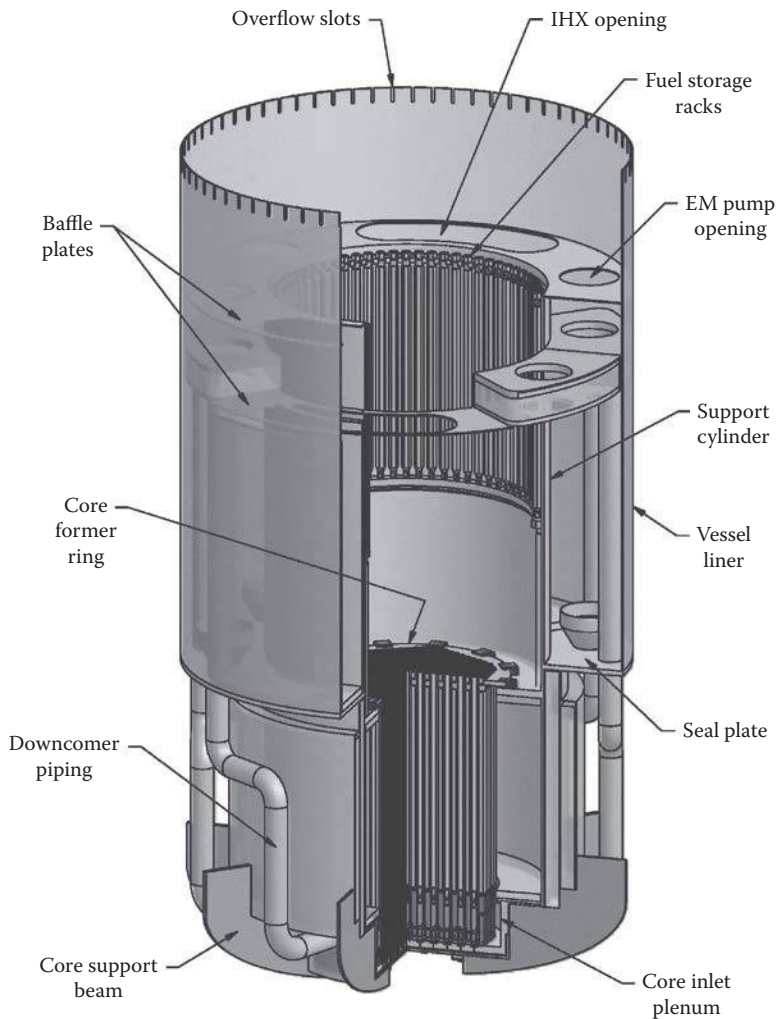
vessel, which forms part of the primary system boundary, is designed and constructed to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB and Case N-47.

The bottom head of the reactor vessel is hemiellipsoidal. The upper head is the reactor closure. The only attachments to the reactor vessel are to those of the core support structure.

Two IHXs and four EM primary pumps are suspended in the reactor vessel by the reactor closure. Primary control rod drives, ultimate shutdown rod drives, in-vessel instrumentation, and an in-vessel transfer machine (IVTM) for fuel handling are suspended from the rotatable plug in the closure [3]. The reactor closure with major equipment is shown in Figure 6.9.

#### 6.4.3.1 Intermediate Heat Transport System

The intermediate heat transport system (IHTS) is located within the reactor vessel. The IHTS consists of piping and components required to transport the reactor heat from the primary system, through the IHX, to the SG system (SGS). The IHTS is shown schematically in Figure 6.10. The IHTS is a closed loop system with an expansion plenum in the SG top head and an argon cover gas space to accommodate thermally induced system volume changes.



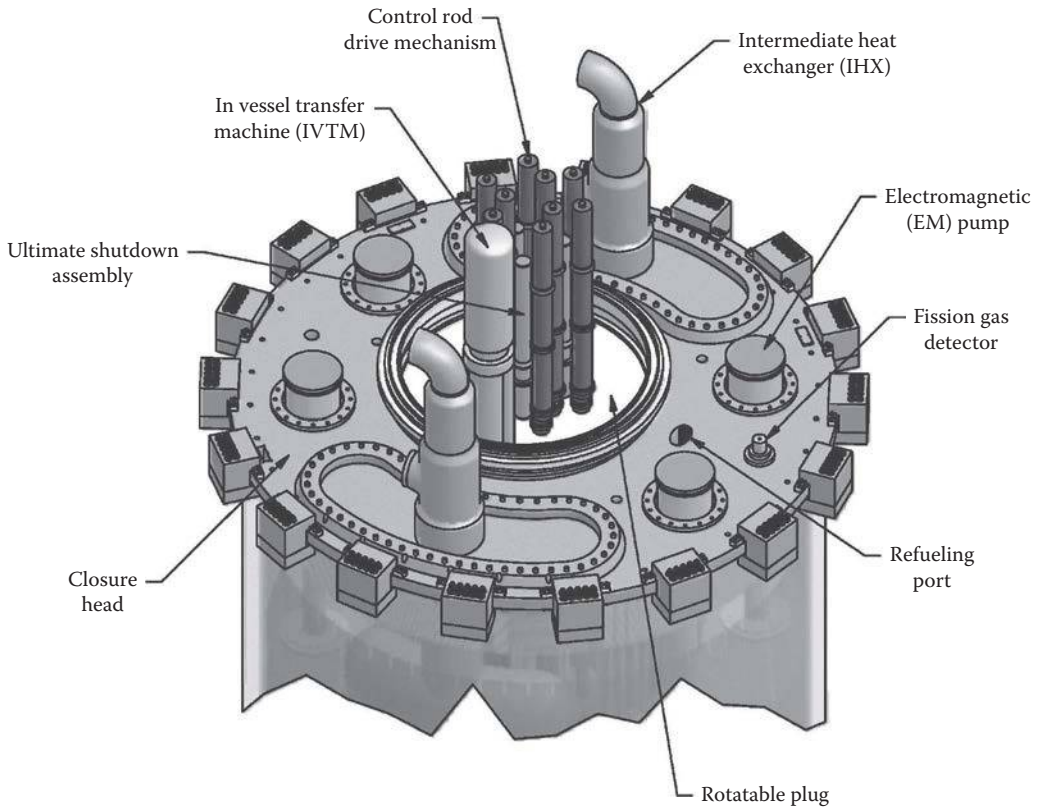
**FIGURE 6.8**  
Reactor vessel internals.

Intermediate sodium is circulated through the shell-side of the IHX and the shell-side of the SG by two EM pumps, each located in the cold leg of the loop in the SG facility. The internal EM pumps—pumps with no moving parts that move conductive fluids by way of a magnetic field—circulate the molten sodium through the reactor core and then to the IHTS. Permanent magnet flow meters are located in the cold leg to monitor sodium flow in the loop.

The IHTS main sodium piping, the SG shell, and the piping up to and including the SGS water-side isolation valves are a part of the containment boundary (see Figures 6.3 and 6.4).

The arrangement and relative elevation of the IHTS piping and components, shown in Figure 6.5, are designed to promote natural circulation for decay heat removal.

Hot leg sodium is transported in separate stainless steel pipes from the two IHXs to the SG top head through vertical inlet nozzles. The cold leg piping returns the sodium to the IHX. Two EM pumps, located in the SG facility, transport the sodium through the IHTS loops. The pump design is similar to the primary system pumps.



**FIGURE 6.9**  
Reactor closure with major equipment.

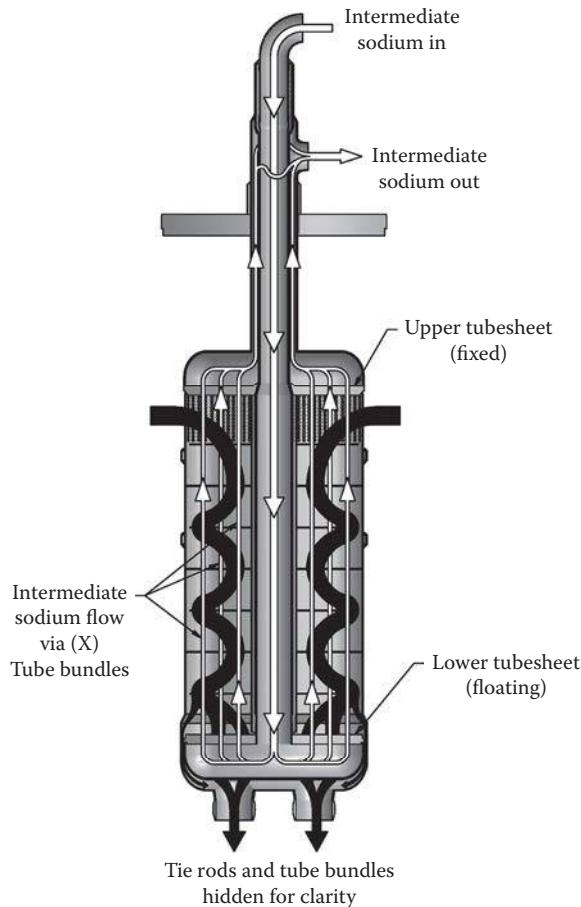
A sodium leak detection system provides early warning of any sodium-to-air leaks from the IHTS piping. In the event of an SG tube leak, the sodium-water reaction pressure relief subsystem (SWRPRS) provides overpressure protection of the IHTS and IHXs. The SWRPRS consists of a safety-grade rupture disk, a sodium dump tank, a cyclone/separator tank, a vent stack, and a hydrogen igniter. To separate the reactants, the SWRPRS also initiates the water-side isolation of the SGS and pressure relief.

#### 6.4.3.2 Primary EM Pumps

PRISM has four submersed EM pumps to provide primary sodium circulation within the reactor vessel. The pumps are inserted through penetrations in the reactor closure. These pumps have no moving parts and no external shaft penetrations. The pumps draw cold pool sodium from an inlet plenum beneath the pump. Within the pump, the sodium enters the tapered inlet section of the pump duct. The sodium is discharged from the top of the pump and is passed radially outward into the pump outlet plenum from which it flows to the core inlet plenum.

#### 6.4.3.3 Reactor Control Rod System

Reactivity control for normal startup operation, load following, and shutdown is accomplished with control rods consisting of  $B_4C$ . Each control unit consists of a drive mechanism, a driveline, and a control assembly. A stepping motor, controlled by the plant control



**FIGURE 6.10**  
Intermediate heat transport system.

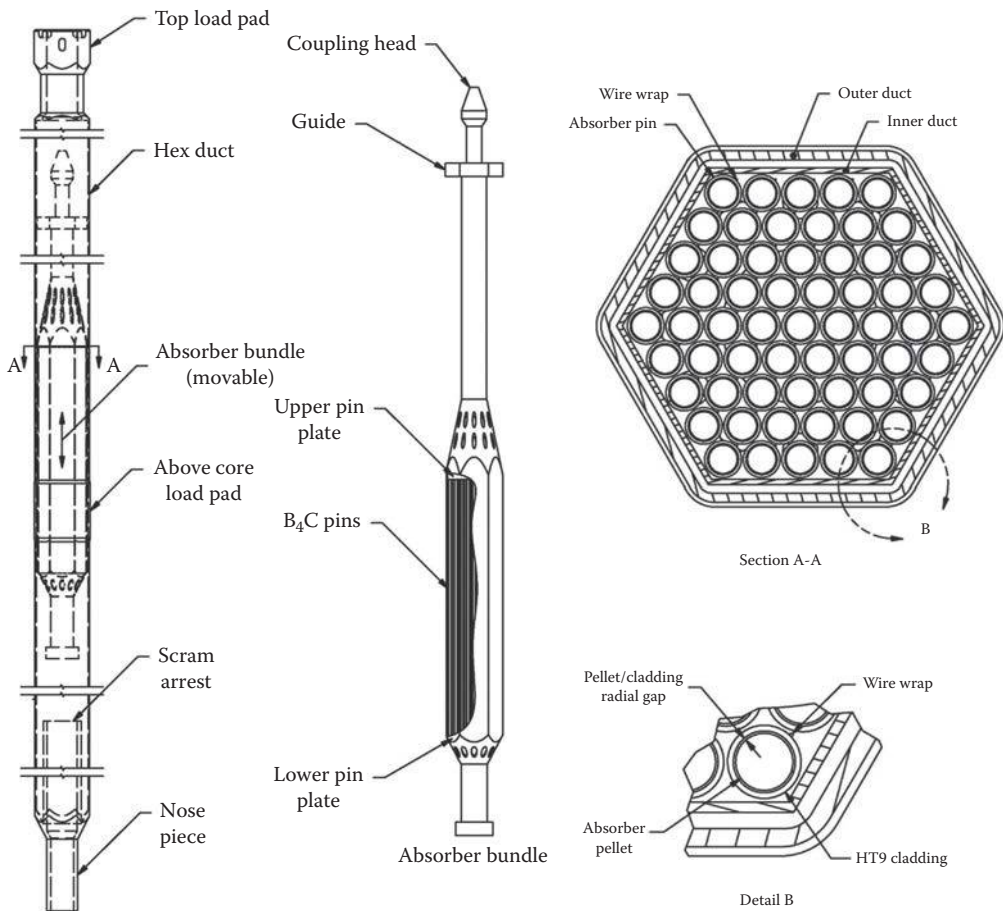
system (PCS), actuates a lead screw to insert and withdraw the control rod. The nine control rods have scram diversity and shutdown redundancy. A schematic of a control assembly is shown in Figure 6.11.

The primary shutdown system is backed up by an ultimate shutdown system (USS). These control rods use magnetic latches, which can be actuated by either the reactor protection system (RPS) or automatically when the latch temperature exceeds the magnetic curie point temperature of the latch.

#### 6.4.4 Safety Systems

##### 6.4.4.1 Reactor Vessel Auxiliary Cooling System

A feature of the PRISM reactor is the reactor vessel auxiliary cooling system (RVACS). The RVACS provides ultimate passive cooling for the reactor if all other methods are compromised or otherwise unavailable. Based on natural circulation, the RVACS is designed to maintain reactor temperature well below design limits for all postulated scenarios. The natural circulation cooling is always operating and removing heat from the vessel and accounts for approximately 0.5 MWt heat loss during normal operation.



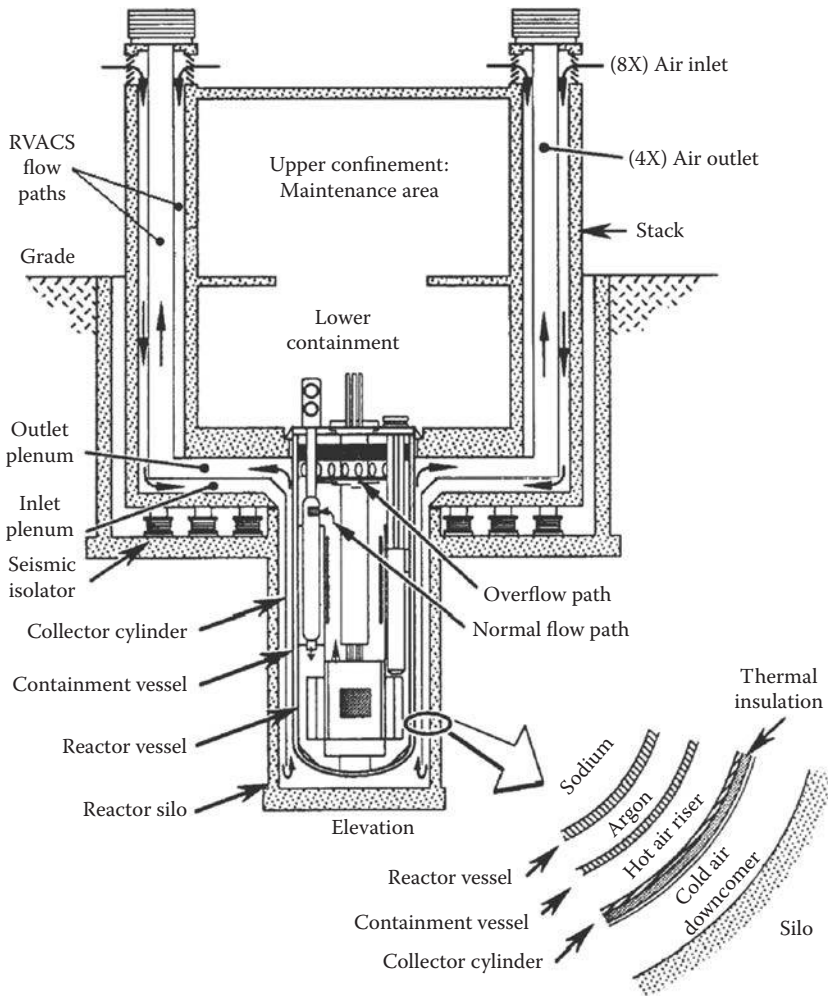
**FIGURE 6.11**  
Control assembly.

The RVACS uses natural air convection to remove heat from the reactor module. Atmospheric air is drawn into the reactor building and flows over the outside of the containment vessel. Radiative heat transfer drives cooling from the reactor vessel, through containment, and to the naturally circulating atmospheric air. Warm air then returns to a stack and is exhausted using the environment as the ultimate heat sink. The RVACS layout is shown in Figure 6.12.

Flow paths for primary sodium and RVACS air are illustrated in Figure 6.12. Initially, primary sodium flows through the IHX as in normal operation until thermal expansion raises the level above flow slots in the reactor vessel liner. At this point, sodium flows along the exterior of the reactor vessel behind the liner, further convectively cooling the thermal mass. Flow paths follow natural circulation patterns, and thus are inherently safe, governed only by fundamental physical behaviors.

In the highly unlikely event that the IHTS becomes completely unavailable, the safety-related RVACS will passively remove decay heat from the reactor vessel. As the temperature of the reactor sodium and reactor vessel automatically rise, the radiant heat transfer across the argon gap to the containment vessel increases to accommodate the heat load. With the increase in containment vessel temperature, the heat transfer from the containment vessel to the atmospheric air surrounding the containment vessel increases. The RVACS system



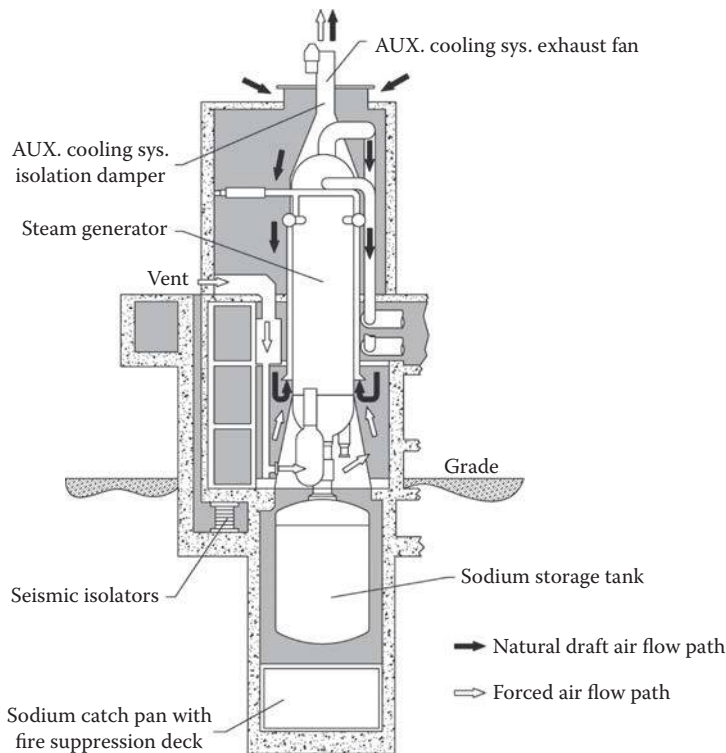


**FIGURE 6.12**  
Reactor vessel auxiliary cooling system.

protects the reactor containment vessel from incurring significant damage during severe transients such as the loss of heat sink.

**6.4.4.2 Auxiliary Cooling System**

During reactor shutdown, heat is removed by the turbine condenser using the turbine bypass. An auxiliary cooling system (ACS) is provided as an alternative method for shutdown decay heat removal during maintenance or repair operations. The ACS removes excess heat from the surface of the SG. The ACS uses natural or forced circulation of atmospheric air to remove heat as the air flows pass the shell side of the SG. The ACS consists of an insulated shroud around the SG shell with an air intake at the bottom through the annulus and an isolation damper located above the SG building. Natural circulation is initiated by opening the exhaust damper [12]. ACS operation in a natural circulation mode, along with RVACS, has the capability to maintain reactor temperatures well below design limits. The ACS is shown in Figure 6.13.



**FIGURE 6.13**  
Auxiliary cooling system.

To increase the heat removal rate, an auxiliary fan located in the exhaust stack may be activated. The fan is also started if augmented decay heat removal capacity is required to reduce system temperatures for maintenance activities or to benefit shutdown margin.

#### 6.4.4.3 Seismic Isolation Systems

Seismic isolation, also shown in Figure 6.12, was incorporated into the PRISM design to enhance safety margins and to allow application of a standard nuclear island design sites of varying soil and seismic characteristics. The entire NSSS and all safety systems are located on a common platform that is supported by seismic isolation bearings. The seismic isolation bearings provide a horizontally flexible element between the base and the nuclear island platform to absorb seismic shocks by allowing the superstructure to displace rather than remain rigid. Isolator specifics can be tailored to site conditions based on local regulations and seismic history to further improve safety [13,14].

#### 6.4.4.4 Reactor Protection System

PRISM offers multiple layers of defense against events that challenge plant safety. The PCS provides the normal plant operation and control function and also prevents challenges to the RPS by taking action before RPS setpoints are reached. However, if RPS setpoints

are reached, the RPS scrams the reactor and ensures that reactor safety limits are met with significant margin. Reactor inherency features (along with the auxiliary safety systems) provide an additional layer of defense and assure that in the unlikely event that the RPS fails to scram the reactor, the reactor remains in a safe state with no radioactive release to the environment.

The primary purpose of the RPS is to automatically scram the reactor if safety setpoints are exceeded. The RPS performs this function by unlatching the control rods for a fast gravitational insertion of all nine control rods. To ensure rod insertion, the RPS simultaneously activates the rod drive-in motors which drive in the control elements. Upon positive indication of control element insertion (decreasing flux), the RPS trips the primary pumps as part of the scram sequence.

The RPS also monitors safety-related reactor parameters to ensure that safe shutdown has been achieved.

#### **6.4.4.5 Reactivity Control**

PRISM has multiple and diverse means for reactivity control and shutdown. Each PRISM reactor has nine control rods (Section 6.4.5.4.4), controlled by a triply redundant reactivity controller, which is part of a highly reliable PCS. The reactor can be reliably shutdown by PCS-directed rod run-in. In addition, PRISM has an RPS (Section 6.4.4.4) which is totally separate from the PCS and provides two diverse means for scrambling each of the nine control rods.

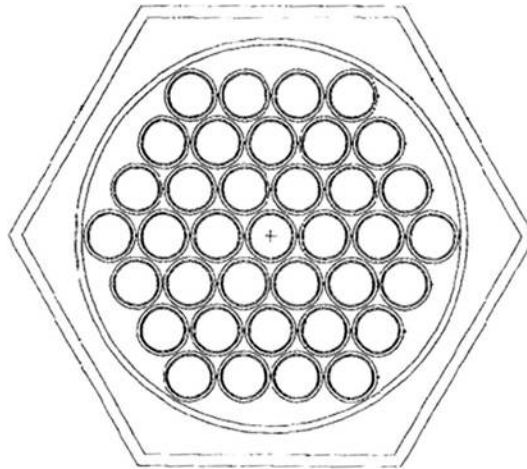
The nine control rods each with three insertion means, two being diverse (PCS shim motor run-in, RPS gravity scram, and RPS drive-in motor scram), provide a high level of redundancy and diversity for reactor shutdown.

An electronically positioned mechanical rod stop system (RSS) prevents unprotected control rod withdrawal and excessive reactivity insertion. Components in the RSS include a redundant controller, a rod stop drive selector, and a limited capacity power supply which controls power to the rod stop adjustment drive motor for each control rod.

PRISM also has strong negative reactivity feedback with rising temperature that makes the reactor capable of passively withstanding severe undercooling and overpower accidents without scram. Gas expansion modules (GEMs) (Section 6.4.5.4.5) enhance the negative reactivity feedback during a loss of flow without scram event. The RSS limits reactivity addition during a rod withdrawal without scram event. The manually actuated USS (Section 6.4.4.6) consists of three devices. During anticipated transients without scram (ATWS) events, passive/inherent reactivity feedback effects of the core bring the reactor to a safe, stable zero fission power state at an elevated temperature until cold shutdown is achieved with the USS. The passive/inherent reactivity feedbacks of the core and the USS provide additional means for reactor control and shutdown beyond the RPS.

#### **6.4.4.6 Ultimate Shutdown System**

The reactor is equipped with three shutdown rods containing pins loaded with enriched B<sub>4</sub>C. The shutdown rods provide a diverse system to bring the reactor to cold shutdown in the event that the control rod system fails to operate. The USS rods are positioned above the active core during operation and remain there until the end of cycle. In the event of the USS actuation (a manual reactor trip), the rods can be inserted in the core by gravity or by a fast run in motor. Insertion of two of the three shutdown assemblies will bring the core to cold shutdown.



**FIGURE 6.14**  
Ultimate shutdown system rod cross section.

The shutdown absorber bundle is similar to the control assembly bundle except fewer absorber rods are used to create a circular configuration as shown in Figure 6.14. The shutdown assembly outer duct is hexagonal, having the same external dimensions as the control assembly duct except for the nosepiece which precludes installation in the wrong position in the core. The absorber assembly is always within the duct during operation above the active core. Minimum cooling is required in the absorber, as it is above the core. During insertion, fluid flows past the falling absorber assembly easily due to the large flow area.

#### **6.4.4.7 Thermal Shutoff System**

The thermal shutoff system (TSS), separate from the RPS, is designed to automatically shut off the EM pumps on high sodium temperature, in case the heat sink (IHTS) is lost and the RPS fails to scram. For this scenario it is assumed that the loss of heat sink event starts from full power and that neither the PCS nor the RPS can shut down the reactor so that the reactor is left with only RVACS to remove heat.

The temperature within the reactor rises rapidly, which triggers reactivity feedback mechanisms to bring the reactor to a critical level at an elevated equilibrium temperature, if the EM pumps are not operating. However, if the EM pumps continue to operate, each EM pump adds heat to the reactor. If the EM pumps are not shut off, the reactor's integrated heat input could exceed the heat removal capability of RVACS at structurally acceptable temperatures. Thus, there is a need for a mechanism to automatically turn the EM pumps off should the reactor temperature exceed a given threshold.

The RPS provides the control for the RPS breakers in the EM pump/synchronous machine power circuit. As part of a normal scram sequence, the RPS will release these breakers to initiate an EM pump coast down of the primary flow.

#### **6.4.4.8 Inherent Negative Reactivity Control**

The PRISM shutdown systems are backed up by the inherent negative power reactivity feedback of the reactor core. This inherently negative reactivity feedback brings the core to a safe, stable, power state following accidents.

The moderator expansion characteristic of LWRs is not present in the fast spectrum sodium-cooled reactor; yet negative reactivity feedback due to thermal expansion prevails. Doppler feedback is generally the fastest acting feedback mechanism since it is almost instantly affected by core power level. The Doppler feedback inserts negative reactivity into the system as the temperature rises and can thus help limit the extent of power increases. However, as the fuel temperature drops with a power reduction, the Doppler feedback adds reactivity which tends to limit the power decrease of the system.

Thermal expansion of the sodium results in a net positive reactivity feedback. Thermal expansion results in fewer sodium atoms within and surrounding the core. The reduced density surrounding the core results in fewer neutrons being scattered back into the core and produces a small negative feedback effect by increasing the leakage around the periphery. However, the dominant effect is the reduction of collisions between neutrons and sodium atoms, which hardens the neutron energy spectrum and yields a net positive reactivity feedback effect.

Volumetric expansion of all aspects of the core inherently provides negative reactivity feedback. As the temperature of the sodium increases, the core structure thermal expands resulting in a larger core surface area and greater neutron leakage. This causes a reduction in core reactivity, which is enough to bring the core to a safe, stable, power state following accidents.

PRISM has this capability for accommodating the following severe, but extremely unlikely, accidents:

1. Inadvertent withdrawal of all control rods without automatic scram or manual actuation (unprotected transient overpower)
2. Loss of all cooling by the IHTS without scram (unprotected loss of cooling)
3. Loss of primary pump power and loss of all cooling by the intermediate heat transport loop without scram (unprotected loss of flow/loss of cooling)

These accident scenarios were evaluated experimentally by ANL in EBR-II [15–18] and at the TREAT facility [19]. These tests empirically demonstrated the safety of a pool-type, metal-fueled SFR like PRISM.

The inherent negative reactivity feedback response of the core to temperature increases and the USS can also shut down the reactor, or, in the case of the negative reactivity feedbacks, at least bring the core to a safe, and stable, zero fission power state. Owing to the negative reactivity feedbacks, PRISM has a unique capability for accommodating severe, but extremely unlikely, accidents with very benign consequences. The three postulated, beyond design basis scenarios listed above illustrate this.

For the transient overpower event without scram, it is assumed that all nine rods would be withdrawn from their normal full power position at the maximum withdrawal rate. Initially, there would be a rapid rise in power, which is halted by negative reactivity feedbacks from radial core expansion, fuel assembly bowing, and Doppler feedback associated with rise in sodium and fuel temperatures. The PRISM core and fuel can safely accommodate this transient without melting of fuel or sodium boiling and void formation in the cooling channels; that is, the consequences are benign.

For the loss of flow/loss of cooling event without scram, the increased core and bulk sodium temperatures would result in negative reactivity feedbacks from the thermal expansion of the above-core load pad (ACLP), the radial thermal expansion of the core support grid plate, and core assembly bowing, resulting in a power reduction to near zero fission power. Decay heat is removed by the passive air natural circulation heat removal

system (RVACS, Section 6.4.4.1). This natural circulation heat removal is sufficient to limit the nominal bulk sodium outlet temperature to within the structural capabilities of the reactor and well below the in-core boiling limit.

All three ATWS events result in a stable condition at temperatures below design limits without any operator action and without any automatic safety system action. This condition can be safely sustained for many days, until operator action is taken to bring the reactor to a cold shutdown condition. Since it is assumed that in such an event, both the PCS and the RPS have failed and cannot be reactivated, the manually activated USS would bring the reactor to a cold shutdown.

## 6.4.5 Auxiliary Systems

### 6.4.5.1 SG System

The nuclear heat supply system generates heat in the reactor core and produces steam in the IHTS. The IHTS SG is composed of a helical coil counterflow heat exchanger, startup recirculation tank and pump, leak detection system, and isolation valves. Intermediate sodium flows on the shell side of the helical coil tubes, through which water flows to produce superheated steam. Two reactor SGs feed one turbine island to form a power block.

SG design leverages experience from the ALMR program, where a 76 MWt prototype SG was fabricated and tested at the DOE Energy Technology Engineering Center for 4 years. Based on this development work, testing, and GEH trade studies, this design was selected for the PRISM SG design. Inherent safety and resistance to sodium–water interaction made this design most fitting for the PRISM application.

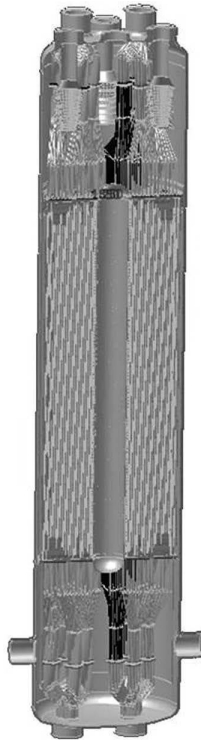
The design configuration of the once-through superheated steam helical coil SG is a vertically oriented helical coil sodium-to-water counterflow shell-and-tube heat exchanger with four sodium inlet nozzles in the upper head and a single sodium outlet nozzle on the lower head. The SG shell (Figure 6.15) encloses the steam generating helical coil tube bundle and support system, feedwater inlet tube assembly, steam outlet tube assembly, and sodium inlet distribution assembly. A small cover gas space is contained in the expansion cap attached to the upper head, with provision for sodium overflow and make-up to accommodate changes due to sodium expansion.

Feedwater enters the SG through four axial inlet nozzles and tube sheets in the bottom head and is heated as it flows upward through the inlet tubes, the helical coil tube bundle, and the outlet tubes connecting the tube bundle to four outlet tube sheets located axially in the top head. Flow restrictors ensure static and dynamic flow stability over a wide load range.

Hot sodium enters the SG vessel through four, vertical, sodium inlet nozzles on the upper head into four distributors. The sodium flows past the partially shielded steam outlet tubes and into the free flow area around the helical coil tube bundle. The colder sodium leaves the bottom of the tube bundle, then flows through a single sodium outlet nozzle located in the center of the lower ellipsoidal head, beyond which the two cold leg pipes attach laterally; and still downstream is the SWRPRS.

Cover gas in the SG expansion cap mitigates the pressure transients during large sodium–water-reaction (SWR) events. The SWRPRS is connected to the exit nozzle at the center of the lower ellipsoidal head.

The SG is cooled by circulating air over the exterior shell surface to provide an alternative method of reactor decay heat removal. The ACS supplements the RVACS and reduces the time required to cool the reactor system and IHTS to hot standby temperature following



**FIGURE 6.15**  
Steam generator shell.

a reactor shutdown when the steam system is not available. The ACS is used to increase plant availability by avoiding the slow plant cooldown which would occur if only the RVACS were used. The ACS consists of a steel shell enclosing the SG, forming a 6-in. wide air cooling annulus. A fan and backdraft dampers are provided in the air cooling circuit.

#### 6.4.5.1.1 SG Isolation System

In a sodium-heated SG, the hot sodium from the in-vessel heat exchangers heats the SG tube bundle on the shell side. The heat transferred to the tube side heats up and vaporizes water provided by the feedwater system. In the event that water and sodium are brought into contact with one another, as through a leak in a water tube, a rapid chemical reaction can take place with the release of hydrogen gas and heat. The reaction, if significant, can destroy additional tubes and this can result in a large economic penalty. Thus, it is important to separate the chemically reacting components as soon as detected.

The hydrogen gas from the chemical reaction increases the pressure within the vessel. When the pressure within the SG reaches a predetermined higher level, the rupture disks burst and the sodium within the SG drains rapidly into a sodium dump tank. For such large sodium/water reaction events, it is also necessary to seal off the reactor/SG system for the purpose of containment isolation. An SG isolation system (SGIS) is provided to terminate a sodium/water reaction (by rapidly removing the water from the SG) and to isolate the water side with safety-related isolation valves which serve a safety-grade barrier for the purpose of containment isolation.

The SGIS separates the chemically reacting components by isolating the water side of the SG and rapidly expelling water from the tubes by opening water dump and nitrogen purge gas valves. If the leak is detected quickly (through the integrated leak detection system) the SGIS can trigger isolation without rupture disk activation. However, if the safety-grade rupture disk does activate, the system is designed to automatically isolate the water side with the safety-grade isolation valves. Sensing wires on the back of each sodium rupture disk are broken by the tearing of the disk during actuation, and the breaking of these wires activates the SGIS automatically.

#### 6.4.5.2 Primary Sodium ACS

PRISM includes a nonsafety-related auxiliary decay heat removal system referred to as the primary sodium ACS (PSACS) for use during (1) IHTS and SG maintenance operations and (2) in case of an inadvertent IHTS loop drain, or a drain performed because of a steam to sodium leak. By adding this system, plant downtime as a result of scheduled SG inspections and IHTS maintenance operations can be reduced.

The PSACS uses the primary sodium cleanup lines and pump to reject heat via an Na-to-N<sub>2</sub> heat exchanger coupled to the same N<sub>2</sub> loop used for the primary sodium cold trap. The N<sub>2</sub> loop rejects heat to an air dump heat exchanger. Electrical power is needed for this system to actuate valves, and run fans and pumps. The PSACS has a minimal impact on plant cost since it shares equipment with the sodium cleanup system.

#### 6.4.5.3 Core Configuration

The PRISM core has a heterogeneous layout of fuel, blanket, control, reflector, and shield assemblies. Three hundred and ninety-one core assemblies are broken down into 192 fuel assemblies in two TRU enrichment zones, 114 reflector assemblies, 66 shield assemblies, 6 GEMs, 9 control assemblies, and 3 ultimate shutdown assemblies. The baseline core configuration is shown in Figure 6.16.

The fuel composition and assembly configuration varies based on the core mission. Metallic fuel is fabricated on-site in a devoted fuel cycle facility which varies for fuel missions. Possible core missions include low conversion ratio (UNF actinide recycle), unity conversion ratio (breakeven), high conversion ratio (breeder), or legacy fissile material consumption. A change in core mission requires changes to plant orificing and to fuel design and configuration. The power level of the associated cores is primarily limited by the shutdown heat removal capabilities of the passive safety systems. A comparison of the various number of assemblies for each mission is given in Table 6.2.

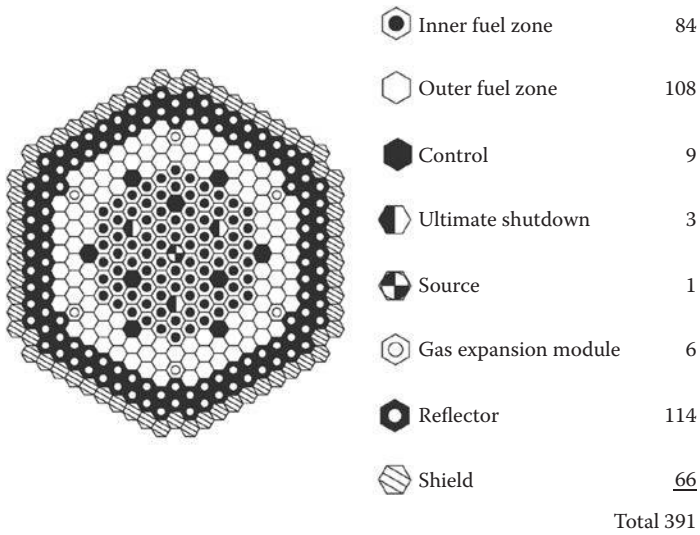
Electrometallurgical separation (also known as pyroprocessing) distills away any volatile elements and leaves behind high-purity fuel material. A typical fuel rod is pictured in Figure 6.17 with indication of active fuel region, bottom reflector, and top plenum.

For the UNF recycling core mission the fuel is comprised of a uranium–TRU–zirconium alloy (U-TRU-Zr) with two fuel zones. The assembly configuration for this core mission is shown in Figure 6.18.

The fuel alloys are assumed to expand axially due to initial fission gas generation. This expansion is accommodated in the core reactivity design, and its impact on nuclear performance parameters is explicitly included.

The reactor core is designed to operate at rated power without any limitations, while delivering the total cycle length and energy desired by the utility. These design goals are

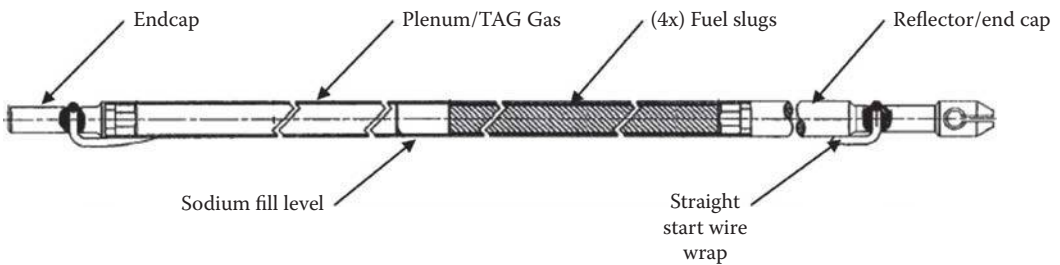




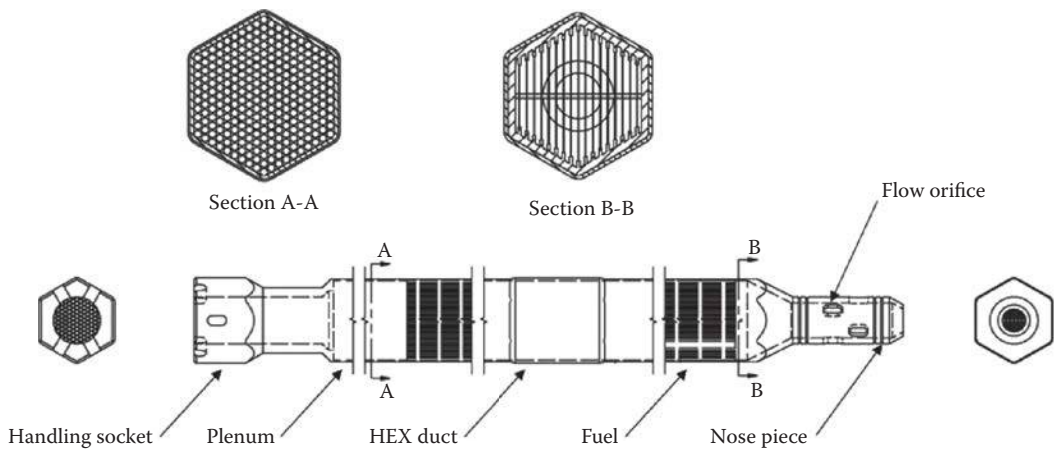
**FIGURE 6.16**  
Core configuration.

**TABLE 6.2**  
PRISM Assembly Count

Assembly Type	Core Configuration			
	UNF Recycle	Breakeven	Breeder	Weapons Pu Consumption
Driver fuel (inner + outer)	192	108	108	108
Internal blanket	–	42	42	42
Radial blanket	–	42	42	42
Control	9	9	9	9
Ultimate shutdown	3	3	3	3
Neutron source	1	1	1	1
Gas expansion module	6	6	6	6
Reflector	114	114	114	114
Shield	66	66	66	66
<b>Total</b>	<b>391</b>	<b>391</b>	<b>391</b>	<b>391</b>



**FIGURE 6.17**  
PRISM fuel rod.



**FIGURE 6.18**  
PRISM fuel assembly.

achieved by designing with sufficient margin to thermal and reactivity limits to accommodate the types of uncertainties encountered in actual operation. Based on its extensive experience in BWR and SFR core design, GEH has developed a consistent set of design margins to ensure meeting these objectives without compromising overall efficiency due to the use of undue conservatism.

#### 6.4.5.4 Core Assembly Descriptions

The design of the core assemblies maximizes the use of common structural components, with the exception of the assembly internals.

##### 6.4.5.4.1 Fuel

PRISM uses a metallic fuel for compatibility with the coolant, its desirable thermomechanical properties, and ease of fabrication [20,21]. Metal fuel bundles allow a higher heavy metal fraction in the fuel resulting in a lower fissile enrichment and better internal transmutation compared to oxide fuel. Thus, the metal fuel core is capable of satisfying nuclear goals with fewer fuel assemblies and a more compact core.

Fuel rod cladding is stainless steel alloy HT-9, as well as much of the internal core structure [22]. HT-9 is favored in fast neutron applications for its neutron transparency, resistance to radiation damage, and thermal performance.

PRISM fuel rods are bundled together into a hexagonal lattice. A fuel assembly schematic is shown in Figure 6.18. Rod spacing is maintained by a spiral wire wrap around each rod. Each fuel assembly includes a hexagonal duct around the fuel pins to maintain coolant flow around the pins. After the entire fuel cycle, spent assemblies decay for one fuel cycle in in-vessel storage.

##### 6.4.5.4.2 Reflector

One hundred and fourteen reflector assemblies are located around the core periphery to decrease fast neutron leakage and promote flux in peripheral bundles.

#### 6.4.5.4.3 Shield

Sixty-six shield assemblies are located about the core to absorb neutrons and prevent excessive irradiation damage.

#### 6.4.5.4.4 Control

Reactivity control for normal operations of startup, load following, and shutdown is accomplished by bank (uniform) movement of nine control assemblies in the fuel region of the core (see Section 6.4.4.5). While the design requirement is for shutdown with up to two stuck rods, the worth of the control rods is such that any four of the nine assemblies can shutdown the reactor to cold subcritical conditions. Each assembly can be inserted into the core in three different ways: a rod run-in by the PCS (a fast run-back), a fast run-in scram initiated by the RPS, and a gravity drop scram also initiated by the protection system.

Three ultimate shutdown assemblies constitute the USS, described in Section 6.4.4.6. In the event of a complete failure of the normal scram system, the USS along with inherent negative reactivity feedback brings the core to a safe critical state at elevated temperature. The USS employs sliding absorber rod bundles suspended above the core during normal operation. In an emergency situation, USS assemblies are manually actuated and insert using gravity.

#### 6.4.5.4.5 Gas Expansion Module

For safety margin in the event of loss of primary coolant flow, six GEMs are included at the periphery of the active core. A GEM, which is of the same external size and configuration as the other core assemblies, consists of a hollow assembly duct which is open to flow at the bottom but is closed to flow at the top. The GEMs are filled with vessel cover gas before insertion into the core, and this gas is compressed as the GEMs are filled with sodium. Each GEM communicates with the inlet plenum through an opening in the nosepiece. With the primary pumps on, the high pressure in the inlet plenum compresses the gas captured in the GEMs and raises the sodium level in the GEMs to a region above the active core. When pumping power is lost in the primary system and the pressure drops, the gas expands, displacing the sodium in the GEMs to a level below the active core. The resultant void near the core periphery increases neutron leakage and introduces significant negative reactivity which limits the peak temperatures attained during the loss of flow events.

### 6.4.5.5 Reactor Refueling

Refueling and reloaded cores are fabricated in the associated fuel service facility, with stock material from recycled PRISM and LWR fuel. Minor actinides in LWR reactor spent fuel are included to be fissioned in the fast neutron spectrum. The actinides largely fission as part of the burnable fuel, resulting in energy production and elimination of long-lived waste products. Through multiple recycling iterations, minor actinides are fissioned in the PRISM fuel; the resulting ultimate waste stream is composed of fission products whose radiotoxicity is on the order of hundreds of years, as opposed to hundreds of thousands for long-lived actinide waste.

#### 6.4.5.5.1 Reactor Refueling System

The reactor refueling system (RRS) provides the means of receiving, storing, transporting, and handling reactor core assemblies. Fuel, control, shield, reflector, and ultimate shutdown assemblies are handled by the RRS for all core configuration operations.

Reactor refueling occurs between 12 and 24 month intervals, depending on the fuel cycle operation and mission. A transfer cask is used to exchange spent fuel and other core assemblies in the reactor with new assemblies from the fuel service facility.

Multiple assembly storage locations are utilized to facilitate fuel cycle operations. In-vessel storage is provided to allow decay and cooling prior to transfer to the fuel service facility. Within the reactor vessel, assemblies are moved between the core, storage racks, and a transfer station below the transfer port by the IVTM. The IVTM provides the means for moving the core assemblies between positions within the reactor. The IVTM is an integral component of the reactor module.

#### **6.4.6 Sodium as Coolant**

Sodium is a chemical that has been used extensively and has an established record of safe handling on a large scale. The PRISM reactor has been designed with the potential for sodium–water reaction in mind. The primary sodium system is entirely encompassed within the reactor vessel and surrounded by the containment vessel, to eliminate the potential for loss of primary coolant and to reduce the potential for sodium–water reaction. The IHTS transports heat from the primary system to the SG where the heat is transferred to the cooling water. The sodium–water interaction is mitigated in the SG with the pressure cap inert gas and the SGIS as described in Section 6.4.5.1.

---

### **6.5 PRISM Fuel Cycle**

#### **6.5.1 Closing the Fuel Cycle**

PRISM is a technology that can close the nuclear fuel cycle using the energy contained in spent nuclear fuel. PRISM can generate stable base load electricity to help meet growing electricity needs and enhance energy security. This technology can also reduce the need for additional geologic storage capacity.

#### **6.5.2 Fuel Fabrication**

The PRISM NFRC uses an established electrometallurgical separations process employed by the aluminum industry for separating aluminum from alumina. This electrometallurgical process has been proven and deployed on a laboratory and engineering scale for nuclear materials by both ANL and Idaho National Laboratory [23]. Separations are accomplished in a molten salt bath that uses an electrical potential in the solution to separate used fuel. The uranium and TRUs plate out on an anode and are subsequently removed from the solution.

With the NFRC process, nonproliferation concerns are alleviated by the intrinsic diversion resistance of the electrometallurgical process [11,24]. All TRUs (Np, Pu, Am, Cm) are separated en masse in one step of the electrometallurgical process; the combined constituents are not usable for military purposes without complex additional processes. These TRUs are continually recycled in the PRISM until they are fissioned completely [25,26], which localizes the material on-site thereby avoiding risks of theft or diversion during transport.

### 6.5.3 Waste Products

The resulting fission product waste requires a few hundred years before returning to a radio-toxicity similar to that of the uranium ore originally removed from the ground during mining, thereby closing the fuel cycle. In comparison, tens-of-thousands of years are required for mixed oxide reprocessed UNF to return to these same levels. The resulting metallic and ceramic waste forms from the PRISM process are significantly more resistant to corrosion and are less susceptible to leaching into the environment than untreated UNF [23].

---

## 6.6 PRISM Technology for the Future

LWR spent nuclear fuel is an asset to be managed using fast reactor technology. PRISM technology is synergistic with LWRs in that they consume unwanted TRUs produced by LWRs to expand the resource of nuclear power. During that consumption, electricity is produced.

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# 7

## *MSR Technology Basics*

David LeBlanc

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## 7.1 MSR Technology Basics

A molten salt reactor (MSR) is a reactor in which fluorides of fissile and fertile elements such as  $UF_4$ ,  $PuF_3$ , and/or  $ThF_4$  are combined with carrier salts to form a fluid fuel. MSRs can operate as simple burner reactors with high fuel economy or with the addition of online fission product removal and can achieve breeder status. Typical operation sees molten salt flowing between a critical core and an intermediate heat exchanger. A secondary coolant salt then transfers heat to a steam or closed gas cycle. The majority of work has involved fluoride salts, as corrosion-resistant alloys have been shown to be compatible with these salts.

Chloride-based salts have also been proposed—for fast spectrum MSR designs. Due to the large neutron absorption cross section of  $^{35}Cl$ , most early work assumed enriched  $^{37}Cl$  (Ottewitte, 1982; Smith et al., 1974). More recently, natural chlorine has been shown to be able to support breeding (Mourogov and Bokov, 2005), but the usage of natural chlorine will lead to very significant production of  $^{36}Cl$  (half-life 300,000 years). As well, chloride salts have more limited materials compatibility and increased corrosion potential compared to fluorides, especially as corrosive sulfur is produced from neutron interactions with chlorine. Processing of these salts is also a challenge, as separation factors for chlorides are inferior to those for fluorides (Ignatiev, 2003). Thus, while their spectrums, being much faster than the spectrums of fluoride salts give impressive breeding potential, the development path for chloride salts is seen to be significantly more challenging than for fluoride salts.

The most common carrier salts proposed are mixtures of enriched (>99.99%)  $^7LiF$  and  $BeF_2$  termed *FLiBe*. Mixtures with upward of 14%  $ThF_4$  and/or  $UF_4$  have melting points below a traditional limit of 525°C, which gives adequate margin for use with the high nickel alloy Hastelloy N™. This alloy is proposed for use up to 704°C with these molten salts and options exist to employ newer, ASME Section III-qualified materials such as Alloy 800H clad with Hastelloy N™. Furthermore, many forms of steel such as SS304 and SS316 have performed well in loop tests (ORNL-TM-4286, 1972; ORNL-TM-5782, 1977) and although somewhat inferior corrosion resistance to Hastelloy N™ bring many other potential advantages. It has also been suggested that carbon-based structures or refractory metals could be used throughout the primary loop including heat exchangers. This would allow higher peak temperatures and a wider set of carrier salt options with higher melting points. However, such options would require a significant development effort to prove their viability.

Molten reactor designs have been configured within a wide range of concepts. Similar to gas-cooled reactors (pebble bed, prismatic, He,  $CO_2$ , etc.) this variety can be a drawback as it can tend to spread effort and support. Early efforts examined both a breeder approach termed the molten salt breeder reactor (MSBR) and a burner approach termed the denatured MSR (DMSR) while currently there is also renewed interest in fast spectrum designs. The following sections attempt to show the varied potential as well as differentiating factors for a variety of MSR designs.



### 7.1.1 Advantages of MSRs

MSRs have numerous potential advantages over solid fuel reactors. Most benefits relate to the fluid nature of the fuel form. It is this fluid nature which at first may be a point of concern for those solely familiar with solid fuels. The listed benefits below will of course vary between MSR concepts but are generally universal in nature.

#### 7.1.1.1 Safety

MSRs have safety advantages over solid fuel reactors in a number of different dimensions as next outlined:

- The fluid nature of the fuel means that the term meltdown is an irrelevant term. The fluid nature of the fuel, besides obviating meltdown, also greatly aids decay heat removal and enables passive response capability; conventionally by means of a freeze plug that melts upon overheating or loss of active cooling of the plug. Fuel salt and its associated decay heat then drains by gravity to passively cooled, critically safe dump tanks. Alternatively, fluid fuels can be cooled in situ by taking advantage of the fact that the salts will naturally circulate within the reactor with operating temperatures roughly 700°C below the salt's boiling point, the latter providing large margins for operation.
- Fission products either quickly form stable fluorides that will stay within the salt during any leak or accident or are volatile or insoluble and can be continuously removed. For example, cesium forms stable CsF and iodine forms an iodide which is relatively stable in the fuel salt.
- While the fuel is considered mobile, many layers of containment are employed at low-pressure operation. The fuel salt is typically run at a lower pressure than intermediate loops such that any leak is inward.
- The continuous removal of the noble gas xenon means that there is no reactor *dead time* after shutdown or a power decrease—a phenomenon that most solid-fueled reactors must deal with due to the production of  $^{135}\text{Xe}$  from the decay of  $^{135}\text{I}$ . As well, there needs to be no excess reactivity in place to deal with such situations.
- Fissile concentrations are easily adjusted on a continuous basis, thereby avoiding excess reactivity and the need for burnable poisons. A single control rod for shutdown and start-up is often included but is not strictly necessary, given the ability to drain fuel out of the core into critically safe storage tanks.
- MSR designs have strong fast-acting negative temperature reactivity coefficients, aiding safety and allowing automatic load following operation. With some designs, concerns have been raised about slow-acting positive reactivity coefficients from graphite heating. Steps can be taken to assure a global negative temperature coefficient.
- MSRs operate at low pressure with low vapor pressure liquids running 700°C below their boiling point. Only thin-walled vessels and piping are required and any rupture would be an event of low energy. The primary loop of an MSR operates within a hot cell that will automatically collect and direct any rare spill or rupture.
- The absence of water or sodium removes the risk of steam explosion or hydrogen production within the containment. There are simply no internal driving forces able to spread release.

### 7.1.1.2 Cost Innovation

Much of the MSR's potential for cost innovation is also related to its fluid fuel nature, listed as follows:

- The passive and inherent nature in which fluid fuels provide for assured safety results in cost innovation in contrast to solid fuels which require highly engineered safety systems and assurance that in every possible situation, coolant keeps reaching the fuel. With MSRs the fuel is the coolant.
- Molten fluoride salts in general are excellent coolants, with a 25% higher volumetric heat capacity than pressurized water and nearly five times that of liquid sodium. This results in more compact primary loop components like pumps and heat exchangers. They also have a lower thermal conductivity than sodium which avoids thermal shock issues.
- MSRs operate at low pressure, resulting in lightweight and simple-to-construct components such as pipes and reactor vessels. Any penetrations needed of the reactor are similarly at low pressure.
- MSRs also operate at high temperature, which allows far higher thermal efficiencies approaching 50%, with a choice of steam Rankin or various gas Brayton technologies. This factor alone represents a significant source of capital cost reduction.
- MSRs operating as breeders have minimal fueling costs in regard to raw materials; \$10K–\$30K of thorium potentially providing a GWe-year of power. Breeders however require salt processing which will dominate fuel cycle costs. Burner MSRs avoid on-site salt processing and have been shown to be able to be many times more efficient on the use of uranium than conventional light water reactors (LWRs). At the same time, there are no fuel elements to fabricate or to replace. Fuel cycle costs under 0.2 cents/kWh for burner designs are readily feasible.

### 7.1.1.3 Resource Sustainability

Whether designed as breeders or burners, MSRs have outstanding fuel efficiency:

- Fuel processing and utilization of thorium permits break-even breeding with ease and the ability to reach a breeding ratio of 1.06 (ORNL 4541, 1971) or even up to 1.13 (Nuttin, 2002). Adding <sup>238</sup>U to denature the uranium content and still break-even is also possible (ORNL-TM-6413, 1978).
- Break-even operation only requires approximately 800 kg of thorium per GWe-year added simply as ThF<sub>4</sub>. Start-up fissile requirements can be below a tonne per GWe for thermal spectrum designs or upward of 5.5 tonnes in harder spectrum designs. Thorium start-up inventory varies from 50 to 200 tonnes per GWe.
- Without any on-site fuel processing, MSRs can run as simple burners with excellent uranium utilization, even on a once-through cycle requiring only a fraction of the input uranium of conventional LWRs. Since even large increases in the price of uranium are acceptable in terms of fuel cycle costs for an MSR burner, the availability of world uranium resources would not become an issue for many centuries at least.

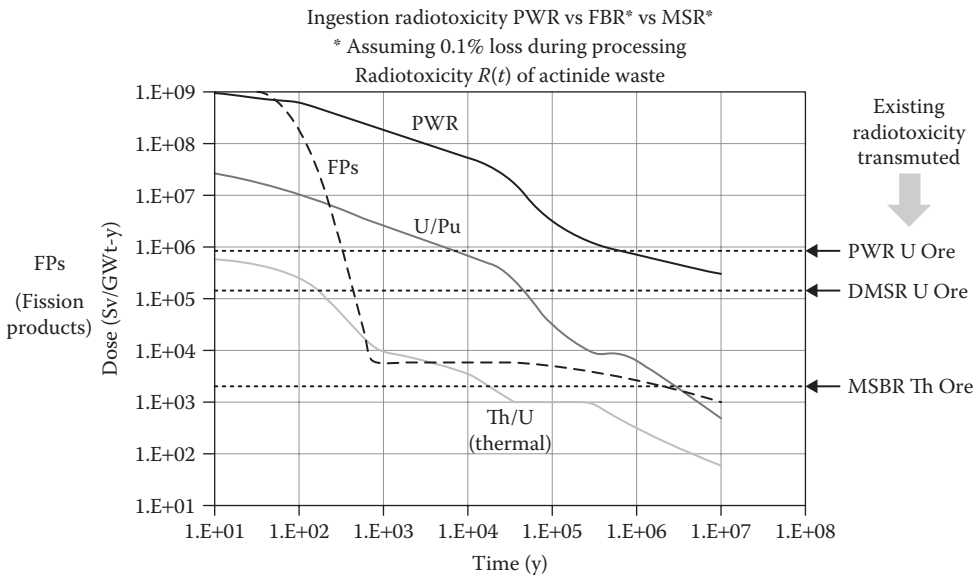
7.1.1.4 Long-Lived Waste

The fluid fuel form of MSRs leads to a very attractive long-lived waste footprint:

- MSRs offer large advantages for the destruction of transuranic wastes (Pu, Np, Am, Cm) from traditional once-through reactors, with these TRUs useful as both startup and makeup fuel. The avoidance of the need for fuel element fabrication and the ability to consume minor actinides greatly improves functionality.
- Thorium-to-<sup>233</sup>U breeders with actinide recycle would produce several orders of magnitude less transuranic waste than a conventional LWR once-through cycle and significantly less than even a U-Pu fast breeder (based on 0.1% losses during fuel processing). This leads to the radiotoxicity of waste being less than the equivalent uranium ore used in LWRs within a few hundred years.
- MSR burner design can also close the fuel cycle by recycling actinides (processing can occur offsite for burner designs). Burners using low enriched uranium (LEU) as startup and makeup fuel also consume and transmute considerable existing long-lived radiotoxicity of the mined uranium. They can achieve the unique characteristic of transmuting or destroying more long-lived isotopes than they produce, effectively lowering long-lived radiotoxicity on earth (Figure 7.1).

7.1.2 MSRs of the First Nuclear Era

MSRs were developed primarily at Oak Ridge National Laboratories (ORNL) beginning in the late 1940s. Decades of funded research and development followed a design evolution leading to the adoption in the late 1960s of what is known as the single fluid, graphite-moderated MSBR (Weinberg et al., 1970). What is important to realize is that this evolution



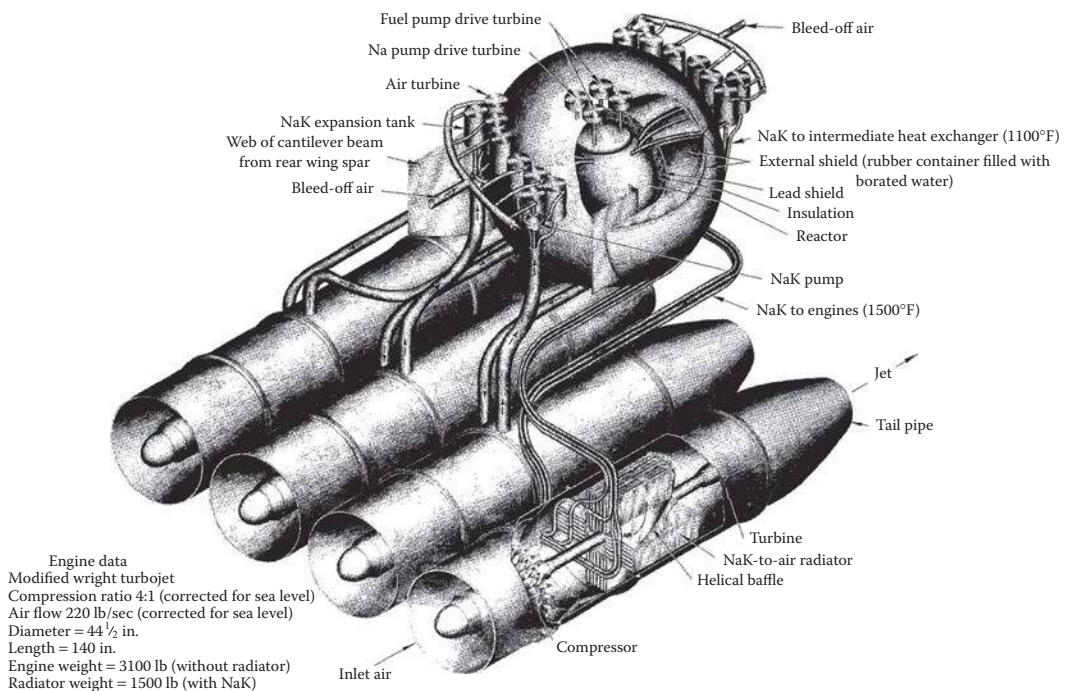
**FIGURE 7.1** Relative radiotoxicity. (Data and graph from Sylvain David, Institute de Physique Nucléaire d’Orsay, Orsay, France.)

was guided by priorities and limitations that are far different than now. In particular, the overwhelming priority given to the MSBR program was a minimization of the doubling time, the time to breed the start-up fissile inventory of the next reactor. The two routes for minimizing this doubling time are decreasing the necessary start-up inventory and increasing the breeding ratio. This mandated priority was due to the early belief that nuclear power would follow an exponential growth, and that world uranium supplies were severely limited. Associated to this was that the MSBR's main competition was the well-funded liquid sodium-cooled, oxide-fueled, fast breeder whose potential doubling time was impressive at the time but later would be greatly lengthened due to the need to soften the neutron spectrum for safety reasons.

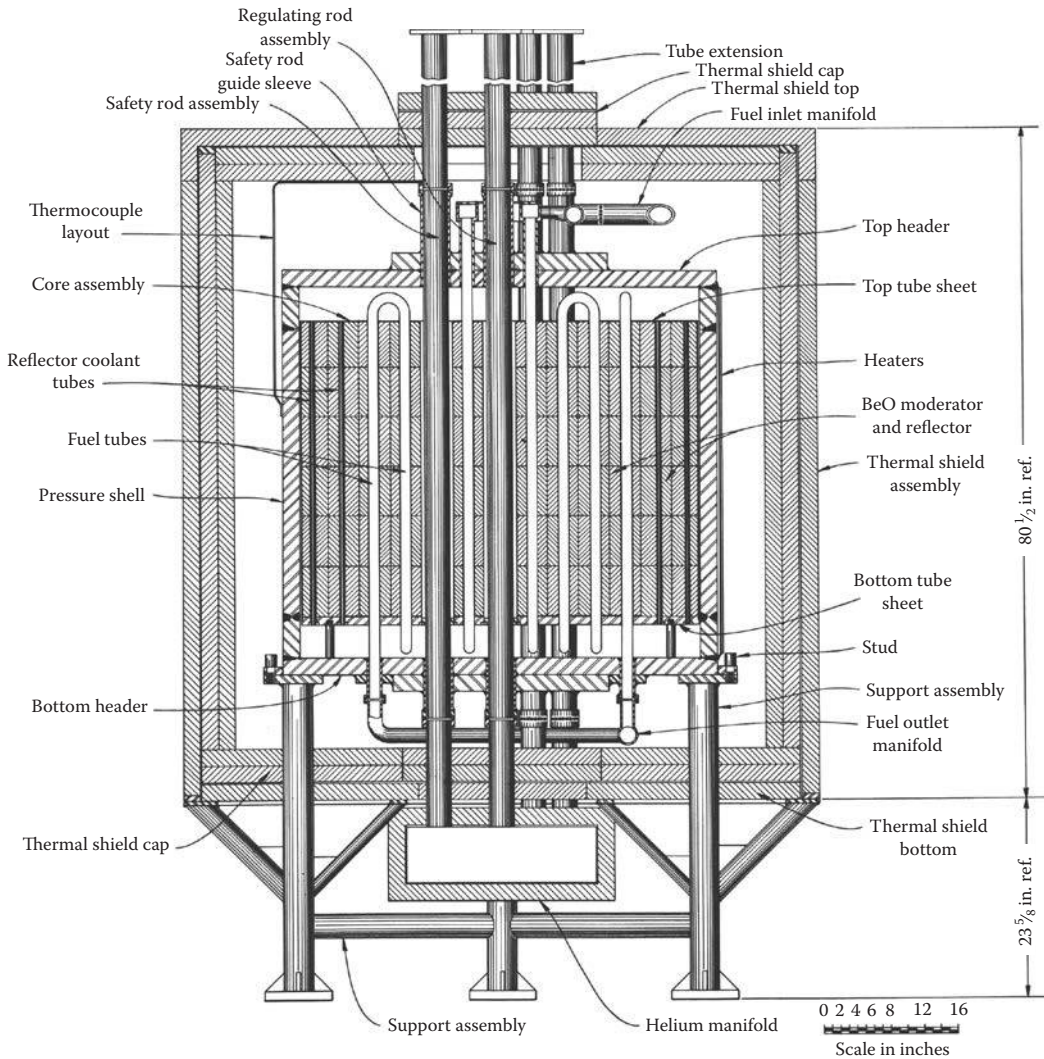
### 7.1.2.1 Aircraft Reactor Program

The very first MSR laboratory work was in support of the Aircraft Reactor Program for the US Air Force. The concept was to have an onboard MSR to replace combustion heat for the jet engines of bombers to enable them to stay aloft for weeks at a time. While this project did not lead to an operational bomber, it did lead to a large knowledge base being developed and to a successful test reactor, the Aircraft Reactor Experiment (ARE) in 1954. The ARE was a high temperature reactor with a peak temperature of 860°C employing a NaF-ZrF<sub>4</sub> carrier salt and fueled with highly enriched <sup>235</sup>U. Clad blocks of BeO provided moderation (Figure 7.2).

The ARE was a simple proof-of-concept reactor, not meant for long duration usage, as little was known of corrosion aspects at that early date. As well, even operational aircraft reactors were not expected to have a long life, potentially just a single, months-long



**FIGURE 7.2**  
Aircraft reactor concept.

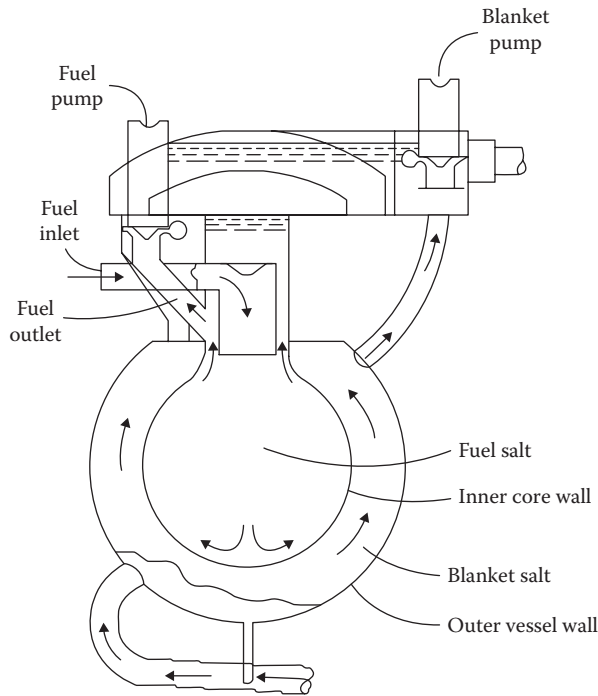


**FIGURE 7.3**  
Aircraft reactor experiment.

mission followed by replacement of the reactor. The ARE ran for several hundred hours at a power of up to 2.5 MWth and demonstrated load following without control rod action and the passive removal of Xe and Kr. Later a fueled but zero power test reactor was run in 1957, the PWAR-1 (Pratt & Whitney Aircraft Reactor) ran at high temperature by electrical heating. It was a validation system for the proposed fireball reactor which was partially constructed but never operated as the advent of ICBMs lead to the cancellation of the Aircraft Reactor Program (Figure 7.3).

### 7.1.2.2 Early Power Station MSR Designs

A true Molten Salt Program began in the mid-1950s to investigate power station designs to be either simple burner reactors or potential breakeven and/or breeder reactors based on the Th-<sup>233</sup>U cycle. The earliest work took the form of very simple, sphere within sphere,



**FIGURE 7.4**  
Two region core.

two region designs as depicted in Figure 7.4. A central molten salt core would contain fissile and also usually fertile material and was surrounded by a  $\text{ThF}_4$ -containing blanket salt. This arrangement was often termed a *one-and-a-half fluid* design by ORNL. The  ${}^7\text{Li}\text{-BeF}_2$  carrier salt itself is capable of modestly efficient moderation of neutrons such that a wide variety of neutron energy spectrums from soft epithermal to modestly hard were examined. A Hastelloy N<sup>TM</sup> barrier between regions was to be employed. In later years it would be found that such a usage within the core and thus within a high neutron fluence would have a severely limited lifetime due to helium embrittlement from  $(n, \alpha)$  reactions in nickel and any boron contamination.

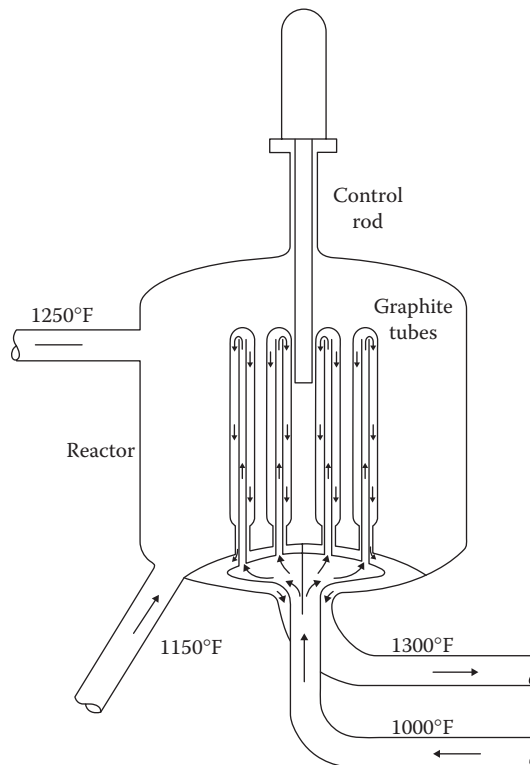
By 1960, it was determined that graphite was indeed capable of long-term interaction with the salts and appeared to offer a more efficient means of neutron moderation plus the potential to lower fissile starting loads. The use of graphite within the core also gave researchers at ORNL the hope that a true two fluid design could be made practical.

A two fluid design uses a fuel salt that carries the fissile  ${}^{233}\text{UF}_4$ , but no fertile material and a separate blanket salt for the fertile  $\text{ThF}_4$ . As  ${}^{233}\text{U}$  is produced in the blanket, it is transferred to the fuel salt by a relatively simple fluorination process. Fluorination is accomplished by bubbling fluorine gas through the salt, thereby converting  $\text{UF}_4$  to volatile  $\text{UF}_6$ . Gaseous  $\text{UF}_6$  is collected and converted back to  $\text{UF}_4$  by a well-established process before being added to fuel salt.

The main advantage of a two fluid system is that the processing-out of fission products from the fuel salt is simplified by the absence of thorium. The prime method is known as vacuum distillation (ORNL 3791, 1966) and was developed in 1964. After removal of all  $\text{UF}_4$ , the carrier salt would be evaporated off at low pressure and high temperature ( $1000^\circ\text{C}$ ) and recycled, leaving most fission products behind in the still bottoms.

An additional feature in a two fluid design is that the neutron losses to  $^{233}\text{Pa}$  can be minimized as  $^{233}\text{Pa}$  is effectively diluted in the much larger volume of blanket salt and experiences a lower neutron flux. ( $^{233}\text{Pa}$  is the 27 day half-life intermediate which decays to  $^{233}\text{U}$  following its production by a neutron absorption in  $^{232}\text{Th}$ .) Thus, two fluid designs do not require complex and rapid processing to remove Pa to allow it to decay to  $^{233}\text{U}$  as was often thought necessary. ORNL, however, often did propose Pa removal in much of their two fluid work, but only to slightly improve breeding ratios (1.07 vs. 1.05). There would be little modern incentive to include Pa removal, and large economic and other benefits from avoiding this process.

In this new two fluid concept, the graphite was to be used to internally separate the fuel salt and blanket salt within the core (ORNL 4528, 1970) as shown in Figure 7.5. However, this use of graphite *plumbing* of hundreds of tubes proved a large challenge, as a crack in any tube might mean replacement of the entire core and vessel due to its complexity. Under fast neutron fluence, graphite will first contract and then expand leading to a limited lifetime depending on power density and graphite temperature. Cores with fairly high power densities were preferred by ORNL, giving typical graphite lifetimes of 4 years before a peak fluence of  $3 \times 10^{22} \text{ n/cm}^2$  ( $E > 100 \text{ keV}$ ) was reached. A further complication with graphite is that the contraction and expansion of graphite also led to a troublesome changing ratio of fuel to blanket salts within the core, which could adversely affect reactivity. It is a testament to the advantages of two fluid design that ORNL stuck with this complex design until the late 1960s, despite these often termed *plumbing problems*. Modern improvements to the two fluid concept have been proposed (LeBlanc 2008, 2010).

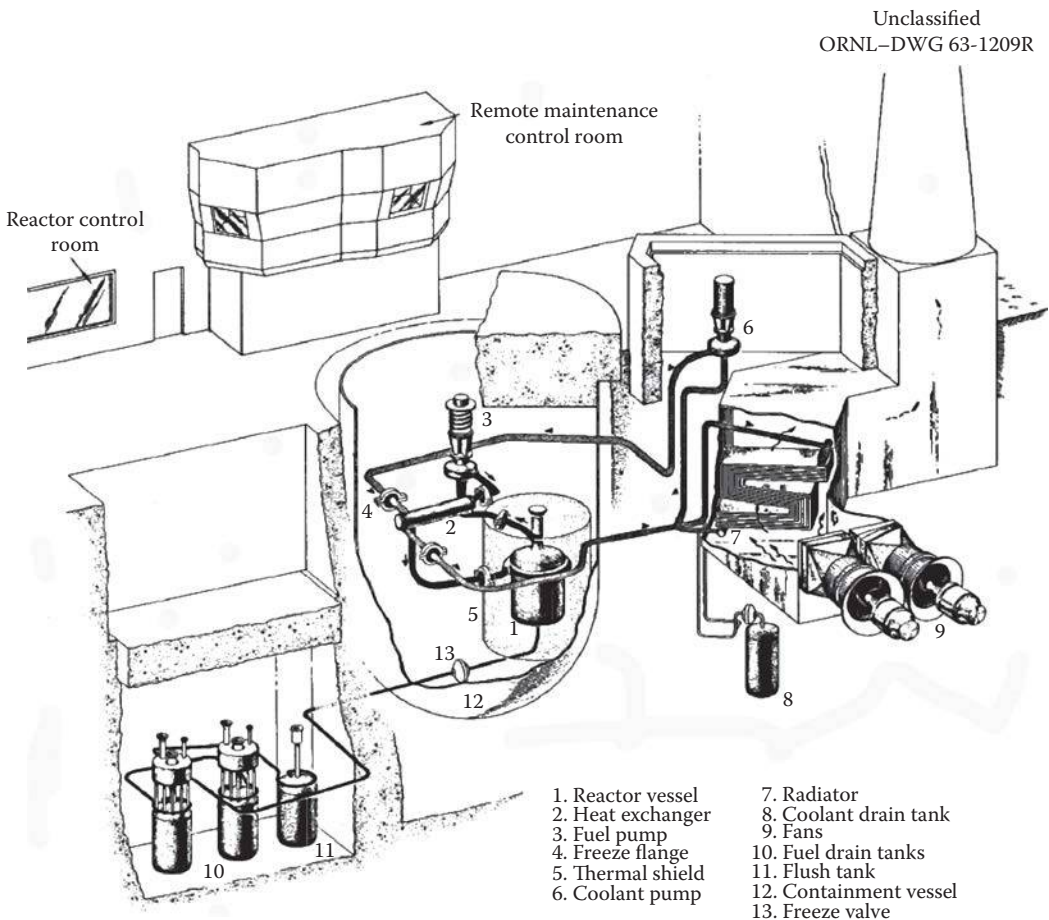


**FIGURE 7.5**  
Two fluid alternative.

### 7.1.2.3 MSRE

Also during the 1960s, the highly successful test reactor, the MSR experiment, was constructed and operated. It was an 8 MWth design chosen to be a single fluid for simplicity and operated for close to 5 years with great success. Two unknown issues with Hastelloy N™ did surface however, one being the corrosion induced by the fission product tellurium and the other irradiation damage caused by (n, alpha) reactions of <sup>59</sup>Ni and boron contaminants. In subsequent years of the program, these issues were largely addressed through modification of the alloy makeup of the Hastelloy and also by adjustment of the reduction potential of the salt, accomplished simply by the occasional addition of metallic beryllium. It is certain, however, that Hastelloy N™ would have a limited lifetime if used within the full neutron flux of the core. Use in the outer vessel walls and heat exchangers should pose far less of an issue; however, substantial work will be required in order to qualify any new alloys for ASME Section III use (Figure 7.6).

MSRE plant diagram: (1) reactor vessel, (2) heat exchanger, (3) fuel pump, (4) freeze flange, (5) thermal shield, (6) coolant pump, (7) radiator, (8) coolant drain tank, (9) fans,



**FIGURE 7.6**  
The MSRE plant.





**FIGURE 7.7**  
MSRE core and vessel.

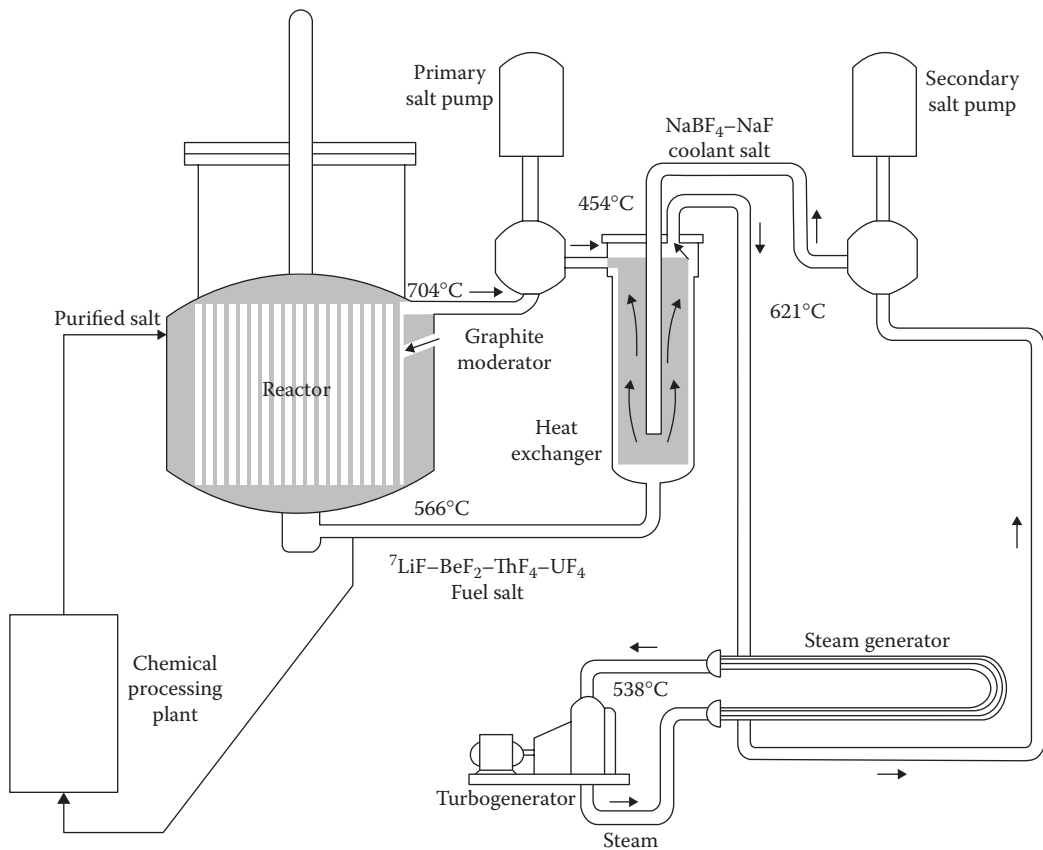
(10) fuel drain tanks, (11) flush tank, (12) containment vessel, (13) and freeze valve. Also note Control area in upper left and Chimney upper right (Figure 7.7).

#### 7.1.2.4 Single Fluid MSBR

In 1968, the development of a new fission product removal method that could function in the presence of thorium and an ingenious technique to limit neutron leakage in a single fluid design along with the plumbing problems of the two fluid design mentioned above led ORNL to abandon its two fluid breeder concept. Liquid bismuth reductive extraction (ORNL-TM-3137, 1971), while not a simple technique, was shown to be able to differentiate between thorium and the very important rare earth fission products.

Single fluid designs with both thorium and  $^{233}\text{U}$  within the same salt also mean that  $^{233}\text{Pa}$  will be a larger drain on the neutron economy. In ORNL's MSBR design, the breeding ratio would drop from 1.06 down to 1.01 if  $^{233}\text{Pa}$  was not removed on a 3–10 day time frame (ORNL 4548, 1970). Given the strong emphasis placed on breeding at the time, it was felt that Pa needed to be removed and allowed to decay outside the core before being returned as  $^{233}\text{U}$ . Pa removal is also accomplished by liquid bismuth reductive extraction.

Another area of advance that made a single fluid graphite-moderated concept appear more attractive was the idea of employing an undermoderated outer zone of the graphite core in which a higher ratio of fuel salt to graphite hardened the spectrum and led to the region being a net absorber of neutrons. This results in a great reduction in neutron leakage and/or reflector losses. Typically, about 12% salt volume proved best in the inner zone while 37% salt volume was proposed for the outer absorbing zone. The 37% value was chosen to allow the possible use of graphite pebbles. A random bed of pebbles occupies 63% of the volume (although typically 61% is now generally assumed).



**FIGURE 7.8**  
Section diagram of MSBR.

The end result was the *traditional* MSBR design (Weinberg et al., 1970), Figure 7.8, which has been included as one of six Generation IV reactors in 2002—relatively unchanged. This design has a starting fissile load of 1500 kg  $^{233}\text{U}$ /GWe, a breeding ratio of 1.06, a 20 day cycle for fission product and 3 day cycle for  $^{233}\text{Pa}$  removal (both by liquid bismuth reductive extraction). Sealed graphite is used to limit xenon retention, with a 4 year graphite replacement cycle. The 1000 MWe design has an average power density of 22 MWth/m<sup>3</sup> and a core 5 m in diameter and 4.3 m in height.

In the early 1970s, however, for reasons many would argue were far more political than technical, the MSBR program was terminated by the AEC. Corrosion issues that were already well on their way to being addressed were the main faults mentioned (Shaw, 1972) and which unfortunately have become engrained in memory. Certainly significant work is still needed to qualify materials for the primary loop including heat exchangers, but the materials challenge is likely far less than the AEC report implied. The sudden termination of funding also meant the single fluid design became the textbook design and little mention of alternatives has appeared until recent years.

Oak Ridge did continue a modest program until the early 1980s, with a greatly increased value placed on increasing proliferation resistance. ORNL examined operation on denatured cycles in which all uranium stayed below the weighted average of 12%  $^{233}\text{U}$  and/or 20%  $^{235}\text{U}$ . The results were surprisingly successful, with two routes examined, both termed

DMSR. The first a DMSR break-even design (ORNL-TM-6413, 1978) with similar fission product processing to the MSBR and showed that break-even breeding could be accomplished even while remaining in a denatured state using depleted  $^{238}\text{U}$  and Th as the fertile makeup. The second was a greatly simplified DMSR burner design called the *30-year once-through design* (ORNL-TM-7207, 1980), without any fuel processing beyond chemistry control for a full 30 years, while still maintaining a very high conversion ratio and excellent uranium resource utilization. Both designs also featured larger, low power density cores that gave a full 30 year graphite lifetime.

Over the next 20 years however, little advance was made as funding worldwide virtually dried up. Many vocal proponents worldwide such as Charles Forsberg and Uri Gat of ORNL and Kazuo Furukawa in Japan at the very least kept interest active during lean times for all nuclear designs (Forsberg, 2006; Furukawa et al., 1990; Furukawa, 1992; Gat, 1992, 1997) (Table 7.1).

### 7.1.3 Unique Operational Issues of MSRs

While the unique aspect of liquid-fueled operation is at the core of many of the benefits of MSRs, there are of course a number of potential operational issues and/or challenges unique to the MSR concept. Being a liquid implies a mobile fuel form which can appear to be a large challenge. In practice however, an MSR can offer numerous levels of containment, starting with the fluid fuel itself, then the primary circuit which is typically housed within a tightly sealed, low pressure cell and all within a reactor building. There

**TABLE 7.1**

ARE, MSRE, Two Fluid MSBR, Single Fluid MSBR, and DMSR

	ARE	MSRE	Two Fluid MSBR	Single Fluid MSBR	DMSR Burner
Program period	1950–1955	1962–1969	1964–1967	1967–1976	1978–1980
Power output	2.5 MWth	8 MWth	1000 MWe	1000 MWe	1000 MWe
Operational Hours critical	1954 221 h	1965–1969 17,655 h	Conceptual design	Conceptual design	Conceptual design
Moderator	Clad BeO	Graphite	Graphite	Graphite	Graphite
Fuel salt	53%NaF- 41%ZrF <sub>4</sub> - 6%UF <sub>4</sub>	65% <sup>7</sup> LiF- 29.1%BeF <sub>2</sub> - 5%ZrF <sub>4</sub> - 0.9%UF <sub>4</sub>	65.8% <sup>7</sup> LiF- 31.3BeF <sub>2</sub> - 0.2%UF <sub>4</sub>	71.7% <sup>7</sup> LiF- 16%BeF <sub>2</sub> - 12%ThF <sub>4</sub> - 0.3%UF <sub>4</sub>	74% <sup>7</sup> LiF- 16.5%BeF <sub>2</sub> - 9.5%(Th, U)F <sub>4</sub>
Conversion ratio	NA, lacked fertile	NA, lacked fertile	1.07	1.06	0.8
Starting fissile mass	14.9 kg <sup>235</sup> U 93% enriched	77 kg <sup>235</sup> U 35% enriched	700 kg <sup>233</sup> U	1500 kg <sup>233</sup> U	3450 kg <sup>235</sup> U 20% enriched
Core H × D	0.91 m × 0.85 m	1.62 m × 1.37 m	3.8 m × 3.0 m	4.3 m × 5 m	8.3 m × 8.3 m
Fuel salt	654°C	632°C	565°C	565°C	565°C
Inlet	827°C	654°C	704°C	704°C	704°C
Outlet temperatures					
Temperature coefficient of reactivity	-17.5 pcm/K	-14.6 pcm/K	-4.3 pcm/K blanket salt positive void coefficient	-2.4 pcm/K recent indications may be positive	-6.7 pcm/K

are though, many other unique aspects of MSR design, some universal and some depending upon design goals, for example, burner versus breeder operation.

### 7.1.3.1 Off-Gas Management

The liquid fuel of MSRs allows fission product gases such as xenon and krypton to be released from the liquid. Along with xenon and krypton, noble metal fission products may entrain and exit as a fine particulate smoke (ORNL 4865, 1975). Iodine tends to form a stable iodide within the liquid salt but a fraction is expected to entrain as well. Finally, while alkali, alkaline, and lanthanide fission products will bind to fluorine to form salt-stable fluorides when born within the salt, there are many unstable isotopes of xenon and krypton that after leaving the salt subsequently decay to daughter isotopes including  $^{137}\text{Cs}$  (30 year) and  $^{135}\text{Cs}$  (2.3 million year) along with numerous other isotopes of Ba, La, Ce, Rb, and Sr.

One obvious advantage is avoiding neutron absorption losses to these isotopes, especially  $^{135}\text{Xe}$ . Indeed, for this reason it has typically been proposed to aid the exit of noble gases by bubbling helium through a side stream of fuel salt. For example, for breeder MSRs a goal has been an effective 50 s removal time for xenon and a poison fraction of only 0.5% (fraction neutrons lost to  $^{135}\text{Xe}$  compared to those absorbed in fissile isotopes). The removal of gaseous and volatile fission products helps remove source term from the primary system but then requires the careful construction of on-site facilities to manage both the heat generation of the off-gas stream and to sequester all solid fission products (entrained nobles and Xe and Kr daughter products).

For example in the 1970s 1000 MWe MSBR design of ORNL, off-gases would be first shunted to the existing decay heat dump tanks for the first 1–2 h with 20 MWth of heat production ( $^{137}\text{Cs}$  remains here from 3.8 min  $^{137}\text{Xe}$ ). Next, it would be transferred to 47 h charcoal delay beds needing to handle 2 MWth and where most  $^{135}\text{Cs}$  is deposited. Finally, 90 day long-term beds would capture final daughter products like stable  $^{133}\text{Cs}$  from 5.2 day  $^{133}\text{Xe}$ . In the end, about 23 m<sup>3</sup> of stable Xe and Kr (including  $^{85}\text{Kr}$  having a half-life of 10 years) would be stored in eight standard gas bottles for each year of operation.

The design and integration of these facilities were not considered a large challenge or capital expense to ORNL developers but the design and especially regulatory challenge of this undertaking is not to be taken lightly.

### 7.1.3.2 Tritium Confinement

Most proposed MSR designs call for the use of enriched lithium-7 and/or beryllium such as in the 70.7% $^7\text{LiF}$ –17% $\text{BeF}_2$ –12% $\text{ThF}_4$ –0.3% $\text{UF}_4$  fuel salt of the 1970s MSBR program. This carrier salt is often termed FLiBe. The presence of either  $^7\text{Li}$  or Be leads to significant production of tritium, on par with production levels in heavy water reactors such as CANada Deuterium Uranium (CANDU). For example in the 1970s, the 1000 MWe MSBR projected 2420 curies per day (ORNL 4541, 1971), 98.3% from lithium and beryllium, and the remainder of 0.4% from fluorine and 1.2% from ternary fissions. For comparison, CANDU operations typically experience a tritium release rate of less than 24 curies per day (CNSC INFO-0793, 2009).

Tritium in MSR designs is a major concern since it can pass readily through the hot walls of heat exchangers and/or vessel walls. As such, effectively trapping tritium and preventing it from reaching the final working medium has been a continuing focus of MSR efforts. Early efforts at ORNL proposed using a secondary intermediate coolant salt of

$\text{NaBF}_4$ - $\text{NaF}$  able to bind up tritium (ORNL 4541, 1971). Alternatively, another consideration was adding a third loop using nitrate-based salt such as Hitec to oxidize and bind tritium (ORNL-TM-3428, 1971).

Graphite itself also has the ability to bind and contain tritium, and some modern efforts seek to employ that behavior through increased surface area such as the case with pebble bed, molten salt-cooled designs (Peterson et al., 2008). Graphite also has been seen to increase its tritium retention with neutron irradiation, which aids in this area (Causey, 1989).

Another solution is to simply avoid the use of either lithium-7 or beryllium in the carrier salts. This is not considered an option for most breeder MSR designs due to increased neutron absorptions in the potential substitutes such as  $\text{NaF}$ ,  $\text{RbF}$ ,  $\text{KF}$ , and/or  $\text{ZrF}_4$ . Furthermore, for the new field of molten salt cooled, solid-fueled work (see Section 7.1.5.1.2),  $\text{FLiBe}$  is needed to assure a negative void coefficient. Burner MSR versions are the notable exception and allow increased carrier salt options, as neutron losses are far less of a concern.

### 7.1.3.3 Noble Metal Plate Out

Noble metal fission products will not form stable fluorides in the fuel salt but will tend to plate out on surfaces in the primary loop. The main complication that arises from this is in terms of the primary heat exchangers. Work with the MSRE indicated that upward of 40% of noble metals plated-out on the walls of heat exchanger. If shell and tube designs are proposed, then the heat generation by noble metals attached to tube surfaces can be a concern if both primary and secondary salts are drained from the heat exchangers, a rare but plausible event. ORNL determined this a manageable but concerning situation (ORNL TM-3145, 1971).

The other aspect of noble metal plate out is the fact that heat exchangers will be substantially activated, which greatly complicates service and/or repair. The design philosophy of early developers was to strive for the highest quality of construction, and if tube leaks or other defects did arise, to shutdown and swap out the entire tube bundle. Only after years in shielded storage would a decision be made on tube bundle repair.

### 7.1.3.4 Corrosion

The issue of corrosion of Hastelloy N<sup>TM</sup> on exposure of fluoride-based fuel salts is one that has attracted a perhaps unwarranted concern in the past. Numerous long-duration salt loops employing various fuel salt and/or coolant salts performed very well (ORNL-TM-4286, 1972), and also several stainless steels such as 316SS (ORNL-TM-5782, 1977), while somewhat inferior to Hastelloy N<sup>TM</sup>, showed significant promise.

During operation of the MSRE however, an issue of modest concern did surface. That being that the fission product tellurium was depositing within the grain boundaries of Hastelloy N<sup>TM</sup>. Upon observation of test samples and reactor components post-shutdown, shallow surface cracking, less than 15 mils deep, was evident on surfaces exposed to the fuel salt.

A program by ORNL later showed this issue to be manageable by a combination of changing the alloy makeup and reduction potential of the fuel salt (ORNL 4812, 1972). The tellurium cracking however was widely cited in the WASH 1200 (Shaw, 1972) report used to justify the cancellation of the MSBR program in the early 1970s.

While the issue of tellurium cracking has been largely addressed, it is of course evident that the concept of decades long service of materials within an MSR environment will require significant theoretical and experimental validation.

### 7.1.3.5 Redox and Chemistry Control

Any discussion of corrosion issues must stress in the strongest possible terms the need for proper chemistry control and the monitoring and adjustment of the operating salts reduction potential. Oxide or moisture contamination are crucial to avoid, and the proper pre-treatment of salt and entire primary loop are vital to perform properly. Various stages of helium and nitrogen purges of systems followed but flushing with clean salts are employed in this regard. In the MSRE, the fuel salt contained 5%  $ZrF_4$  as a precaution against oxide contamination (acting as an oxygen getter). In practice, it was found possible to avoid such contamination and future MSBR designs did not propose including  $ZrF_4$ .

The reduction potential of the operating salts is equally important in terms of corrosion control to maintain at a proper level. This is typically measured by the ratio of  $UF_4$  to  $UF_3$  in the salts and ranges of 20–50  $UF_4/UF_3$  are desired. During operation, the fission process liberates minor amounts of free fluorine (four released from a  $UF_4$  molecule and just over three bind to fission products). By contacting the salt as desired with a metal such as beryllium or thorium, excess fluorine is absorbed producing  $BeF_2$  or  $ThF_4$ .

### 7.1.3.6 Helium Embrittlement

Another issue that MSRE operations brought to light, and one discovered in many other types of reactors is that of embrittlement caused by the production of helium, in this case within Hastelloy N™ exposed to significant neutron fluence. Early investigations were complicated by a mistaken belief that helium production was primarily from (n, alpha) reactions in boron impurities. Boron's large absorption cross section meant that ORNL believed that after a modest neutron fluence, boron would burn out and helium production would cease. However, it was later realized that the actual prime component was (n, alpha) reactions in  $^{59}Ni$ , itself produced by radiative capture in  $^{58}Ni$ . The high energy of alpha recoil in these predominantly thermal-spectrum interactions can also add significant displacement damage.

For subsequent MSR work at ORNL and elsewhere, more conservative assumptions have been made on the neutron fluence for any structural Hastelloy N™ elements. The level of neutron flux reaching the vessel wall can be limited by many means especially as thermal neutrons dominate He production and thermal neutrons can be more easily shielded, by boron carbide for example. In general as well for some MSR designs, combinations of graphite reflectors and/or undermoderated outer zones can be employed, greatly limiting neutron flux impingement on the reactor vessel wall.

Some MSR designs call for dual fluids and the need to provide physical separation while in a strong neutron flux. In these cases, other materials more favorable to neutron flux, molybdenum for example, might be employed. These other material choices however often bring their own unique challenges.

### 7.1.3.7 Delayed Neutrons

Most MSR designs involve a liquid fuel that transits back and forth from the critical core region to heat exchangers. This represents the potential loss out of core of delayed neutrons that aid the control of the reactor. Fortunately, fast acting temperature reactivity coefficients are typically strongly negative and reactor control is already otherwise very stable. As well, many of the shorter-lived delayed neutrons are still released during transit through the core (core transit time is typically several seconds, as much as 20 s in the

DMSR design). The basic design goal has been to limit the fraction volume of salt out of core to be one-third to one-half of the total volume. In the MSRE however, it operated with nearly two-thirds of salt out of core with no real control concerns (ORNL TM 0730, 1964).

Another aspect of delayed neutrons out of core involves potential activation of heat exchangers and secondary coolants. Further, the removal of a fraction of delayed neutrons during normal operation means that if salt flow is suddenly arrested, those normally lost neutrons are emitted in the core and represent a modest reactivity insertion. In the MSRE, stoppage of flow was measured as a 21.2 pcm ( $1 \text{ pcm} = 10^{-5} \Delta k/k$ ) insertion. As the MSRE fuel salt temperature reactivity (not including the slower acting graphite term) was about  $-7 \text{ pcm/K}$  (Haubenreich and Engle, 1970), it would mean a power increase until the fuel salt temperature increased about  $3^\circ$  in compensation. Obviously, stoppage of flow would itself lead to a salt temperature increase in the core and without any intervention would drive the core subcritical.

#### 7.1.3.8 Protactinium Issues

Many MSR designs assume the use of fertile thorium. Neutron capture does not immediately produce  $^{233}\text{U}$  but must go through the 27-day half-life intermediate  $^{233}\text{Pa}$ . For breeder MSR designs, there is concern of neutron absorption in  $^{233}\text{Pa}$  affecting breeding ratios. As early efforts were strongly focused on maximizing breeding ratios, it was typically proposed to rapidly process the entire fuel salt (3–10 day cycle) to remove and isolate  $^{233}\text{Pa}$  to let it decay to  $^{233}\text{U}$ . This represents a very significant engineering and capital cost challenge and also adds proliferation concerns.

$^{233}\text{Pa}$  within a fuel salt also adds an interesting effect during long-term shutdown of the reactor. In this situation, there is potential increase in reactivity as  $^{233}\text{Pa}$  decays to fissile  $^{233}\text{U}$ . If it is intended that the reactor be kept in hot standby, shutdown systems must take this factor into account.

#### 7.1.3.9 Graphite Issues

The use of graphite as moderator in MSR designs brings many advantages, primarily due to the provision of a more thermalized neutron spectrum. Starting fissile loads are greatly reduced, neutron lifetime is increased, aiding reactor control and transient response as well as offering increased options to reduce neutron flux leaving the core region and reaching the reactor vessel wall.

A significant operational issue with graphite use is a limited lifetime proportional to peak power density in the reactor. Under neutron irradiation graphite undergoes dimensional changes, first shrinking modestly and then undergoing expansion. Lifetime is typically dictated by the fast neutron fluence causing the graphite to commence expanding beyond its original volume. This limit is also temperature dependent, and as such, steps are taken to limit peak temperature.

A fundamental decision has typically faced MSR designers: to either design for a low enough peak power density to allow a full reactor lifetime of the core graphite or to allow a higher power density but to make provisions for the replacement of graphite elements periodically from the core. The early 1970s MSBR design, for example, planned for a peak power density of roughly  $70 \text{ MWth/m}^3$  and to replace core graphite as a whole every 4 years. Later work on the 1980 DMSR burner reactor looked to greatly lower peak power density to  $10 \text{ MWth/m}^3$  by increasing core dimensions significantly, to allow a 30-year operational life without graphite replacement (24 full power years).

### 7.1.3.10 Fuel Salt Processing

The use of a liquid fuel form greatly aids the employment of fuel processing for the goal of removing fission products to improve conversion ratios (fissile consumed/fission produced). This is primarily to allow conversion ratios of 1.0 or higher to attain breeder operation. In solid fuel reactors, fuel has a limited lifetime and then must be converted to a liquid form, processed as a liquid to separate fission products, converted back to a solid and finally refabricated into solid fuel elements. Liquid fuel has no burn up limitations in the reactor and can be siphoned off as need be, processed as a liquid, and sent back to the reactor as a liquid. Processing ability in an MSR is also aided by the fact that if desired, the uranium content can be removed by a process of fluorination. Fluorine gas is bubbled through the salt and the contained  $\text{UF}_4$  is converted to gaseous  $\text{UF}_6$ .

For breeder MSR versions, on-site continuous processing is typically proposed. In the early work of ORNL, as rapidly as the entire fuel salt on a 10-day cycle. In more recent proposed designs with harder spectrums, this time can be extended to several months and still allow breeding. With the thorium– $^{233}\text{U}$  cycle, a factor that greatly complicates processing is that thorium behaves chemically very similar to the lanthanide fission products.

Mid-1960s efforts at ORNL sought to bypass this issue by using two separate liquids, a fuel salt containing the produced  $^{233}\text{U}$  and a blanket salt containing the thorium to produce  $^{233}\text{U}$ . The blanket salt would require periodic fluorination to remove the produced  $^{233}\text{U}$  (although often rapid  $^{233}\text{Pa}$  removal was also suggested). The fuel salt, in a continuous side stream, would be first fluorinated to remove  $^{233}\text{U}$  and then salt processed by vacuum distillation where at high temperature ( $1000^\circ\text{C}$ ) the carrier salt would evaporate off and leave behind most fission products in the still bottoms.

By 1967 however, ORNL deemed another method suitable to remove lanthanides without also removing thorium and thus allow simpler single fluid operation. Here  $^{233}\text{U}$  would also first be removed by fluorination and then the salt would undergo a series of liquid metal reductive extraction steps where a reductant (lithium and/or thorium) in a liquid bismuth solvent would trade places with various fission products and transuranics (TRUs). Carefully staged operation could allow separate removal of thorium, and most fission products and the cleaned carrier salt could then be recombined with the extracted  $^{233}\text{U}$  and thorium.

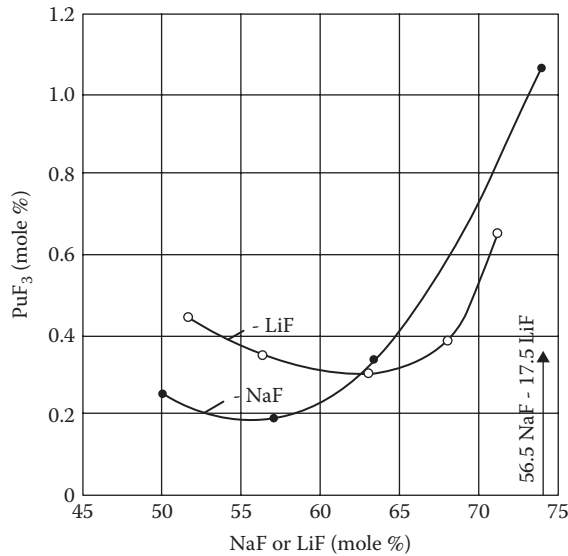
All processing methods however require far more development before commercial operations. While molten salt processes may be far simpler than solid fuel processes, they still represent a large capital and regulatory burden. As an example, the 1970s 1000 MWe MSBR design called for processing the entire fuel salt every 10 days, which would equate to 5600 tonnes processed per annum.

### 7.1.3.11 $\text{PuF}_3$ Solubility

In fluoride-based salts, while the amount of either  $\text{ThF}_4$  or  $\text{UF}_4$  contained in the salt can be high, the ability to contain  $\text{PuF}_3$  is often quite limited and highly dependent upon the minimum temperature of the salt. If minimum salt temperature is  $600^\circ\text{C}$  or lower, it is rare for  $\text{PuF}_3$  solubility to reach 3% molar (Figure 7.9 and Table 7.2).

For Th– $^{233}\text{U}$  breeder designs, the level of  $\text{PuF}_3$  does not build up to an appreciable degree. However, in breeder designs there is often the desire to utilize reactor-grade plutonium as startup fissile inventory. For graphite-moderated versions with a softer spectrum, the fission cross section for Pu is high enough that a low percentage will suffice. In the 1970s





**FIGURE 7.9**  
Solubility of  $\text{PuF}_3$  in  $\text{LiF-BeF}_2$  and  $\text{NaF-BeF}_2$  at  $565^\circ\text{C}$ . (Data from Barton, C.J., *J. Phys Chem.*, 64, 306–309, 1960.)

**TABLE 7.2**  
 $\text{CeF}_3$  Solubility as a  $\text{PuF}_3$  Surrogate

Salt Composition (mole %)			$\text{CeF}_3$ Solubility (mole %)		Heat of Solution (cal/mole)
LiF	$\text{BeF}_2$	$\text{ThF}_4$	$600^\circ\text{C}$	$800^\circ\text{C}$	
72	16	12	1.6	5.5	11,500
73	0	27	2.6	9.0	11,560
72.7	4.8	22.5	2.4	7.9	11,100
68	20	12	1.35	6.0	13,890
72.3	11.0	16.7	2.1	6.3	10,230
67.8	25.2	7.0	1.05	5.0	14,530

Source: Rosenthal, M.W. et al., Molten salt reactor program semiannual progress report, ORNL 4344, Feb 1969. With permission.

Note: Solubility and heat of solution of  $\text{CeF}_3$  in mixtures of LiF,  $\text{BeF}_2$ , and  $\text{ThF}_4$ .

MSBR design, starting with Pu required only 0.2%  $\text{PuF}_3$  (Perry and Bauman, 1970) in the fuel salt whereas in recent fast-spectrum, graphite-free work approximately 5%  $\text{PuF}_3$  would be required (SNETP, 2012).

### 7.1.4 Fundamental MSR Design Options

Achieving goals of safety, cost innovation, resource sustainability, and long-lived waste minimization has involved various potential design choices both during early development and recent MSR proposals. Early MSR development was mandated as a breeder program and the focus was upon breeding ratio and fuel cycle costs at perhaps the expense of capital cost reduction.

### 7.1.4.1 Internally Cooled

A rarely discussed option on MSR design is one in which the fluid fuel salt remains within the core and is cooled by the flow of another liquid coolant such as another salt, liquid metal such as lead, or a blanket salt containing fertile material. The main goal in this case is to reduce the volume of fuel salt and thus starting fissile inventory. The amount of heat transfer surface needed is however very large and any barrier material is within a high neutron flux. One such early design termed MOSEL in 1964 (Kasten, 1964), for example, needed such a high amount of heat transfer surface that the arrangement called for parallel plates of 2 mm of fuel salt, then 0.6 mm of metal barrier, and then 2 mm of flowing blanket salt, repeated throughout the meters-wide core.

As breeders, such designs can have impressive doubling times but typically are rather impractical. They remove the large advantage of other MSR designs wherein the fuel is also the coolant. Thus, as for solid-fueled reactors, the flow of coolant fluid must be assured in all cases. As well, these designs typically would have large positive void coefficients of the coolant in a loss of coolant scenario.

### 7.1.4.2 Blanket Salts

Blanket salt containing fertile material, typically  $\text{ThF}_4$ , can be employed in another fashion. In this more commonly proposed area, designs call for two separate fluids, one blanket and one fuel salt both flowing in and out of the core region.

ORNL used the term *two fluid* for MSR breeder concepts where the fuel salt contained only  $\text{UF}_4$  and no  $\text{ThF}_4$ , such as its design work in the mid-1960s with interlaced fuel and blanket salts separated by graphite plumbing. This presented large design challenges. Recent works (LEBLANC, 2008, 2010) showed a method to avoid intermixing by going to a unique elongated core geometry. In all cases, a barrier material between fluids must be proven. Also, any use of blankets can tend to invoke proliferation concerns.

Another MSR variation was conceived wherein a fertile blanket salt is only used to partially or fully surround the core region which contains the fuel salt—a mix of fertile and fissile. This was termed a one-and-a-half fluid design by ORNL.

### 7.1.4.3 Softer versus Harder Spectrum

The degree of neutron moderation, that is, the hardness of the neutron energy spectrum, can be quite widely varied in MSR design. Graphite provides an ideal moderator for the purpose of attaining a softer spectrum, but even with graphite use, the spectrum is most often more epithermal than thermal. Graphite use also allows for a core design that greatly limits the neutron flux reaching the vessel wall. First, by the use of graphite reflectors but more uniquely, by creating an undermoderated outer zone in the final few tens of cm of core, where the ratio of fuel salt to graphite is increased. This drives reactivity down in the region, that is,  $k_{\text{inf}} \ll 1.0$ , and promotes absorption by fertile over fissile absorption, beneficially capturing neutrons that may otherwise escape the core.

There are however advantages to attaining a harder neutron spectrum, such as potentially higher conversion ratios and more compact reactors. With fluoride salts, both elastic and inelastic collisions in fluorine result in spectrums that are not nearly hard as those seen in liquid metal-cooled fast reactors but are still hard enough for a large improvement in conversion ratios. This is partly due to a decrease in the absorption cross section of most fission products.

Below is a simplified comparison guide:

Harder Spectrum	Softer Spectrum
<ul style="list-style-type: none"> <li>• Avoids graphite or other use of moderator</li> <li>• High start up fissile inventory</li> <li>• High enrichment of fuel</li> <li>• Accidental criticality of spills or during fuel transport an issue</li> <li>• Low core mass and lower heat capacity lead to faster temperature rises in transients or loss of secondary cooling</li> <li>• Avoiding neutron leakage and/or reactor vessel damage a challenge</li> <li>• Unless a surrounding blanket is employed, only larger and higher power output cores (GWe+) will enable breeding as otherwise neutron leakage dominates</li> <li>• Higher conversion ratios possible</li> <li>• Higher power density of reactor</li> </ul>	<ul style="list-style-type: none"> <li>• Graphite use represents separate disposal issue</li> <li>• Low fissile start up inventory</li> <li>• Low enrichment of fuel</li> <li>• Criticality of salts only possible within core (for single fluid design)</li> <li>• High thermal inertia means slow temperature rise from decay heat</li> <li>• Reflectors and/or undermoderated zones to greatly lower neutron flux reaching vessel walls</li> <li>• Much longer prompt neutron lifetime aids reactor control</li> <li>• Must remove fission products rapidly <i>if</i> breeding desired</li> <li>• Rapid consumption if TRUs used as starting fuel or specifically for TRU destruction (far larger fission cross sections)</li> </ul>

There are certainly potential overall advantages of the fast spectrum MSR approach, especially in terms of achieving breeding status while minimizing on-site fuel processing. However, the significant challenges and development time may indicate that soft spectrum MSRs, benefitting from minimal departure from the operational MSRs of the past, would represent a first generation of design to reach commercial status.

#### 7.1.4.4 Burner versus Breeder

MSR designs have historically been focused on achieving breeding. The term *resource sustainability* is often used to describe attainment of breeding or at least break-even status with a fuel cycle. However, for burner designs, using conventional LEU as both startup and makeup fuel for example, there is also a strong case for resource sustainability. Natural uranium costs of all existing burner reactors (LWR, CANDU, etc.) is typically only about 0.2 cents/kWh, a small fraction of power production costs. Increases in the price of uranium thus have only a marginal effect and with any increase in uranium price, world uranium resources can be greatly expanded as will be explained next.

As with any mineral resource, an increase in price leads to an increase in exploration and an increase in reserves. To illustrate this point—in 2007 there was a spike in the price of uranium, which led to increased investment in exploration, with the result that world uranium resources increased by 15% in just 2 years. Compared to metals like copper and iron, which have been mined for millennia, uranium exploration is a relatively young field. Uranium can be expected to follow the resource expansion model of older commodities such as copper and iron. In the 1950s and 1960s, there was great doubt about future uranium resources but this is no longer the case, as large reserves have been discovered, and little exploration has been needed since the 1970s.

Also a positive for the uranium supply–demand balance, MSR burner designs are typically far more efficient than either LWR or even CANDU reactors. This is partially due to an improved neutron economy since no metallic structure or burnable poison need to be within the core. This is also due to unlimited fuel burn-up with liquid fuels since fission products produced (Pu and/or  $U^{233}$ ) remain within the salt until when a large fraction are consumed. In contrast, the solid fuel of conventional reactors is actually cracked and distorted

by the creation of volatile fission products and must be replaced, thus removing from use any unconsumed original fissile and much fissile that had been produced in situ from fertile absorptions.

Removing the necessity to achieve breeding status opens up a great variety of potential simplifications in MSR design. The largest advantage of burners over breeders is the avoidance of the R&D, capital, and regulatory hurdles of on-site fuel salt reprocessing needed for breeder designs. Further, alternate salts can be investigated, for example, ones that do not involve costly enriched lithium and/or beryllium (and the hard to manage tritium they produce). MSR burner designs offer a simpler approach. With nuclear, simplicity equates to cost savings, far more so than within other industries.

### 7.1.5 MSR's Second Nuclear Era

MSR's first nuclear era was firmly rooted in thorium-based breeder concepts at ORNL. Since that early work in the 1950s, 1960s, and 1970s, both ORNL and a surprising number of nations worldwide have embarked on their own MSR efforts in a wide variety of designs. The selection of MSRs as one of only six for the Generation IV forum in 2002 has contributed to a resurgence of interest in MSR design and its possibilities.

#### 7.1.5.1 US Efforts

##### 7.1.5.1.1 The Oak Ridge DMSR Burner 1980

As previously mentioned, the fission product processing called for in all early MSR design work at ORNL represented a significant cost both in equipment and for the R&D needed to verify processes on a commercial scale. Any fuel processing on-site or centrally must also be examined to assure a high level of proliferation resistance. Mainly with these thoughts in mind, ORNL performed a valuable study of the operation of a graphite-moderated, single fluid design as a simple converter or burner reactor without the requirement for fuel processing for an entire 30-year plant lifetime (ORNL-TM-7207, 1980). As with previous MSR designs, gaseous fission products would be collected and removed, while fission products forming stable fluorides would simply remain in the fuel salt for the life of the plant.

The DMSR converter or *30-year once-through design* entailed a larger, low power density core such that the graphite would have a full 30-year lifetime. Start-up for the 1000 MWe design was with 3450 kg of  $^{235}\text{U}$  added as 20% LEU. Along with this LEU was added 110 tonnes of thorium to greatly improve the neutron economy in the long term. The results of this study were most impressive. The cycle always remained in a denatured state and the conversion ratio averaged 0.8 over the 30-year design. Lifetime uranium utilization (30 year, 75% capacity factor) was only 1525 tonnes compared to 5430 tonnes for a once-through LWR (see Table 7.3). Furthermore, if a one-time-only simple fluorination to remove and reuse the remaining LEU is performed at the end of 30 years, then lifetime uranium needs drop from 1525 tonnes to about 850 tonnes.

The major resource advantage over other burner designs may seem surprising given that a conversion ratio of 0.8 does not appear that much superior to LWR and PBMR (both 0.5–0.6) or CANDUs (0.7). However, conversion ratios do not take into account the limited residency time of fuel in solid-fueled reactors. Perhaps, a new term of *effective* conversion ratio would be to compare fissile consumption versus needed annual fissile additions. By this metric, most other reactors on a once-through cycle have effective conversion ratio of near zero since they consume about 1000 kg/GWe-year but need to add 1000 kg of fissile  $^{235}\text{U}$  per year. Even with Pu recycle, they do not improve significantly. Thus, the great

TABLE 7.3

Uranium Utilizations—LWR to DMSR

Reactor Type	Lifetime Uranium (t)	Annual Uranium (t)	Annual Enrichment SWU	For Uranium at 100\$/kg M\$/year (¢/kWh)	For Uranium at 2500\$/kg M\$/year (¢/kWh)
LWR	5430	170	140,000	50 (0.6)	460 (7)
LWR U-Pu Recycle	3460	106	87,500		
DSMR Once through	1525	30	35,100	8.6 (0.1)	80 (0.9)
DMSR U recycle	850	30	35,100	8.6 (0.1)	80 (0.9)
DMSR 15 year batches	~500	~17	~20,100	4.9 (0.07)	46 (0.7)

Source: Nero, A., *A Guidebook to Nuclear Reactors*, University of California Press, Berkeley, CA, 1979. With permission.

Note: Based on 0.2% tails, 75% capacity factor, 30-year lifetime LWR.

advantage of molten salt burner reactors is that all plutonium produced stays within the core for the full lifetime where it is mostly consumed.

In terms of long-term radiotoxicity of wastes, these converter designs also perform remarkably well. All TRU remain in the salt during operation and will not reach high concentrations due to the very large cross sections for fission and/or absorption. At the end of 30 years, there is less than a tonne present. It is presumed that it will be deemed prudent to perform a onetime-only process to remove these TRU for recycle into the next core salt. If this is done and a typical processing loss of 0.1% is assumed, this represents a mere 1 kg of TRU going to waste over 30 years which is about a 10,000-fold improvement over LWR once through. An average then of about 30 g per GWe-year of TRU waste is even better than most pure Th-<sup>233</sup>U MSR concepts that process smaller amounts of TRUs but do so much more frequently. It should be mentioned that removing and reusing TRUs does not involve isolating plutonium. The likely process would be liquid metal reductive extraction and Pu would remain with all U, Np, Am, Cm, Cf, and the fission product zirconium. All these could be recycled into a subsequent batch of salt.

Such a recycling of TRUs also has a positive impact upon lifetime uranium utilization. In ORNL studies, at the end of the 30-year cycle, there would be 736 kg of Pu of which 331 kg would be fissile <sup>239</sup>Pu and 241 Pu. Along with this would be 136 kg of <sup>237</sup>Np and small amounts of Am and Cm (quantities not reported). There would also be enough uranium left to start the next cycle with clean salt, leaving the TRUs as source of fissile top-up. This should save approximately 5%–10% on the lifetime uranium utilization. Thus, the DMSR without any recycling requires about 1525 tonnes uranium lifetime, with U only recycling requires 850 tonnes, and with U + TRU recycling requirements are down to perhaps 800 tonnes.

Another aspect of this design which is perhaps little known is that operation as a converter or burner with <sup>238</sup>U present leads to much improved reactivity coefficients. Recent work in Europe (Nuttin, 2002) brought to light that the overall global temperature reactivity coefficient of the ORNL MSBR design may have been a slightly positive +0.6 pcm/K, as opposed to the small negative value of -2.4 pcm/K that ORNL calculated (1 pcm = 10<sup>-5</sup> Δk/k). For the converter design though, ORNL calculated (ORNL-TM-7207, 1980) a much higher value of -7.2 pcm/K. Much of the reactivity

benefit is derived from switching the fertile-fissile combination from Th +  $^{233}\text{U}$  to contributions from  $^{238}\text{U}$  +  $^{235}\text{U}$  and Pu.

The DMSR design was not taken beyond the conceptual stage but showed the very attractive features of a graphite moderated, burner MSR.

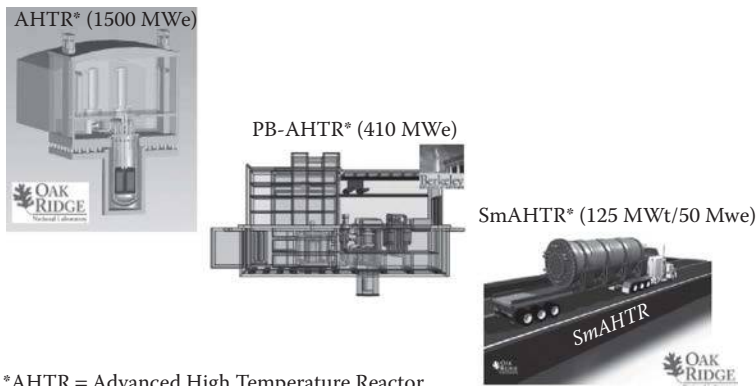
#### 7.1.5.1.2 FHR Molten Salt Cooled

Funding of MSR efforts in the United States have been virtually nonexistent since 1980 in the United States. However, important work has been underway for several years on a concept directly related to MSRs—the concept being to employ more conventional solid fuels, such as TRISO, but to change coolants to molten salts to take advantage of their high temperature and low-pressure operation. The molten salt coolants also offer the advantage of being able to bind volatile fission products released from any rare TRISO fuel failures. These efforts have become known as fluoride salt-cooled high temperature reactors or FHR but are often described as molten salt-cooled designs.

Efforts are centered upon both an academic alliance of UC Berkeley, MIT, and University of Wisconsin with, as well, significant efforts at ORNL and INL. The former group is primarily focused upon pebble bed concepts (Peterson et al., 2008) of various power outputs whereas ORNL is looking toward prismatic fuel forms. ORNL efforts have involved an attractive, truck deliverable SMR unit, the 50 MWe SmAHTR design (ORNL-TM-2010/199, 2010), along with a far larger proposal to take advantage of economies of scale, the 1500 MWe AHTR. For now, most efforts are with TRISO fuels and softer spectrums but there is also interest in faster spectrum designs; however coolant-cladding interactions will need to be much further advanced for these concepts (Figure 7.10).

The TRISO-based FHR efforts appear to many to be a more practical way to achieve licensing in a US regulatory environment geared toward solid-fueled designs. FHRs retain many of the advantages of MSR work, such as high temperature and high thermal efficiency along with a high degree of passive safety response. As heat must be removed from the solid fuel, FHR power densities are quite limited compared to MSRs. As well, there is a significant penalty on resource utilization compared to MSR designs, with FHR

FHR concepts are being developed for diverse applications



\*AHTR = Advanced High Temperature Reactor  
 PB-AHTR = Pebble Bed Advanced High Temperature Reactor  
 SmAHTR = Small Modular Advanced High Temperature Reactor

**FIGURE 7.10**  
 FHR program.

uranium requirements being calculated as roughly equal to LWR designs. Pebble beds have a modest uranium requirement advantage over prismatic efforts as the possible continuous replacement of pebble fuel can avoid the use of burnable poisons. Fuel reprocessing and re-fabrication could improve uranium needs; however, TRISO fuels are even more of a challenge than other solid fuels in this respect.

A major challenge in FHR work has been to assure a negative void coefficient of the molten salt coolant. Unlike helium coolant, the salts can have significant neutron absorbing behavior and thus their voiding could drive up reactivity. The use of FLiBe as coolant, which has minimal neutron absorption and also offers modest neutron moderating effects, allows softer spectrum FHR TRISO work to achieve negative void by running designs somewhat undermoderated. Thus if coolant voids, the loss of moderation overrides the lower parasitic neutron losses. In most cases, the amount of FLiBe coolant needed can be excessive in volume, for example, 1000 m<sup>3</sup> in the AHTR design (ORNL-TM-2011/365, 2011). Besides the high cost and current unavailability of 99.995% <sup>7</sup>Li, FLiBe use also brings the same issue in many MSR designs of producing significant amounts of tritium. Tritium capture is thus also a major part of FHR work. Closed cycle He or CO<sub>2</sub> Brayton cycles are often proposed which aid in tritium management compared to steam Rankin. Special coatings on heat exchanger walls can also aid. As well, reliance is being placed upon the graphite itself which can be a strong absorber of tritium.

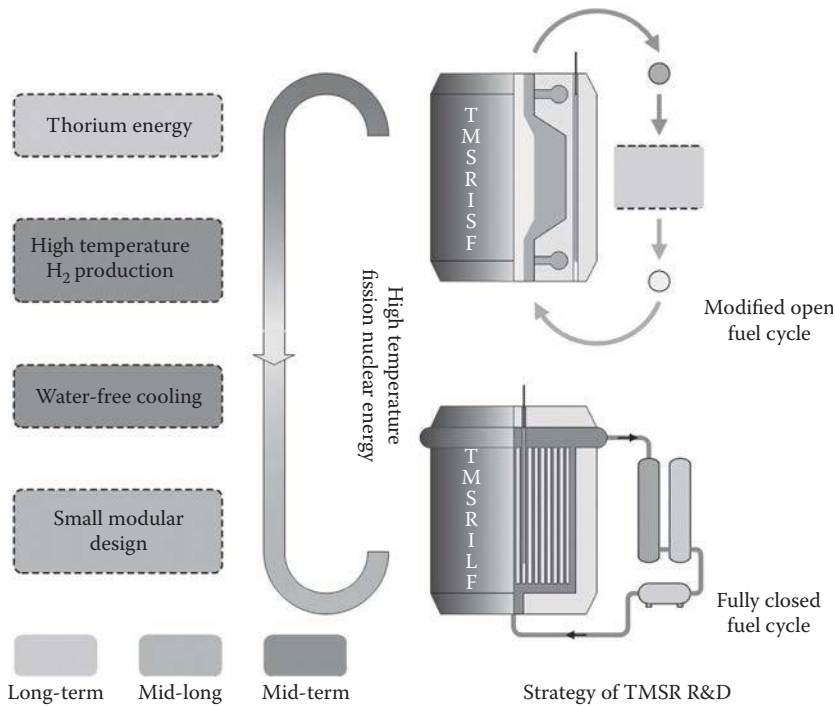
The development of the FHR concept has received modest funding in the United States and is receiving more international attention. This is evidenced by the fact that a Chinese MSR program has adopted development of both salt-cooled FHRs and true salt-fueled MSRs.

### 7.1.5.2 China

China's involvement with MSR concepts surprisingly dates back as far as 1970, as in fact the nation's first test reactor was a zero power MSR operating in 1970–1971 (Xu, 2012). Like other early MSR programs in Russia, India, France, and the United Kingdom who were also following ORNL's lead, international MSR efforts went cold after the US program was canceled. Much has recently changed however.

Under the early direction of Jiang Mianheng, the son of former premier Jiang Zemin, China has initiated a research and development project focusing on the commercialization of MSR technology, officially designated as the thorium MSR (TMSR) program. It was formally announced by the Chinese Academy of Sciences (CAS) in January 2011. The initial budget is reportedly \$350–\$500 million, with over 500 technical personnel being allocated to the project. The initially stated goal of this program was to construct an MSR test reactor by 2017. After receiving technical consulting from US experts, the CAS has aimed toward a pebble bed FHR (solid TRISO fuel, molten salt-cooled) as FHR work is the current focus of US efforts. This test reactor will have a low power rating of 2 MWth and low power density. Early indications were a partitioned, random pebble bed similar to work at University of California, Berkeley, but more recently a simpler ordered array is stated as the prototype goal. While a specific date has not been set, the CAS has announced their plan to build a fluid fuel MSR, to be completed by around 2020. Commercialization of the project is forecast for 2030.

For their FHR efforts, the pebble bed will take advantage of current pebble manufacturing ability and employ standard 6 cm pebbles from Chinese efforts on the helium-cooled HTR-10. Reactor control is to be accomplished entirely by control rods in peripheral locations in a surrounding graphite reflector. An ordered bed requires shutdown for pebble replacement and re-ordering as central pebbles will undergo far higher burn up than in outer layers. Current plans are for a follow up of a larger demonstration unit of 100 MWth



**FIGURE 7.11**  
SINAP 2 MWth FHR.

(45 MWe) with smaller and more optimal 3 cm pebbles and retaining control rods in radial reflector positions (Figure 7.11).

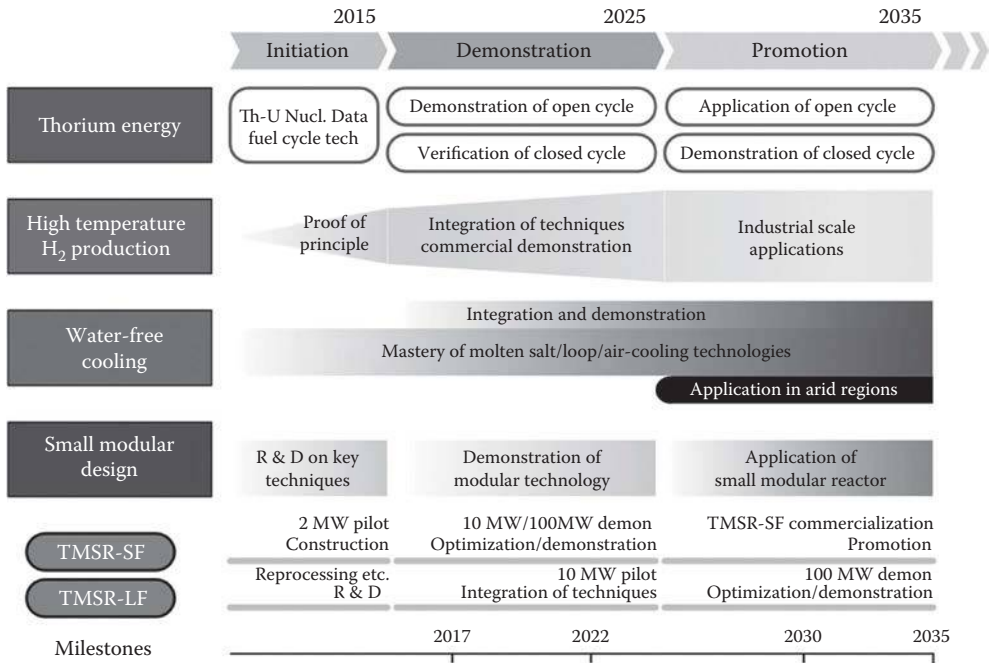
For their first salt-fueled MSR, it appears likely that the Chinese will largely attempt to reproduce the proven design of the 1960s MSRE, with an even lower power of 2 MWth to start. Follow on designs to reach higher power output and commercialization of MSR variants is not yet clear (Figure 7.12).

The Chinese program has been designated as a thorium reactor program. However, there are no current plans to include thorium in either the first FHR or MSR versions. It appears likely that breeder versions are considered a very long-term goal, and as such burner designs running off LEU, with or without the aid of thorium, are the more immediate goals.

### 7.1.5.3 Japan

Japanese interest in MSRs has been a long one, primarily led by the late Kazuo Furukawa with a series of what have been known as FUJI reactor plans, taking much from DMSR design including a 30-year sealed system at low power density for full graphite lifetime and the burner approach without on-site reprocessing. The FUJI approach however has preferred to not provide <sup>238</sup>U for denaturing of the uranium content and to use solely the Th-<sup>233</sup>U cycle. This approach attains a somewhat higher conversion ratio than burner designs using LEU. As FUJI is not meant to function as a breeder, it would require both startup and makeup fuel supplies. While Pu can provide a limited resource, especially for startup, the primary source of makeup fissile has been proposed as accelerator-produced <sup>233</sup>U shipped to individual FUJI plants. Approximately 100–200 kg of <sup>233</sup>U per GWe-year





**FIGURE 7.12**  
Design of the 2 MW Chinese MSR.

would be required. The capital costs needed for an accelerator capable of this production would be significant, and electricity costs alone would cost an estimated \$200 per gram of <sup>233</sup>U produced (MOIR, 2007).

**7.1.5.4 Europe**

In terms of general power reactor designs, the most intensive new efforts outside of China have been in Europe, centered in France where a major modeling, design, and salt chemistry program is taking place. This work has included publication (Nuttin, 2002) of a potential reactivity problem with the traditional single fluid MSBR design. Researchers in the program have reached the conclusion that moving away from graphite moderation will attain the best results. Their latest design offering utilizes a 78%<sup>7</sup>LiF—22%(Th + U)F<sub>4</sub> fuel salt as the core, surrounded by a radial-only blanket of LiF-ThF<sub>4</sub> and axial metallic reflectors. This 3000 MWth design was earlier termed the TMSR (Merle-Lucotte et al., 2007), and now the molten salt fast reactor (Brovchenkao, 2012). The combination of high fissile load (~5 tonnes <sup>233</sup>U) and at least a partial thorium salt blanket results in a high breeding ratio of 1.13 for a 6 month fission product removal rate and the ability to extend this processing time to 20 years and still breakeven.

The fast spectrum approach, while certainly advantageous in potential breeding gain, does involve many significant developmental challenges. Axial reflectors and a radial barrier between core and blanket salt will be in a strong neutron flux, which will likely impose periodic replacement of metallic structures. As well, while some <sup>233</sup>U could be produced in LWR cores, a convenient startup fissile source has long been stated as being spent-fuel Pu. However, in the fast spectrum, this requires on the order of 11 tonnes of Pu at a concentration of about 5.6% molar. Such a concentration is likely beyond the solubility limit of

the salt unless the temperature is kept higher than traditional limits on Hastelloy N™. It is evident that a significant development period with smaller prototype and demonstration reactors is in order. A small demonstration reactor by 2030 and a larger prototype version by 2040 are proposed (SNETP, 2012).

#### **7.1.5.5 Russia**

Russia has had a long-standing interest in MSR designs and intensive program in the 1970s at the Kurchatov Institute (Novikov et al., 1990). Early efforts continued up to 1986 when Chernobyl largely put an end to many advanced reactor project funding in Russia. Besides investigating various novel MSR concepts, Russian scientists also advanced metallurgy in regard to nickel alloys for use with the fluoride fuel salts. More recently, efforts have been led by Victor Ignatiev both with novel TRU burner concepts and in aiding the European MSFR program. For TRU burning, a fast spectrum was chosen for work on MOSART (Ignatiev et al., 2007) in which no fertile Th or <sup>238</sup>U is present, just continuous additions of spent fuel TRUs. The challenge will be to maintain long-term criticality without surpassing the limited solubility of PuF<sub>3</sub> in the carrier salts. The carrier salt, LiF-NaF-BeF<sub>2</sub>, was determined adequate for this goal.

#### **7.1.5.6 India**

With India's well-known program involving heavy water reactors and fast breeders, their early involvement with MSR development is surprising to some. The Bhabha Atomic Research Centre (BARC) was involved with official collaborations with ORNL from 1969 to 1975 on MSBR research. Facilities were built based on salt chemistry and, in particular, studies carried out on PuF<sub>3</sub> solubility remain of great value today (Venugopal, 2013). The 2013 Conference on Molten Salts in Nuclear Technology (CMSNT) held at BARC detailed India's renewed interest in a wide variety of MSR and FHR concepts (<http://moltensaltindia.org/speakers-presentations/>).

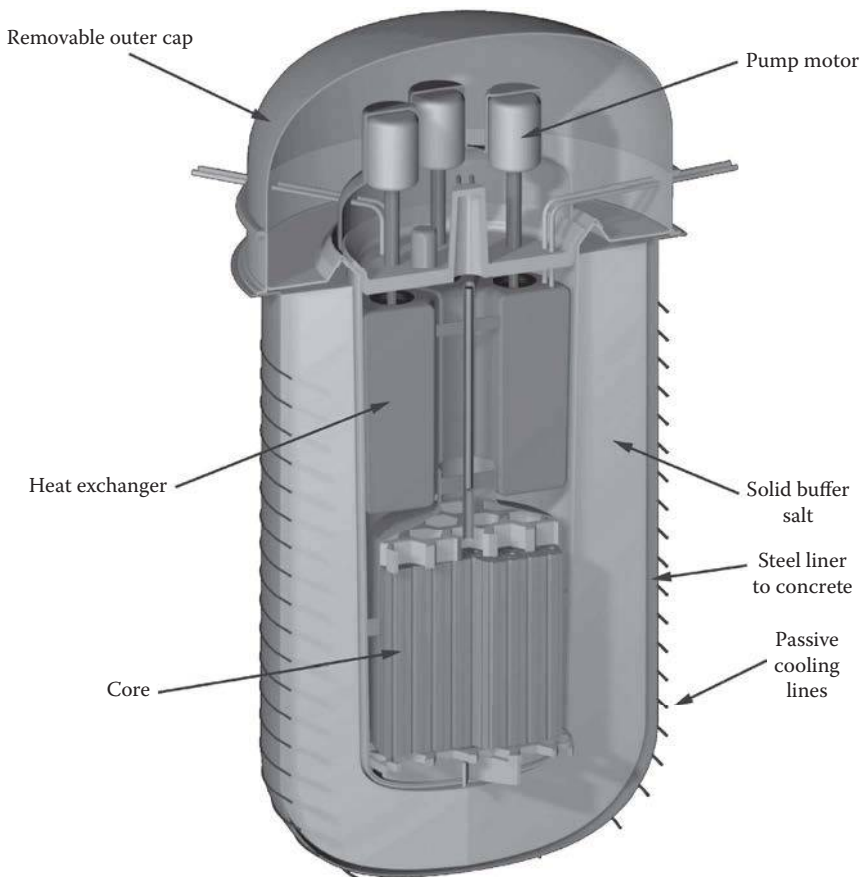
#### **7.1.5.7 Canada**

While Canada saw little involvement with early MSR efforts, Canada may be uniquely placed as a focal point for future efforts with, for example, a wide need for remote industrial heat and power from the far north to the Athabasca oil sands and a regulatory environment which appears comparatively conducive to alternative reactor design. More importantly, with a slowdown in both CANDU development and a future retreat from isotope production at Chalk River Nuclear Laboratories, there is also an enormous nuclear brain trust going underutilized and showing interest in MSR development. Canada is home to Terrestrial Energy, which is advancing its integral MSR (IMSR) design (Irish and LeBlanc, 2014) (Disclosure Note, author is CTO of Terrestrial Energy).

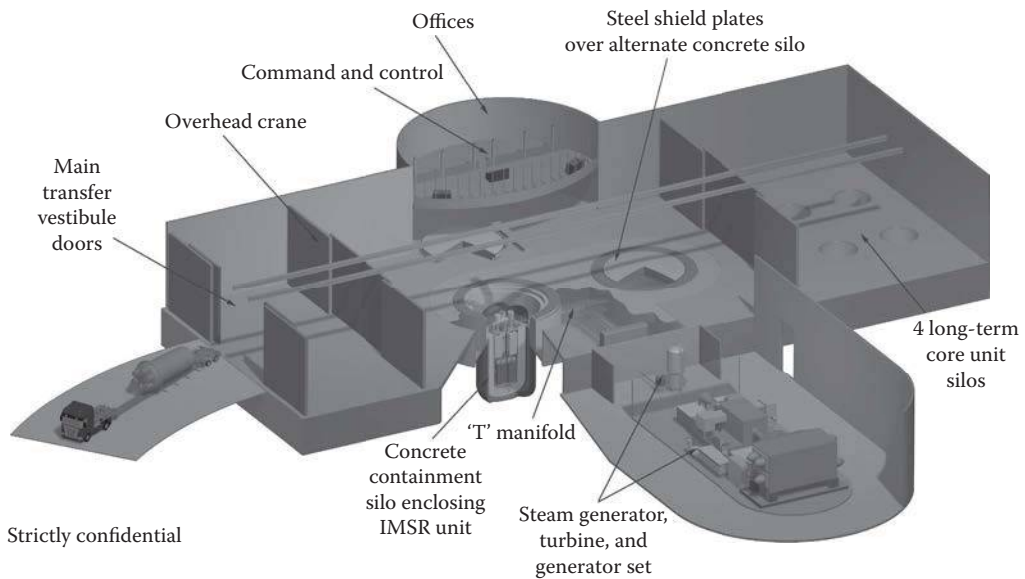
As with any nuclear reactor, when safety areas are fully addressed, the largest remaining challenges revolve around assuring materials and/or components last the full operational lifetime of the reactor or otherwise can be replaced or repaired. Lacking solid fuel or internal core structural material aids but the main components of interest are the reactor vessel itself, primary heat exchangers, and any graphite moderator. In the case of heat exchangers, any metal surface will become quite radioactive due to plate out of noble metals from the fuel salt. In situ repair is unlikely, which means that the ability to swap out tube bundles or entire units is typically planned for. Graphite moderator gives substantial benefit in many

areas but suffers from a limited lifetime proportional to the power density. Historically only two choices appeared possible, going to large sealed low power cores that allowed a full 30 year lifetime of graphite or higher power density but needing to open the reactor periodically to swap out highly radioactive graphite—an extremely challenging prospect.

The main innovation of Terrestrial Energy's IMSR is its novel solution to the long-standing *sealed* versus *swap* dilemma. Stated simply, the IMSR integrates all primary systems into a sealed reactor vessel. This includes the primary heat exchangers and core graphite. The IMSR is run at an economically attractive high power density and can avoid any need to replace graphite or heat exchangers by keeping the entire compact core unit sealed, but replacing this core unit itself. Seven years is the currently planned life cycle with the facility planned for 60+ years. The reactor vessel thus has a second life as a medium- to long-term storage vessel for the low-to-intermediate level spent graphite and heat exchangers. The IMSR employs multiple independent primary heat exchanger units, each with their own secondary coolant inlet and outlet lines. Thus in the event of any HX failure, individual segments can be isolated and the IMSR continue its planned lifetime. The IMSR units are planned in three power outputs from 80 to 600 MW(t). All are planned to be factory fabricated and truck deliverable (Figures 7.13 and 7.14).



**FIGURE 7.13**  
IMSR core unit inside guard vessel.



**FIGURE 7.14**  
IMSR facility layout.

As shown in Figures 7.13 and 7.14, the default design of a facility is to have two containment shells. After the first IMSR unit has completed its design life, a second identical unit is connected in the adjoining bay. The first unit has thus several years of cooldown time before the IMSR unit holding the low level graphite and heat exchanger waste is moved to long-term storage bays. Fuel salt is drained to shielded storage tanks before the core unit is transferred.

By the simplicity of design possible with a burner MSR of a replaceable nature, costs of the primary unit are kept low and the reduced amortization time is of minor concern. Even in more complex previous MSR designs, the primary systems of reactor, pumps, and heat exchangers was typically only 10% of plant costs, which thus bodes well in this respect.

### 7.1.6 Conclusions

MSR research has been largely focused upon breeder designs early on due to government mandate resulting from a mistaken belief in extremely limited world uranium resources. There is still a large draw to the breeder approach partially from an ingrained viewpoint that breeders alone represent long-term resource sustainability. The MSR burner approach however offers perhaps a more pragmatic and commercially expedient choice—a choice that is also sustainable for an almost unlimited period given the expansion of uranium resources possible by simply allowing a commodity price increase—an increase which has a negligible effect on energy production costs. Other benefits often only associated with the breeder approach such as waste profile are shared by the burner approach.

Besides the advantages that burner MSRs share with breeders, there are numerous pluses that are specific to the burner approach. Primarily are those related to system simplicity and cost innovation deriving from the lack of the need for continuous on-site reprocessing and the great optimization possible when losses to neutron economy are no longer

paramount. More specifically attributed to burner MSRs are the important proliferation resistance advantages of running in a denatured state.

In general, MSRs and their cousin technology of FHRs offer a broad array of benefits and potential challenges. For any future seeing a significant expansion of nuclear power, these promising systems should be given a chance to demonstrate their potential. They represent a technology simply too important to leave on the shelf.

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# 8

## *Small Modular Reactors*

**Richard R. Schultz and Kenneth D. Kok**

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As noted in a paper authored by N. G. Trikouros of NRC and published in *Nuclear Technology* in 2012, small modular reactors (SMRs) are not a new idea. Trikouros writes:

...the first generation of nuclear power plants in the United States could be properly categorized as SMRs. The progression from these early SMRs to the increasing power ratings of the current generation of nuclear power plants to the even larger generation III and III+ plants is associated with the economies of scale.

Although most of the first-generation reactors in the United States are no longer operating, throughout that period SMRs were still being constructed internationally. According to the IAEA, there are currently 133 SMRs (<700 MWe) in operation in 28 countries, with 12 more under construction, for a total generating capacity of 60.3 GWe. These SMRs consist of small pressurized water reactors, boiling water reactors, pressurized heavy water reactors, gas-cooled reactors, and fast breeder reactors. Small reactors have also been used in the United States and internationally for many years to power nuclear submarines and to provide electricity for remote military installations.

It should be noted that SMRs of earlier generations in the United States are essentially smaller versions of current-generation light water reactor (LWR) designs and helped to establish the current regulatory framework. Future SMRs that utilize LWR technology will benefit from this in the licensing process. Many of the SMRs of the future are quite different, however, and pose significant licensing and implementation challenges, even the integral LWRs (iLWRs) being proposed...

The early reactors constructed and operated in the United States are listed in Table 8.1.

Today the concept of SMR is being developed using modern technologies with advanced safety systems. The International Atomic Energy Agency (IAEA) defines *small* as under 300 MWe; IAEA also defines a medium-sized reactor as one up to 700 MWe. The driving

**TABLE 8.1**

Early Reactors That Were Operated during the Development of Commercial Nuclear Power

Reactor	Type	Dates of Operation	Fuel	Coolant	Moderator	Electricity Generation
EBR-1	FBR		U235/238 metal	Sodium	NA	45 KWe
EBWR	BWR	1957–1967	Enriched uranium metal	Light water	Light water	4.5 MWe
Dresden	BWR	1960–1978	Enriched uranium oxide	Light water	Light water	200 MWe
Big Rock Point	BWR	1965–1997	Enriched uranium oxide	Light water	Light water	66 MWe
Pathfinder	CRBWR	1966–1967	Enriched uranium oxide	Light water	Light water	66 MWe
Shippingport	PWR	1957–1982	Enriched UO <sub>2</sub>	Light water	Light water	68 MWe
Indian Point	PWR	1963–1976	Mixed UO <sub>2</sub> –ThO <sub>2</sub>	Light water	Light water	275 MWe
CVTR	PHWR	1963–1967	Natural UO <sub>2</sub>	Light water	Heavy water	17 MWe
Calder Hall	GCR	1956–2003	Uranium metal	CO <sub>2</sub>	Graphite	50 MWe
THTR	HTGR	1987–1989	Mixed UO <sub>2</sub> –ThO <sub>2</sub>	Helium	Graphite	296 MWe
Piqua	OCR	1963–1966	Enriched uranium metal	Organic liquid	Organic liquid	12 MWe
Hallam	LMGMR	1963–1964	Molybdenum–uranium alloy	Sodium	Graphite	75 MWe
Fermi Unit 1	LMFBR	1966–1972	Molybdenum–uranium alloy	Sodium	NA	61 MWe

force behind the ideas for reactors of this size is based on several factors including the following:

- Smaller reactors can be produced in a factory and moved as a single unit to the reactor site to be installed. This will shorten the time for construction of the power station and should be less costly.
- If a large power station is needed in the future, it could be made up of multiple small units each being added as the need arises. This is not unlike the manner in which fossil-fired plants were originally built with multiple smaller units.
- Smaller reactors can be used to serve remote areas where large amounts of power are not required and where service via transmission from larger plants is difficult.
- Smaller reactors can more easily be designed to operate with passive safety systems since the residual heat loads are smaller.
- They can also be built underground which would protect them from attack and also make siting possible in more urban areas.

The development of SMRs is in progress across the world by nuclear system suppliers. This development is shown in Table 8.2. The whole industry in this area is in a state of flux and details on specific reactor are changing rapidly. Specific current information of designs being considered is best found in the World Nuclear Association information library.



**TABLE 8.2**

Small Nuclear Reactors at Various Stages of Operation, Construction, and Design Worldwide

Name	Capacity (MWe)	Type	Developer
<i>Small (25 MWe up) Reactors Operating</i>			
CNP-300	300	PWR	CNNC, operational in Pakistan
PHWR-220	220	PHWR	NPCIL, India
<i>Small (25 MWe up) Reactor Designs Under Construction</i>			
KLT-40S	35	PWR	OKBM, Russia
CAREM	27	PWR	CNEA and INVAP, Argentina
HTR-PM	2 × 105	HTR	INET and Huaneng, China
<i>Small (25 MWe up) Reactors for Near-Term Deployment—Development Well Advanced</i>			
VBER-300	300	PWR	OKBM, Russia
IRIS	100–335	PWR	Westinghouse-led, international
Westinghouse SMR	225	PWR	Westinghouse, USA
mPower	180	PWR	Babcock & Wilcox + Bechtel, USA
SMR-160	160	PWR	Holtec, USA
ACP100	100	PWR	CNNC and Guodian, China
SMART	100	PWR	KAERI, South Korea
NuScale	45	PWR	NuScale Power + Fluor, USA
PBMR	165	HTR	PBMR, South Africa; NPMC, USA
Prism	311	FNR	GE-Hitachi, USA
BREST	300	FNR	RDIPE, Russia
SVBR-100	100	FNR	AKME-engineering, Russia
<i>Small (25 MWe up) Reactor Designs at Earlier Stages</i>			
EM2	240	HTR, FNR	General Atomics, USA
VK-300	300	BWR	RDIPE, Russia
AHWR-300 LEU	300	PHWR	BARC, India
CAP150	150	PWR	SNERDI, China
SC-HTGR (Antares)	250	HTR	Areva
Gen4 module	25	FNR	Gen4 (Hyperion), USA
IMR	350	PWR	Mitsubishi, Japan
Fuji MSR	100–200	MSR	ITHMSI, Japan–Russia–USA
EM2	240	HTR, FNR	General Atomics, USA

## 8.1 SMR Designs under Consideration by USNRC

The discussion in this chapter is focused on the SMR designs which form the basis for significant interactions between the designer and the US Nuclear Regulatory Commission (USNRC), that is, the designs developed by mPower, NuScale, and Westinghouse. These designs are light water reactors (LWRs) and have many design features characteristic of the advanced LWRs. A fourth SMR design, the SMR-160 Holtec LWR design, will not be addressed since it is not as mature and has had only very preliminary interactions with the USNRC.

The mPower, NuScale, and Westinghouse designs are closely linked with existing technology and are perceived to have a high technological readiness.\* These designs have the following design characteristics in common:

- The investment costs are expected to be approximately one order of magnitude less than the capital investment presently required for a large LWR nuclear power plant (NPP).†
- Construction schedules are shorter for smaller NPPs, so that interest costs are decreased; thus demand and capacity are better matched.
- SMRs depend on the economy of mass production instead of the economy of scale.
- All primary components are housed within the reactor vessel; thus the designs are known as integral LWRs.
- The power output per unit ranges from 45 to 225 MWe, depending on the designer with a 60-year design lifetime.
- Fuel is standard (UO<sub>2</sub> enriched up to 5 wt% U-235) and fuel assembly design (but with shorter fuel pins).
- No boron is used and they have long irradiation cycles (i.e., up to 4 years).
- All safety systems are passive and function principally via density gradient-driven flow and/or compressed gas-driven accumulators. To ensure the functioning of density gradient-driven flow provided by gravity-drain tanks, automatic depressurization systems are also included to depressurize the reactor to operational pressure levels for these systems.
- Their containments are small, high pressure, and located underground.
- These designs are factory-built in the United States and may be shipped by rail, barge, or ship.
- These designs use an in-vessel retention approach, that is, the reactor pressure vessel cavity is flooded and the decay heat is removed by boiling on the outer surface of the reactor pressure vessel.

The power produced by each unit module for mPower, NuScale, and Westinghouse is about 150, 45, and greater than 225 MWe, respectively (NuScale Power 2014; Westinghouse Power 2014). Each of these designs may be grouped into NPPs with two or more modules. mPower shows their modules grouped in twin packs, while NuScale shows their modules in groups of as many as 12.

To minimize the core damage frequency, the mPower, NuScale, and Westinghouse designs have the following safety features:

- Low core linear power to minimize core temperatures during challenging design basis events.
- Low fluid velocities to minimize flow-induced vibrations and enhance the component lifetime.
- Large coolant volume to core power ratio to maximize the time available for safety system response during design basis events.

\* Technology readiness = The degree to which a system is ready for commercialization.

† Capital investment for a large LWR NPP is on the order of one billion dollars, depending on the number of units and NPP type.

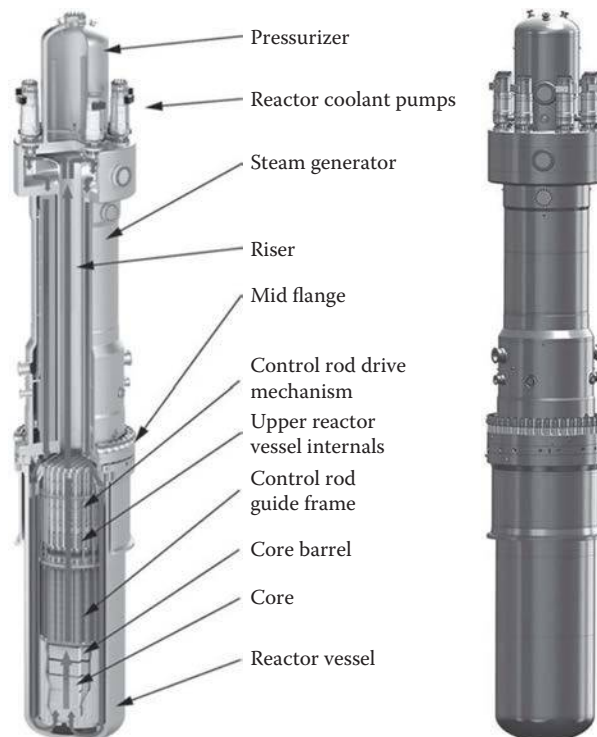
- Large pipes are eliminated by design, thus eliminating the large-break loss of coolant accident (LOCA) design basis accident.
- Penetrations to the reactor vessel are all small in diameter and located at elevations significantly above the core to increase the quantity of water available to cool the core. Small penetrations also minimize the energy loss to the containment for pipe leaks, resulting in lower containment pressures during design basis events.
- Automatic depressurization systems enable quick depressurization of the injection pressure of gravity force-driven systems.

A summary of key design features of the mPower, NuScale, and Westinghouse SMRs follows.

## 8.2 mPower Design

The mPower design features, as given in the following publicly available documents: Buongiorno (2012), USNRC (<http://www.nrc.gov/reactors/advanced.html>), and B&W publicly available information (<http://www.babcock.com/products/Pages/mPower-Reactor.aspx>) forms the basis for the following description.

The mPower design is rated at 150 MWe with all major reactor components contained within or located immediately adjacent to the reactor vessel as shown in Figure 8.1.



**FIGURE 8.1**  
mPower reactor vessel showing relative locations of components.

The pressurizer, steam generators, and core are located within the reactor vessel. Although the reactor vessel design has a tall riser section above the core, the core flow is driven by reactor coolant pumps under operational conditions. The reactor coolant pumps are located outside the reactor vessel, although the pump impellers are located within the pressure boundary of the reactor vessel. Key design parameters include the core average linear heat rate in KWth/m, reactor coolant system (RCS) volume to the power ratio ( $\text{m}^3/\text{MWth}$ ), and the RCS volume to LOCA area ratio ( $\text{m}^3/\text{m}^2$ ). Table 8.3 gives characteristic values of these key parameters for mPower in comparison to a typical Generation III/III+ design. Notice the core average linear heat rate is considerably smaller, while the RCS volume to power ratio and to LOCA area ratio are substantially larger. These parameters are indicative of the enhanced safety margin characteristics of the mPower design.

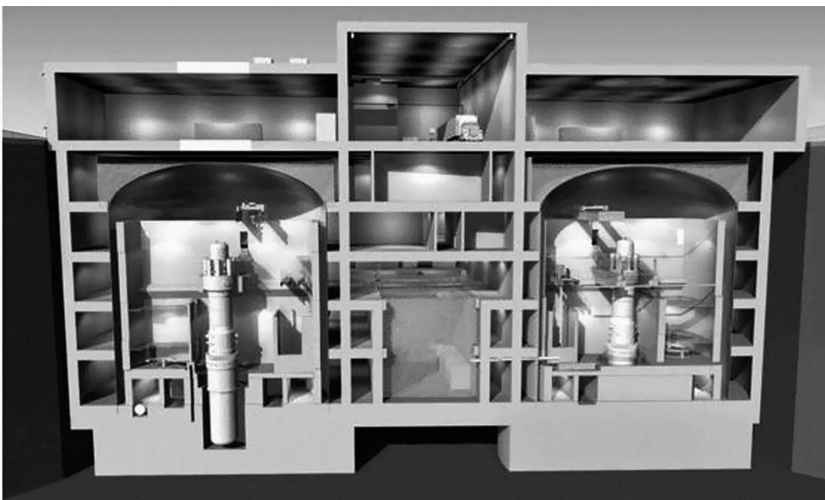
The design itself lends to a *twin-pack* plant configuration that has a 40-acre footprint, low profile architecture, and underground containment and an underground spent fuel pool—as shown in Figure 8.2.

The mPower balance-of-plant components are conventional.

**TABLE 8.3**

Comparison—Safety Features mPower to Typical Generation III/III+ Design

Parameter	mPower	Typical Generation III/III+ System
Core average linear heat rate (KWth/m)	11.5	18.7
RCS volume to power ratio ( $\text{m}^3/\text{MWth}$ )	0.18	0.08
RCS volume to LOCA area ratio	13,700	270



**FIGURE 8.2**  
mPower twin-pack plant configuration.

---

### 8.3 NuScale Design

The NuScale design features, as given in the following publicly available documents: Buongiorno, USNRC (<http://www.nrc.gov/reactors/advanced.html>), and NuScale publicly available information forms the basis for the following description.

NuScale power plants are scalable meaning a single facility may have from 1 to 12 modules. Figure 8.3 shows a grouping of 12 NuScale modules to create an NPP with a total power output of 540 MWe. Each of the modules is self-contained and operates via natural circulation at operational conditions. A typical 45 MWe module is shown in Figure 8.4. The reactor vessel for each module is fixed within a containment vessel which is located within a water-filled pool that is below ground. The water-filled pool is a steel-lined concrete structure with a 30-day supply of cooling water. The operating pressure is only 8.7 MPa.

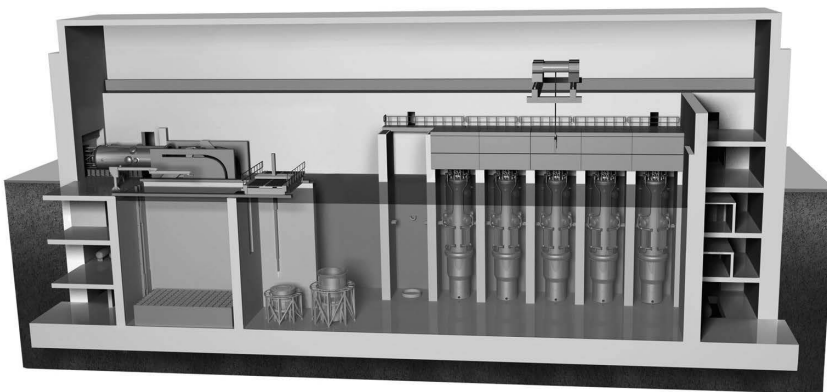
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### 8.4 Westinghouse Design

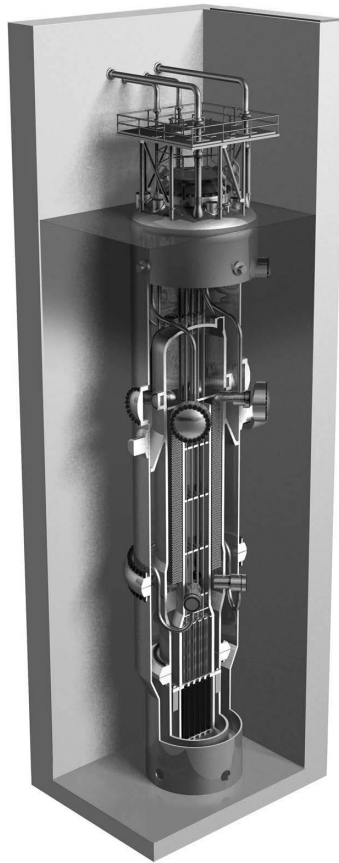
The Westinghouse design features, as given in the following publicly available documents: Buongiorno, USNRC (<http://www.nrc.gov/reactors/advanced.html>), and Westinghouse publicly available information forms the basis for the following description.

The 225 MWe Westinghouse design was specifically aimed to reverse the traditional decrease in cost of electrical power generation with increasing plant size paradigm. This objective was achieved by reducing the size of key components to dimensions even less than that projected using power scaling.

The Westinghouse SMR is designed to produce one-fifth of the power of the AP1000 plant, yet 25 Westinghouse SMR containment vessels will fit within one AP1000 containment vessel. The smaller SMR containment vessels will experience less wall stresses associated with the smaller diameter pressure vessel.



**FIGURE 8.3**  
Artist's rendering of 12-module, 540-MWe NuScale plant.

**FIGURE 8.4**

NuScale power module: reactor pressure vessel and containment vessel located within steel-lined concrete pool.

The Westinghouse SMR is designed to operate using reactor coolant pumps although the RCS harbors a very tall riser component above the core similar to the risers in both the mPower and NuScale designs. The reactor vessel located within the containment vessel is shown in Figure 8.5.

The core uses  $17 \times 17$  fuel assemblies with 8 ft long active lengths. The pressurizer and steam generators are integral to the RCS.

---

## 8.5 Summary

The SMR designers actively interacting with the USNRC, mPower, NuScale, and Westinghouse have many common design features. From a high-level perspective the LWR SMRs may be appealing to utilities since their technology is based on proven technology already used in the industry, the initial cost of construction is considerably less than a large- or medium-sized LWR, and they have a number of exciting improvements that enhance their appeal to stakeholders—such as enhanced safety.



**FIGURE 8.5**  
Westinghouse reactor pressure vessel within containment vessel.

---

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# 9

## *Generation IV Technologies*

Edwin A. Harvego and Richard R. Schultz

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The development of nuclear power occurred in three general phases. The initial development of prototype reactor designs (Generation I) occurred in the 1950s and 1960s, development and deployment of large commercial plants (Generation II) occurred in the 1970s and 1980s, and development of advanced light water reactors (LWRs; Generation III) occurred in the 1990s.

The earlier generations of reactors have effectively demonstrated the viability of nuclear power, but the nuclear industry faces several challenges that must be addressed for nuclear power to achieve its full potential. Among these challenges are (1) public concern about the safety of nuclear power in the wake of the Three Mile Island accident in 1979 and the Chernobyl accident in 1986; (2) high capital costs and licensing uncertainties associated with the construction of new nuclear power plants; (3) public concern over potential vulnerabilities of nuclear power plants to terrorist attacks; and (4) issues associated with the accumulation of nuclear waste and the potential for nuclear material proliferation in an anticipated environment of expanding nuclear power production. However, concerns over rising energy demand and shrinking oil reserves, along with concerns over the potential impact of fossil fuel emissions on global warming, have made the nuclear option (which does not emit greenhouse gases) more attractive.

To address public concerns and to fully realize the potential contributions of nuclear power to future energy needs in the United States and worldwide, the development of a new generation of reactors, termed Generation IV, was initiated in 2001. The intent or objective of this effort is to develop multiple Generation IV nuclear power systems that would be available for international deployment before the year 2030. The development of Generation IV reactor systems is an international effort, initiated by the US Department of Energy (DOE) with participation from 13 members. These members established a formal organization referred to as the *Generation IV International Forum* (GIF). The early GIF members included Argentina, Brazil, Canada, China, Euratom, France, Japan, the Republic of Korea, the Republic of South Africa, Russia, Switzerland, the United Kingdom, and the United States, but it is anticipated that this membership will grow as the program develops. The intent of the GIF is "...to develop future-generation nuclear energy systems that can be licensed, constructed, and operated in a manner that will provide competitively priced and reliable energy products while satisfactorily addressing nuclear safety, waste, proliferation, and public perception concerns."

The process used by the GIF to identify the most promising reactor concepts for development (referred to as the *Generation IV Technology Roadmap*) consisted of three steps. These steps were to (1) develop a set of goals for new reactor systems, (2) solicit proposals from the worldwide nuclear community for new reactor systems to meet these goals, and (3) evaluate the different concepts to select the most promising candidates for further development using experts from around the world.

The eight goals developed by the GIF for Generation IV nuclear systems<sup>1</sup> are detailed below.

- *Sustainability-1*: Generation IV nuclear energy systems will provide sustainable energy generation that meets clean air objective and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.
- *Sustainability-2*: Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long-term stewardship burden in the future, thereby improving protection for the public health and the environment.
- *Economics-1*: Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.
- *Economics-2*: Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.
- *Safety and reliability-1*: Generation IV nuclear energy systems operations will excel in safety and reliability.

- *Safety and reliability-2*: Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.
- *Safety and reliability-3*: Generation IV nuclear energy systems will eliminate the need for offsite emergency response.
- *Proliferation resistance and physical protection-1*: Generation IV nuclear energy systems will increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials and provide increased physical protection against acts of terrorism.

Approximately 100 Generation IV candidates were evaluated by experts from the GIF countries. Six reactor systems were selected for further evaluation and potential development:

- Gas-cooled fast reactor (GFR)
- Very-high-temperature reactor (VHTR)
- Supercritical-water-cooled reactor (SCWR)
- Sodium-cooled fast reactor (SFR)
- Lead-cooled fast reactor (LFR)
- Molten salt reactor (MSR)

Of these six, the first five concepts have been initially pursued at varying levels of effort within the United States based on the technology status and the ability of the concepts to meet national energy needs. Because of its limited technology base, further development of the MSR is not being pursued in the United States, but may be pursued by other countries. However, as these concepts continue to be developed, it is anticipated that future research within the United States will become focused on only one or two of the most promising concepts.

Each of these reactor concepts is discussed further in the following sections, which include a brief discussion of the design, an assessment of how each concept is expected to meet the Generation IV goals, the current status of the technologies, and future research and development needs to bring the technologies to fruition.

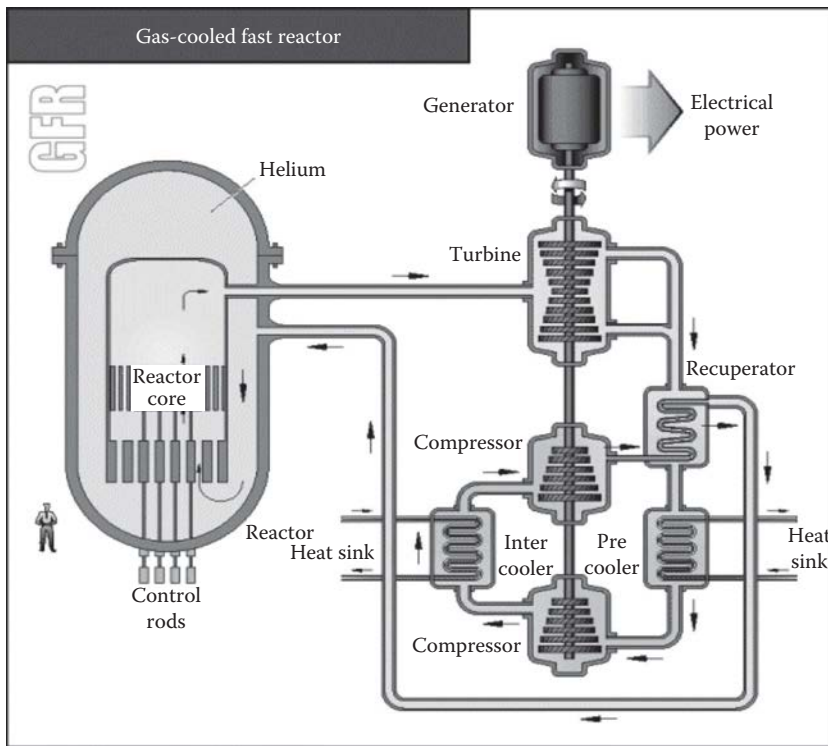
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## 9.1 GFR System

The GFR is a fast neutron spectrum reactor that uses helium as the primary coolant. It is designed to operate at relatively high helium outlet temperatures, making it a good candidate for the high-efficiency production of electricity or hydrogen. The fast neutron spectrum will also enable more effective management of actinides through recycling of most components in the discharged fuel. The combination of these attributes makes the GFR an attractive option for continued evaluation in the United States.

### 9.1.1 GFR Description

A direct Brayton cycle is used for the production of electricity with the helium gas delivered from the reactor outlet to a high-temperature gas turbine connected to a generator that produces electricity (Figure 9.1). In alternative designs, high-temperature helium can



**FIGURE 9.1**

Gas-cooled fast reactor. (From US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, A technology roadmap for generation IV nuclear energy systems, GIF-002-00, 2002.)

also be used to produce hydrogen using a thermochemical process or high-temperature electrolysis, or for other high temperature-process heat applications.

The reference plant is designed to produce 600 MWt and 288 MWe using the direct Brayton cycle with a reactor outlet temperature of 850°C. This results in a net plant efficiency of 48% with an average core power density of 100 MWt/m<sup>3</sup>. The fuel forms being considered for high-temperature operation include composite ceramic fuel, advanced fuel particles, or ceramic clad elements of actinide compounds. Alternative core configurations include prismatic blocks, pin- or plate-based assemblies. The GFR's fast neutron spectrum also makes it possible to efficiently use available fissile and fertile materials in a once-through fuel cycle.

### 9.1.2 GFR Achievement of Generation IV Goals

In terms of achievement of the Generation IV goals described above, the fast neutron spectrum, efficient actinide management, and the ability to close the reactor fuel cycle (recycle used fuel) gives the GFR a high rating for the first two Generation IV goals of sustainability. The benign nature of the helium coolant used in this reactor concept and the high power conversion efficiency also contribute to the achievement of the safety and economic goals for this reactor concept. To address the Generation IV proliferation resistance and physical protection goals, the GFR utilizes an actinide recycle strategy with on-site fuel cycle facilities to minimize the need for off-site transport of nuclear materials.

### 9.1.3 Basis for Development of GFR

The technology base for the development of the GFR includes a number of past and current thermal-spectrum gas reactors and a limited number of fast-spectrum gas-cooled reactor designs. Past reactor designs include the Dragon Project, built and operated in the United Kingdom; the AVR and the THTR, built and operated in Germany; and Peach Bottom and Fort St. Vrain, built and operated in the United States. Current reactor demonstration plants include the 30-MWt HTTR reactor in Japan and the 10-MWth HTR-10 reactor plant in China. High-temperature thermal-spectrum gas reactors currently being designed include the 300 MWth pebble-bed reactor in South Africa and the 300-MWt GT-MHR prismatic core design being developed in the United States by General Atomics.

### 9.1.4 GFR R&D Requirements for Commercial Development

The R&D requirements for the GFR include many of the same needs as the current thermal-spectrum high-temperature gas reactors currently under development. However, there are also additional R&D needs to bridge the technology gap between the thermal-spectrum and fast-spectrum gas reactor concepts.

The major R&D needs that the fast-spectrum and thermal-spectrum reactors have in common are associated primarily with the high operating temperatures of these reactors. These R&D needs include the demonstration of fuel forms that will maintain their integrity and retain fission products under accident or off-normal operating conditions, the development of high-temperature materials for the core and external structural components, development of high-performance gas turbines for efficient generation of electricity at high temperatures, and development of high-temperature heat exchangers for improved cycle efficiency and efficient heat transfer for process heat applications.

R&D needs unique to the fast-spectrum GFR include development and demonstration of safety features, and decay heat removal systems to accommodate the high core power density and low thermal inertia of the core, optimization of fuel forms for the fast-neutron spectrum, development of materials that are resistant to fast-neutron influence under very high temperature conditions, efficient conversion without fertile blankets, and the demonstrated integration of an on-site spent fuel treatment and prefabrication process that is simple and compact.

Safety features of the GFR may include passive and active safety systems that will require new analysis tools for design and safety evaluations. These analysis tools will in turn require verification and validation with a combination of separate effects and integral experimental programs that will be developed specifically to address phenomena unique to the GFR.

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## 9.2 VHTR

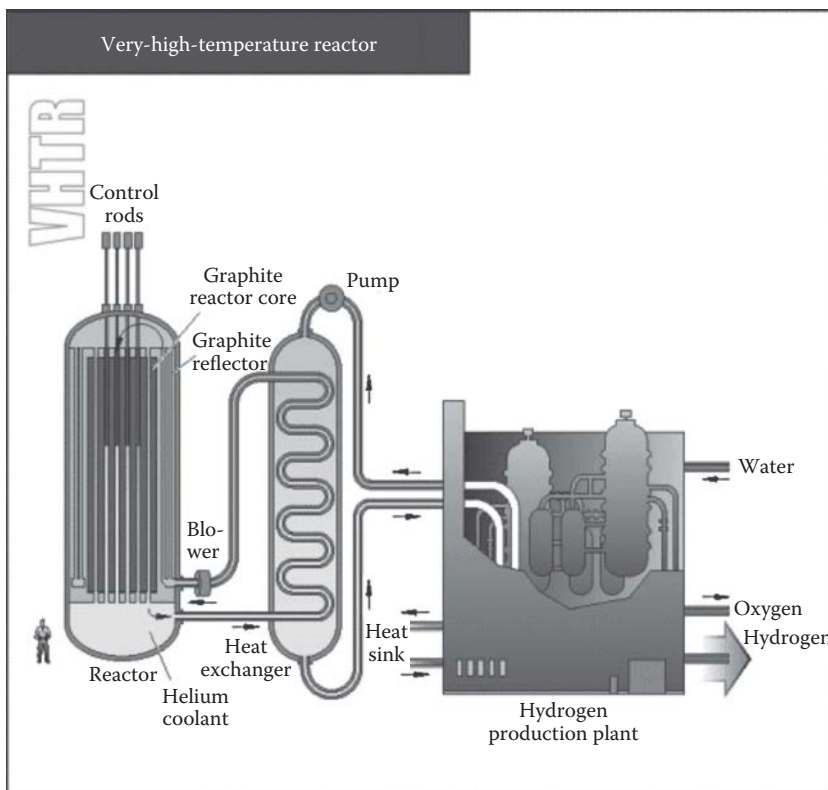
The VHTR is a helium-cooled reactor designed to provide heat at very high temperatures, in the range of 950°C, for high-temperature process heat applications. In particular, the 950°C reactor outlet temperature makes it a good candidate for the production of hydrogen using thermochemical or high-temperature electrolysis processes. The VHTR represents an evolutionary development from predecessor gas-cooled reactor designs

like Peach Bottom and Fort St. Vrain built and operated in the United States. This previous experience, along with the potential for the efficient production of electricity and hydrogen, makes this concept an attractive option for continued development in the United States.

### 9.2.1 VHTR Description

Heat for the production of hydrogen is delivered through an intermediate heat exchanger that serves to isolate the reactor system from the hydrogen production process (Figure 9.2).

The reference design for the VHTR is a 600 MWt reactor with an outlet temperature of 950°C. The reactor core uses graphite as a moderator to produce the thermal neutrons for the fission process. The core configuration can be graphite blocks or pebbles about the size of billiard balls in which fuel particles are dispersed. For electricity production, a direct Brayton cycle gas turbine using the primary helium coolant as the working fluid, or an indirect cycle using a secondary working fluid can be used. The high temperature characteristics of this reactor concept also make it an ideal candidate for cogeneration applications to meet electricity and hydrogen production or other high-temperature process heat needs.



**FIGURE 9.2**

Very-high-temperature reactor. (From US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, A technology roadmap for generation IV nuclear energy systems, GIF-002-00, 2002.)

### 9.2.2 VHTR Achievement of Generation IV Goals

The VHTR utilizes a once-through open fuel cycle and is therefore not intended to address Generation IV sustainability goals. However, its capability to produce electricity at high power conversion system efficiencies, as well as provide high-temperature thermal energy for high-temperature process heat applications, such as the production of hydrogen or the generation of synthetic gas from coal, gives it a high rating for the Generation IV economic goals. The inherently safe design features of VHTR also give it a high rating for Generation IV safety goals. Finally, the general design of the reactor and its flexible fuel cycle addresses the Generation IV proliferation resistance and physical protection goal.

### 9.2.3 Basis for Development of VHTR

Compared with other advanced reactor concepts, the basis for initial development of the graphite-moderated helium-cooled VHTR with thermal neutron spectrum is quite compelling. In addition to the past construction and operating experience of Peach Bottom and Fort St. Vrain in the United States, the Dragon Project in the United Kingdom, and the AVR and THTR in Germany, demonstration high-temperature gas-cooled reactors are now operating in Japan and China. The 30-MWt Japanese HTTR demonstration plant employs a prismatic block core design, and the 10-MWt Chinese HTR-10 demonstration plant utilizes a pebble-bed reactor core design. Either of these core configurations could be incorporated into the VHTR concept. Therefore, a relatively strong technology base exists for the evolutionary development of the VHTR.

### 9.2.4 VHTR R&D Requirements for Commercial Development

The R&D requirements for the VHTR include the development of fuels and materials to meet the high-temperature operating conditions with reactor outlet temperatures between 850°C and 950°C during normal operation and maximum fuel temperatures reaching 1600°C under accident conditions. Other R&D needs include the development and demonstration of fuels that can achieve maximum fuel burnup of 150–200 GW days/metric ton of heavy metal (GWD/MTHM). Additional operational issues to be addressed by R&D include avoiding excess power peaking and temperature gradients in the core, as well as hot streaks in the coolant gas. During accident conditions, the ability of passive safety features and heat-removal systems to effectively limit fuel and primary system temperatures to acceptable values must be demonstrated. In addition, balance-of-plant performance and safe operation must be evaluated for electricity and/or hydrogen production applications.

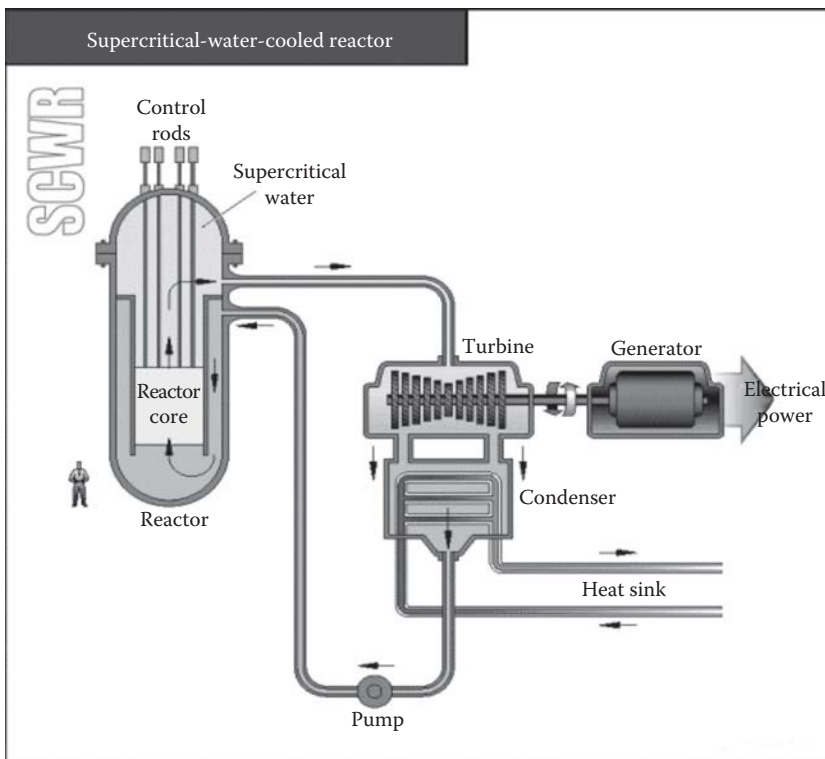
To meet these R&D needs, the extension and validation of existing analysis tools will be required to include new materials, operating regimes, and component configurations into the models. New analysis tools will also be required for the analysis of the balance-of-plant when the VHTR is used for hydrogen production or for co-generation applications requiring electricity productions and process heat. The extension to current modeling capabilities and the development of new modeling capabilities will require that analysis tools be verified and validated using separate effects and integral test data. While some data may already exist, it is anticipated that new experimental data will need to be developed to address specific phenomena that may occur in the VHTR.

### 9.3 SCWR

The SCWR is a relatively high-temperature, high-pressure reactor designed to operate above the thermodynamic critical point of water (374°C and 22.1 MPa). Although there is very little experience in the United States relating to the design and operation of an SCWR, the relative simplicity of the design and the potential for higher net electric power production efficiency (44% for the SCWR compared with 33%–34% for current commercial pressurized water reactors [PWRs]), make it an attractive option for continued evaluation in the United States.

#### 9.3.1 SCWR Description

The balance-of-plant design (Figure 9.3) utilizes a relatively simple direct cycle power conversion system. The reference design for this concept is a 1700-MWe reactor operating at a pressure of 25 MPa with a reactor outlet temperature between 510°C and 550°C. This reactor can be designed as a fast neutron spectrum or thermal neutron spectrum reactor. The relatively simple design also allows for the incorporation of passive safety features. However, unlike the previously discussed concepts, the lower reactor outlet temperature



**FIGURE 9.3**

Supercritical-water-cooled reactor. (From US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, A technology roadmap for generation IV nuclear energy systems, GIF-002-00, 2002.)



is not well suited for the efficient production of hydrogen, which requires minimum temperatures in the range of 850°C–900°C. Therefore, this reactor concept is primarily intended for the efficient, low-cost production of electricity.

### 9.3.2 SCWR Achievement of Generation IV Goals

If successfully developed and deployed, the fast-spectrum version of the SCWR would effectively address the Generation IV sustainability goals because it would allow for effective utilization of fuel in a closed fuel cycle. In addition, because of its simple design and potential ability to produce electric power at low costs, the SCWR would be ranked high in addressing the Generation IV economic goals. The general design characteristics of the SCWR also make it an attractive candidate for addressing the Generation IV safety goals and the proliferation and physical protection goal.

### 9.3.3 Basis for Development of SCWR

The technology base for development of the SCWR relies primarily on technologies developed for commercial LWRs and on technologies developed for the supercritical-water fossil-fired power plants. Because of similarities in primary system design between the SCWR and commercial PWRs, it is anticipated that current LWR technology will be applicable in this area. For the balance-of-plant, including turbine generators, piping and other components, it is expected that some of the technology developed for commercial supercritical-water fossil-fired power plants can be transferred to the development of the SCWR. However, despite the availability of this technology base, there have been no prototype SCWR plants built or tested to date.

### 9.3.4 SCWR R&D Requirements for Commercial Development

The R&D requirements for commercialization of the SCWR include addressing issues associated with SCWR materials and structures, safety and licensing considerations, and plant design issues.

Material and structural issues to be addressed are primarily related to the potential for corrosion and stress corrosion cracking under irradiation at the high temperatures and pressures associated with the SCWR. Materials for cladding and structural components must be identified and tested to demonstrate their performance in thermal and fast-spectrum reactors. Radiolysis and water chemistry at supercritical conditions must be investigated to understand the effect on reactor materials. Specific material properties to be investigated include dimensional and microstructure stability, and strength, embrittlement, and creep resistance characteristics of the materials.

Safety and licensing R&D should include power-flow stability assessments, critical flow evaluations and modeling, development of heat transfer correlations, fuel rod ballooning and rupture behavior during accidents, and measurement and prediction of important thermal-hydraulic phenomena occurring during loss-of-coolant accidents. This R&D effort will require separate effects and integral experimental programs to understand the important phenomena and develop appropriate correlations. These same experiments will also be used for the verification and validation of new or modified computer codes developed for the evaluation of the SCWR design and performance of safety analyses.

Although many of the SCWR components and systems may be derived from currently operating commercial BWRs, PWRs, and supercritical-water fossil plants, there is still

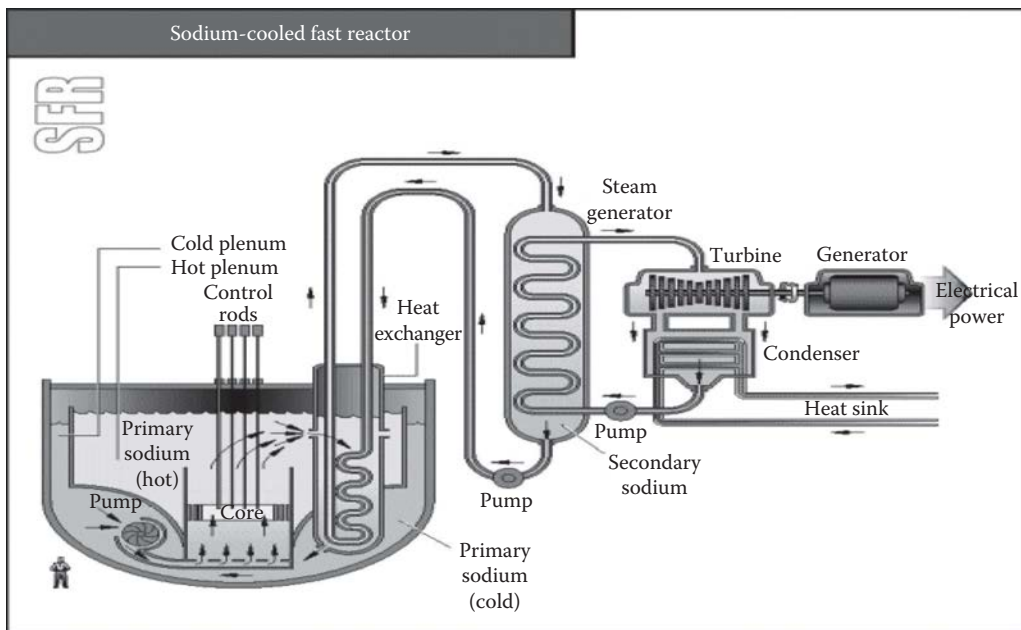
a need for R&D in the optimization of the unique features associated with the SCWR. Examples of design features to be optimized include the fuel assemblies, control rod drive system, internal vessel structures, reactor vessel, pressure relief valves, coolant cleanup system, reactor control logic, turbine configuration, re-heaters, de-aerator, start-up system and procedures, in-core sensors, and the containment building. For the fast-spectrum reactor option, development and demonstration of an optimized fuel cycle also represents a significant R&D effort.

## 9.4 SFR

The SFR is a sodium-cooled fast-neutron-spectrum reactor designed primarily for the efficient management of actinides and conversion of fertile uranium in a closed fuel cycle.

### 9.4.1 SFR Description

Figure 9.4 shows the basic configuration of the SFR. Two reference designs to support different fuel reprocessing options have been defined for this concept. The first is a medium-sized sodium-cooled reactor with a power level of between 150 and 500 MWe that utilized uranium–plutonium–minor-actinide–zirconium metal alloy fuel. This reactor concept is supported by a fuel cycle based on pyrometallurgical processing in which the processing facilities are an integral part of the reactor plant design.



**FIGURE 9.4**

Sodium-cooled-fast reactor. (From US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, A technology roadmap for generation IV nuclear energy systems, GIF-002-00, 2002.)

The second reactor reference design is a large sodium-cooled reactor with a power output capability between 500 and 1500 MWe that utilizes uranium–plutonium oxide fuel. This reactor design is supported by a fuel cycle based on an advanced aqueous process that would include a centrally located processing facility supporting a number of reactors.

Both versions of this reactor concept would operate at coolant outlet temperatures in the range of 550°C and are intended primarily for the management of high-level waste and the production of electricity. In addition to design innovations to reduce capital costs, these reactors incorporate several enhanced safety features: long thermal response time, a large margin to coolant boiling, a primary system that operates near atmospheric pressure, and an intermediate sodium system between the radioactive sodium in the primary system and the water and steam in the power plant.

#### **9.4.2 SFR Achievement of Generation IV Goals**

The SFR is ranked high in addressing the two Generation IV sustainability goals because of its potential for effectively managing actinides in a closed fuel cycle. Potential design innovations to reduce initial capital costs will help to address the two Generation IV economic goals. The enhanced safety features, including long thermal response time of the core, large margin to coolant boiling, primary system operation at atmospheric pressure, and an intermediate sodium loop between the reactor primary system and steam power system will address the three Generation IV safety and reliability goals. It is anticipated that additional design innovations can be developed to address the proliferation resistance and physical protection goal.

#### **9.4.3 Basis for Development of SFR**

The technology base for the SFR is considerably more advanced than that for many of the other Generation IV reactor concepts. SFRs have been built and operated in France, Japan, Germany, the United Kingdom, Russia, and the United States. Sodium-cooled reactors are currently in operation in Japan, France, and Russia. As a result of this experience the R&D needs for the SFR are primarily related to demonstration and optimization of performance and safety characteristics. The databases for the two fuel options being considered (mixed oxide [MOX] and metal fuels) are substantial, but the database for the metal fuels is less extensive than that for MOX fuels. The two fuel cycle options, namely the advanced aqueous process and the pyroprocess, are also based on well-established technologies. For example, the advanced aqueous process is derived from the PUREX process that has been utilized by several countries, and the pyroprocess has been under development in the United States since the inception of the integral fast reactor program in 1984.

#### **9.4.4 SFR R&D Requirements for Commercial Development**

Although the SFR is supported by a relatively strong technology base, there are several areas where additional R&D is needed: (1) demonstration of safety characteristics for design basis and bounding events, (2) demonstration of the pyroprocess at commercial scales, and (3) demonstration of fabrication processes with remote operation and maintenance.

R&D to ensure the passive safety response of the SFR requires verification through testing and analysis that passive safety features and mechanisms are effective in responding to design basis transients, including anticipated transients without scram. For bounding events in which fuel failure might occur, testing would also be required to show that the fuel remains coolable within the reactor vessel, and that recriticality of the damaged

core cannot occur. This will require separate effects and integral experiments to confirm important design features and phenomena, and to validate software used in the design, development, and safety analyses of the SFR.

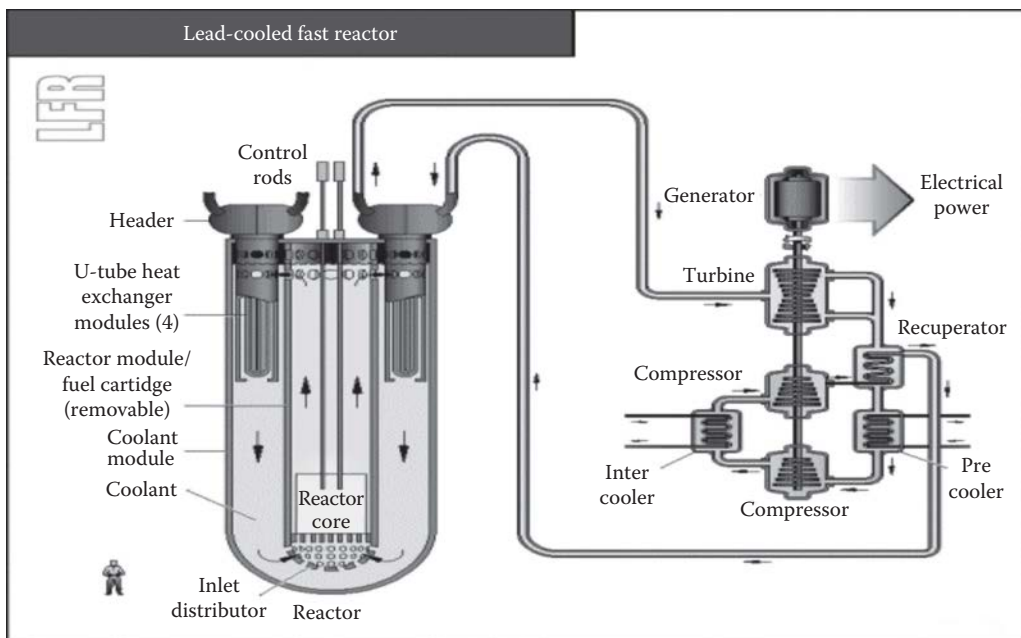
The recovery of plutonium and minor actinides with the pyroprocess is limited to laboratory-scale experiments. The recovery of plutonium and minor actinides needs to be demonstrated at larger commercial scales. In addition, the fission products and minor actinides in the fuels dictate remote fabrication of the fuel forms. Therefore, it is necessary to demonstrate on a large scale the viability of the process from an operations and maintenance standpoint and ensure that the remotely fabricated fuel performs as expected in the reactor.

## 9.5 LFR

The LFR is a fast-neutron spectrum reactor cooled by molten lead or a lead–bismuth eutectic liquid metal. It is designed for the efficient conversion of fertile uranium and the management of actinides in a closed fuel cycle.

### 9.5.1 LFR Description

The reactor core for this design (Figure 9.5) utilizes a metal- or nitride-based fuel containing fertile uranium and transuranics. The LFR relies on natural convection to cool the reactor core (Figure 9.5). The outlet temperature for the current reactor concept is about



**FIGURE 9.5**

Lead-cooled fast reactor. (From US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, A technology roadmap for generation IV nuclear energy systems, GIF-002-00, 2002.)

550°C, but with advanced materials, reactor outlet temperatures of 800°C may be possible. An indirect gas Brayton cycle is used to produce electrical power.

There are three versions of the reference design for this concept. The smallest design, rated at 50–150 MWe, is intended for distributed power applications or electricity production on small grids. This reactor design, referred to as a *battery*, features modular design with a factory fabrication *cassette* core. The reactor is designed for very long refueling interval (15–20 years), with refueling accomplished by replacement of the cassette core or reactor module.

The other two versions of this design are a modular system rated at 300–400 MWe and a large plant rated at 1200 MWe. The different power options for this design are intended to fill different need or opportunities in the power market, and be economically competitive with comparable alternative power sources.

### 9.5.2 LFR Achievement of Generation IV Goals

The LFR is rated high in achieving the Generation IV sustainability goals because it can be used in a closed fuel cycle. Economic goals are addressed by the modular nature of the designs and the option to employ different reactor plant sizes to match the different market needs. The safety goals are addressed primarily by the use of a relatively benign Pb or Pb–Bi liquid alloy reactor coolant. The ability of the different LFR designs to meet the Generation IV proliferation resistance and physical protection goal is considered high because of the design of the long-life reactor cores.

### 9.5.3 Basis for Development of LFR

The technology base for the LFR is primarily derived from the Pb–Bi liquid alloy-cooled reactors employed by the Russian Alpha class submarines. Technologies developed from the integral fast reactor metal alloy fuel recycle and refabrication development, and from the advanced liquid metal reactor (ALMR) passive safety and modular design approach, may also be applicable to the LFR. The ferritic stainless steel and metal alloy fuel developed for sodium fast reactors may also be adaptable to the LFR for those concepts with reactor outlet temperatures in the range of 550°C.

### 9.5.4 LFR R&D Requirements for Commercial Development

The principal R&D issues associated with the LFR are related to fuels and materials. The technology for ferritic stainless steel and metal alloy fuel is reasonably well developed for temperatures up to 550°C. However, in the range of 750°C–800°C, the development of nitride fuels will be required. The development issues include fuel/clad compatibility as well as clad/coolant compatibility. The development of high-temperature structural materials will also be needed.

System design issues include design of heat transport methods and the analysis of heat transfer mechanisms and processes. Other system issues include coolant chemistry control (especially oxygen control); design for seismic isolation; and design of passive safety features, including effective thermostructural reactivity feedback. The use of nitride fuels will also require specialized processes for fuel recycling to specifically address separations technology, remote refabrication technology, enrichment technologies, and irradiation testing requirement unique to the nitride fuels.

The development of a thermal-hydraulic and neutronic database along with validated analysis tool will be required to support the development of conceptual designs as well as supporting the safety analyses and licensing processes.

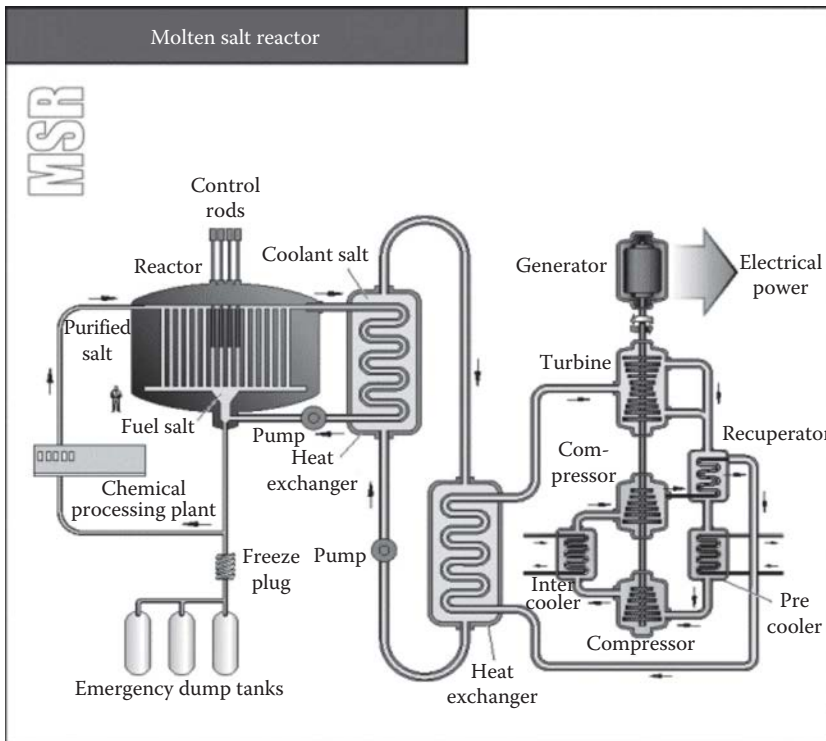
Plant economic issues will also be an important consideration that will require innovations in fabrication and construction techniques, and plant operations including monitoring and maintenance.

## 9.6 MSR

The MSR produces power by circulating a molten salt and fuel mixture through graphite core flow channels. The slowing down of neutrons by the graphite moderator in the core region provides the epithermal neutrons necessary to produce the fission power for sustained operation of the reactor.

### 9.6.1 MSR Description

The MSR is shown in Figure 9.6. Heat generated in the reactor core when the molten salt and fuel passes through the graphite core flow channels is transferred to a secondary system through an intermediate heat exchanger and then through a tertiary heat exchanger to the power conversion system that produces the electric power. The circulating coolant flow for this design is a mixture of sodium, uranium, and zirconium fluorides. In a closed fuel



**FIGURE 9.6**

Molten salt reactor. (From US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, A technology roadmap for generation IV nuclear energy systems, GIF-002-00, 2002.)

cycle, actinides such as plutonium can be efficiently burned by adding these constituents to the liquid fuel without the need for special fuel fabrication. The reference design for this concept is a 1000 MWe power plant with a coolant outlet of 700°C. To achieve higher thermal efficiencies for this concept, coolant outlet temperature as high as 800°C may also be possible.

### 9.6.2 MSR Achievement of Generation IV Goals

The ability of the MSR to address Generation IV sustainability goals is considered to be high because the concept can be used in a closed fuel cycle, and actinides and other nuclear fuel byproducts can be burned by simply adding these constituents to the circulating liquid fuel in a controlled fashion, but without special fuel fabrication needs. The ability of the MSR to address the Generation IV economic goals is unclear because of the early stage of development of the design and the anticipated large number of subsystems and components required for development of a viable design concept. Generation IV safety and the proliferation and physical protection would be addressed in the design approach but remain uncertain because of the relative immaturity of this specific design concept.

### 9.6.3 Basis for Development of MSR

Although the design of the MSR is relatively unproven, the technology base for the development of the MSR dates back to the 1940s and 1950s when MSRs were under consideration for aircraft propulsion applications. The aircraft reactor experiment in 1954 provided the first demonstration of the performance of a high-temperature molten fluoride salt reactor system. The 8-MWt MSR experiment (MSRE) extended the technology using a lithium/beryllium fluoride salt, graphite moderator, and different fuels to demonstrate system performance and stable operation. These programs ultimately led to the development of a detailed 1000-MWe engineering conceptual design of an MSR and helped to address issues relating to system performance, the stability of molten salt fuels, and the compatibility of molten salts with selected materials.

### 9.6.4 MSR R&D Requirements for Commercial Development

Technology R&D needed for commercialization of the MSR includes addressing issues relating to molten salt chemistry, solubility of actinides and lanthanides in the fuel, compatibility of irradiated molten salt with structural materials and graphite, and the metal clustering in heat exchangers. Metal clustering refers to the plate-out of noble metals on cooler heat exchanger surfaces. This plate-out could lead to reduced heat transfer and ultimately damage heat exchanger surfaces with the potential for loss of heat exchanger integrity.

The MSR design includes the primary system fuel salt and the secondary coolant salt. Because the operating requirements for these molten salts are different, the characteristics of the salts will also be different. The fuel salt has to meet requirements that include neutronic properties (low neutron cross section for the solvent components, radiation stability, negative temperature coefficient), thermal and transport properties (low melting point, thermal stability, low vapor pressure, adequate heat transfer, and viscosity), chemical properties (high solubility of fuel components, compatibility with containers and moderator materials, ease of fuel reprocessing, compatibility with waste forms), and low fuel and processing costs. The secondary salt operates in a significantly less severe environment;

because the temperature is lower, there are no fission products or actinides in the salt, and the neutron fluence is much lower. Therefore, there are more options for the selection of the secondary salt. However, the secondary salt must be compatible with the selected secondary materials, and the salt selection may also depend on the power conversion cycle selected. The power conversion cycle could be either a standard steam power cycle or possibly an advanced helium gas-turbine cycle.

Because of significant design differences from current commercial reactors, a comprehensive plan will need to be developed to define a safety case and licensing process equivalent to current commercial reactors. In addition, significant work will be required to define the requirement for the MSR fuel cycle, including proliferation resistance and physical protection issues.

The above technology issues point to the need for a significant R&D program to determine the viability of the concept and provide needed data to adequately evaluate the concept and support the detailed design effort. Data needs include data on material compatibility, cross sections and fuel performance data, thermal-hydraulics data, and component performance data. These data will also be required for the development and validation of analysis tools to be used in the design, development, and safety analyses for the MSR.

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1. US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum. 2002. A technology roadmap for Generation IV nuclear energy systems. GIF-002-00.



# Section II

## Introduction

### *Nuclear Fuel Cycle*

**Kenneth D. Kok**

This section discusses several aspects of the nuclear fuel cycle and the life cycle of nuclear systems. It begins with a discussion of uranium, which is the basic fuel of the current generation of nuclear systems, and is presented in Chapter 10. Uranium-containing minerals are mined and purified so that they can be further processed. Uranium as found in nature is primarily made up of the U-238 isotope, and it contains about 0.7% U-235, which is the isotope that is the primary fuel in today's reactors. The fissioning of 1 g of uranium will yield 1 MW day of energy or 24,000 kW h.

Uranium used in light water reactors must first be enriched by increasing the relative amount of U-235 in the material to be used as fuel in a light water moderated reactor. Heavy water reactors can operate using natural uranium. The enrichment process is described in Chapter 11. In Chapter 12, the processes used to manufacture the enriched uranium into usable nuclear fuel are discussed. The subsequent chapters discuss other processes related to the fuel after it is removed from the reactor, including storage (Chapter 13) and reprocessing (Chapter 14). The recycle of reprocessed fuel is discussed in Chapter 15.

Chapters 16 (waste disposal), Chapter 17 (radioactive materials transportation), and Chapter 18 (decontamination and decommissioning) are related to many radioactive processes and materials. Radioactive waste is generated wherever radioactive materials are handled and used. This includes medical applications of radioactive isotopes and their production, as well as the facilities and processes involved in nuclear power. This waste must be stored and/or disposed of in a way that isolates it from the environment until the radioactive materials decay.

Specialized transportation systems are used to transport radioactive materials. These include small containers used to deliver medical isotopes to containers that may weigh

tens of tons, and which are used to transport spent nuclear reactor fuel. Other types of containers are used to move materials that have been contaminated with radioactive materials. Some of this material is generated when facilities that have been used to process are decontaminated and decommissioned. Decontamination and decommissioning can occur in everything from laboratories that are no longer in use to nuclear power plants that have reached the end of their useful life.

# 10

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## *Nuclear Fuel Resources*

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Stephen W. Kidd

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### 10.1 Introduction

Uranium is the basic raw material for nuclear power. Section 10.2 considers its main characteristics and overall abundance, whereas Section 10.3 looks at uranium resources in rather more detail, showing that uranium is actually a relatively abundant element in the earth's crust. Section 10.4 outlines some basic uranium geology, making the point that it is available in a wide variety of geological settings. Section 10.5 moves on to consider mining techniques that may be employed to exploit known uranium deposits, highlighting the rise of in situ leaching (ISL) techniques in recent years. Section 10.6 briefly highlights the milling of extracted uranium ore, while Section 10.7 considers the important environmental aspects of uranium mining. Section 10.8 examines uranium production in some detail, by country, company, and major mines. Section 10.9 then looks at the world uranium market and considers price determination. Section 10.10 looks at potential secondary supplies of uranium, while Section 10.11 examines uranium conversion, the next necessary step in the nuclear fuel cycle. Finally, Section 10.12 considers thorium as an alternative nuclear fuel.

## 10.2 Uranium and Depleted Uranium

Uranium (chemical symbol, U) is a slightly radioactive metal that occurs throughout the earth's crust. It is about 500 times more abundant than gold; 40 times as silver; and about as common as tin, tungsten, and molybdenum. It occurs in most rocks in concentrations of 2–4 ppm, for example, at about 4 ppm in granite, which makes up 60% of the earth's crust. In fertilizers, uranium concentration can be as high as 400 ppm (0.04%), and some coal deposits contain uranium at concentrations of 100 ppm (0.01%). It is also found in the oceans, at an average concentration of 1.3 parts per billion (ppb).

On a scale arranged according to the increasing mass of their nuclei, uranium is the heaviest of all the naturally occurring elements (hydrogen is the lightest). Uranium has a specific gravity of 18.7.

There are several areas around the world where the concentration of uranium in the ground is sufficiently high that extraction of it for use as nuclear fuel is economically feasible. Such *economic concentrations* are called *ore*. When mined, it yields a mixed uranium oxide product ( $U_3O_8$ ). Uraninite or pitchblende is the commonest uranium mineral.

Uranium was discovered in pitchblende by the German chemist Martin Klaproth in 1789, and was named after the planet Uranus. Uranium was apparently formed in supernovas about 6.6 billion years ago. While it is not common in the solar system, today its radioactive decay provides the main source of heat inside the earth, causing convection and continental drift.

For many years from the 1940s, virtually all of the uranium that was mined was used in the production of nuclear weapons, but this ceased to be the case in the 1970s. Today the only substantial use for uranium is as fuel in nuclear reactors, mostly for electricity generation. Uranium-235 is the only naturally occurring material that can sustain a fission chain reaction, releasing large amounts of energy.

In the past, uranium was also used to color glass (from as early as 79 AD), and deposits were once mined to obtain its decay product, radium. This element was used in luminous paint, particularly on the dials of watches and aircraft instruments, and in medicine for the treatment of diseases.

While nuclear power is the predominant use of uranium, heat from nuclear fission can also be used for industrial processes. It is also used for marine propulsion (mostly naval), whereas research reactors are important for making radioisotopes.

Like other elements, uranium occurs in slightly differing forms known as *isotopes*. These isotopes differ from each other in the number of neutron particles in the nucleus. *Natural* uranium as found in the earth's crust is a mixture of three isotopes: uranium-238 (U-238), accounting for 99.275%; U-235 (0.720%); and traces of U-234 (0.005%). U-235 is important because under certain conditions it can be readily split, yielding a lot of energy. It is therefore said to be *fissile*, and we use the expression *nuclear fission*.

Like all radioactive isotopes, it decays. U-238 decays very slowly, its half-life being the same as the age of the earth. This means that it is barely radioactive, less so than many other isotopes in rocks and sand. U-238 has a specific radioactivity of 12.4 kBq/g, and U-235 80 kBq/g, so natural uranium is 13 kBq/g. In decay it generates 0.1 watts/t, and this is enough to warm the earth's mantle.

For most of the world's reactors, enriched uranium is required as fuel. Enrichment increases the proportion of the U-235 isotope from its natural level of 0.7% to 3%–5%. This enables greater technical efficiency in reactor design and operation, particularly in larger

reactors, and allows the use of ordinary water as a moderator. A by-product (or waste product) of enrichment is depleted uranium (about 89% of the original feed). Every tonne of natural uranium produced and enriched for use in a nuclear reactor gives about 130 kg of enriched fuel (at 3.5% U-235). The balance is depleted uranium (U-238, with 0.25%–0.30% U-235). This major portion has been depleted in its fissile U-235 isotope by the enrichment process. It is commonly known as *DU*.

DU is stored as  $UF_6$  or it is de-converted back to  $U_3O_8$ , which is more benign chemically and thus more suited for long-term storage. It is also less toxic. Every year, approximately 60,000 t of depleted uranium joins already substantial stockpiles in the United States, Europe, and Russia. The world stock is about 1.7 million tonnes.

Some DU is drawn from these stockpiles to dilute high-enriched (approximately 90%) uranium released from weapons programs, particularly in Russia, and destined for use in civil reactors. This weapons-grade material is diluted to about 25:1 with depleted uranium, or 29:1 with depleted uranium that has been enriched slightly (to 1.5% U-235) to minimize levels of (natural) U-234 in the product.

Other uses are more mundane and depend on the very high density (1.7 times that of lead) of the metal. Hence, where maximum mass must fit in minimum space (e.g., aircraft control surface and helicopter counterweights, yacht keels), it is often well suited. Until the mid-1970s it was used in dental porcelains. In addition, it is used for radiation shielding, being some five times more effective than lead in this role. Also because of its density, it is used as solid slugs or penetrators in armor-piercing projectiles, alloyed with about 0.75% titanium.

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### 10.3 Uranium Resources

Measured resources of uranium, the amount known to be economically recoverable from ore bodies, are naturally relative to costs and prices. They are also dependent on the intensity of past exploration effort and are basically a statement about what is known rather than what is present in the earth's crust.

Changes in costs or prices, or further exploration, may alter measured resource figures markedly. At 10 times the current price, seawater may become a potential source of vast amounts of uranium. Thus, predictions of the future availability of any mineral (including uranium) which are based on current cost and price data and current geological knowledge are likely to be extremely conservative.

From time to time concerns are raised that the known resources may be insufficient when judged as a multiple of present rate of use. This is the *limits to growth* fallacy, a major intellectual blunder recycled from the 1970s which does not take into account the very limited nature of the knowledge we have at any time of what is actually in the earth's crust. Our knowledge of geology is such that we can be confident that identified resources of metal minerals are a small fraction of what is present. With those major qualifications, Table 10.1 gives some idea of our present knowledge of uranium resources. It can be seen that Australia has a substantial part of the world's uranium, followed by Kazakhstan and Canada.

Current usage is about 65,000 tU/year, so the world's present measured resources of uranium (3.5 Mt) and used only in conventional reactors are enough to last for about 55 years. This represents a higher level of assured resources than is normal for most minerals.

**TABLE 10.1**

Known Recoverable Resources of Uranium, 2011

	Reasonably Assured Resources		
	<US\$40/kgU	<US\$80/kgU	<US\$130/kgU
Australia	0	961	1,158
Brazil	138	156	156
Canada	238	292	320
China	46	89	110
Kazakhstan	17	245	320
Namibia	0	6	235
Niger	6	6	339
Russia	0	12	173
South Africa	0	96	145
Ukraine	3	45	87
United States	0	39	207
Others	46	68	206
Total	494	2015	3456

Source: OECD NEA & IAEA, Uranium 2011: Resources, Production and Demand, (*Red Book*), Paris, France. With permission.

Further exploration and higher prices will certainly, on the basis of present geological knowledge, yield further resources as present ones are consumed. There was very little uranium exploration between 1985 and 2005, so the significant increase in exploration effort that has got underway since then will probably substantially increase the known economic resources.

This is suggested in the Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (IAEA) figures in Table 10.1 if those covering estimates of all conventional resources are considered—10 million tonnes (beyond the 3.5 Mt known economic resources), which takes us to over 200 years' supply at today's rate of consumption. It omits unconventional resources such as phosphate/phosphorite deposits (22-Mt uranium recoverable as by-product) and sea-water (up to 4000 Mt), which are likely to be uneconomic to extract in the foreseeable future.

Widespread use of the fast breeder reactor could increase the utilization of uranium by approximately 50-fold. This type of reactor can be started up on plutonium derived from conventional reactors and operated in closed circuit with its reprocessing plant. Such a reactor, supplied with natural or depleted uranium for its *fertile blanket*, can be operated so that each tonne of ore yields 60 times more energy than in a conventional reactor.

## 10.4 Geology of Uranium Deposits

The major primary ore mineral is uraninite (basically  $\text{UO}_2$ ) or pitchblende ( $\text{U}_2\text{O}_5 \cdot \text{UO}_3$ , better known as  $\text{U}_3\text{O}_8$ ), though a range of other uranium minerals are found in particular deposits. These include carnotite (uranium potassium vanadate), the davidite–brannerite–absite-type uranium titanates, and the euxenite–fergusonite–samarskite group (niobates of uranium and rare earths).

A large variety of secondary uranium minerals are known, many are brilliantly colored and fluorescent. The commonest are gummite (a general term like *limonite* for mixtures of various secondary hydrated uranium oxides with impurities); hydrated uranium phosphates of the phosphuranylite type, including autunite (with calcium), saleeite (magnesium), and torbernite (with copper); and hydrated uranium silicates such as coffinite, uranophane (with calcium), and sklodovskite (magnesium).

Uranium occurs in several different igneous, hydrothermal, and sedimentary geological environments. Uranium deposits worldwide have been grouped into 14 major categories of deposit types based on the geological setting of the deposits. The most significant are discussed below.

Unconformity-related deposits arise from geological changes occurring close to major unconformities. These constitute approximately one-third of world uranium resources and include some of the largest and richest deposits. Minerals are uraninite and pitchblende. The main deposits are in Canada (Athabasca Basin, Saskatchewan, and Thelon Basin, Northwest Territories) and Australia (Alligator Rivers region in the Pine Creek Geosyncline, Northern Territory, and Rudall River area, Western Australia). Today, all of Canada's uranium production is from unconformity-related deposits—Rabbit Lake (heavily depleted), and McClean Lake and McArthur River deposits. Another large, exceptionally high-grade unconformity-related deposit currently being developed in Saskatchewan is Cigar Lake (averaging almost 20%  $U_3O_8$ , with some zones above 50%  $U_3O_8$ ).

The Olympic Dam deposit is one of the world's largest deposits of uranium, accounting for about 66% of Australia's reserves plus resources and is categorized as a breccia complex deposit. It is overlain by approximately 300 m of flat-lying sedimentary rocks, and the deposit contains iron, copper, uranium, gold, silver, and rare earth elements. Only copper, uranium, gold, and silver have been recovered. Uranium grades average from 0.08% to 0.04%  $U_3O_8$ , the higher-grade mineralization being pitchblende. Copper grades average 2.7% for proved reserves, 2.0% for probable reserves, and 1.1% for indicated resources. Gold grades are 0.3–1.0 g/t.

Sandstone deposits constitute about 18% of world uranium resources. Ore bodies of this type are commonly low-to-medium grade (0.05%–0.4%  $U_3O_8$ ), and individual ore bodies are small to medium in size (ranging up to a maximum of 50,000 t  $U_3O_8$ ). The main primary uranium minerals are uraninite and coffinite. Conventional mining/milling operations of sandstone deposits have been progressively undercut by cheaper in situ leach mining methods. The United States has large resources in sandstone deposits in the Western Cordillera region, and most of its uranium production has been from these deposits, recently by ISL mining. The Powder River Basin in Wyoming, the Colorado Plateau, and the Gulf Coast Plain in south Texas are major sandstone uranium provinces. Other large sandstone deposits occur in Niger, Kazakhstan, Uzbekistan, Gabon (Franceville Basin), and South Africa (Karoo Basin). Kazakhstan has reported substantial reserves in sandstone deposits with average grades ranging from 0.02% to 0.07% U.

Surficial deposits comprise about 4% of world uranium resources, and such deposits represent 5% of Australia's total reserves and resources of uranium. They formed where uranium-rich granites were deeply weathered in a semi-arid to arid climate. The Yeelirrie deposit in Western Australia is by far the world's largest surficial deposit.

Volcanic deposits occur in acid volcanic rocks and are related to faults and shear zones within the volcanics. Uranium is commonly associated with molybdenum and fluorine. These deposits make up only a small proportion of the world's uranium resources. Significant resources of this type occur in China, Kazakhstan, Russian Federation, and Mexico.

Intrusive deposits are associated with intrusive rocks, including alaskite, granite, pegmatite, and monzonites. Major world deposits include Rossing (Namibia) and Palabora (South Africa).

Quartz-pebble conglomerate deposits make up approximately 13% of the world's uranium resources. Where uranium is recovered as a by-product of gold mining, the grade may be as low as 0.01%  $U_3O_8$ . In deposits mined exclusively for uranium, average grades range as high as 0.15%  $U_3O_8$ . Individual deposits range in size from 6,000 to 170,000 t contained  $U_3O_8$ . Major examples are the Elliot Lake deposits in Canada (where mining operations have ceased) and the Witwatersrand gold-uranium deposits in South Africa.

Vein deposits constitute about 9% of world uranium resources. Major deposits include Jachymov (Czech Republic) and Shinkolobwe (Zaire).

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## 10.5 Mining Techniques

The decision as to which mining method to use for a particular deposit is governed by the nature of the ore body, safety, and economic considerations. In the case of underground uranium mines, special precautions, consisting primarily of increased ventilation, are required to protect against airborne radiation exposure. Excavation and in situ techniques are used to recover uranium ore. Excavation may be underground and open pit mining.

In general, open pit mining is used if deposits are close to the surface and underground mining is used for deep deposits, typically above 120-m deep. Open pit mines require large holes on the surface, larger than the size of the ore deposit, because the pit walls must be sloped to prevent collapse. As a result, the quantity of material that must be removed to access the ore may be large. Underground mines have relatively small surface disturbance and the quantity of material that must be removed to access the ore is considerably less than in the case of an open pit mine.

An increasing proportion of the world's uranium now comes from ISL. This technique, also known as *in situ recovery* (ISR) or *solution mining*, involves leaving the ore where it is in the ground, and using liquids that are pumped through it to recover the minerals out of the ore by leaching. Consequently, there is little surface disturbance and no tailings or waste rock generated. However, the ore body must be permeable to the liquids used and located so that they do not contaminate groundwater away from the ore body.

ISL mining was first tried on an experimental basis in Wyoming during the early 1960s. The first commercial mine began operating in 1974. Today, a few projects are licensed to operate in the United States (Wyoming, Nebraska, and Texas) and most of the operating mines date from the 1990s. They are small (under 1000 t/year) but they supply most of the US uranium production. About 45% of world uranium production is now by ISL (including all Kazakhstan and Uzbekistan output). ISL can also be applied to other minerals such as copper and gold.

Uranium deposits suitable for ISL occur in permeable sand or sandstones, confined above and below by impermeable strata, which are below the water table. There are two operating regimes for ISL, determined by the geology and groundwater. If there is significant calcium in the ore body (as limestone or gypsum), alkaline (carbonate) leaching must be used. Otherwise, acid (sulfate) leaching is generally better. Techniques for ISL have evolved to the



point where it is a controllable, safe, and environmentally benign method of mining which can operate under strict environmental controls and which often has cost advantages.

Mining methods have been changing. In 1990, 55% of world production came from underground mines, but this shrunk dramatically to only 27% by 2012. This is despite the commissioning of new Canadian mines. Open-pit mines have also shrunk in significance and now only account for 20% of world production. By-production of uranium (where uranium is produced in combination with other metals such as copper and gold) was fairly constant over time at around 10% of output but has now shrunk to only 7%. Today it mainly reflects the production levels at the Olympic Dam mine in South Australia (copper and gold) rather than South Africa (gold). The share of ISL production has been rising very sharply, reaching 45% in 2012, owing mainly to rapidly increasing production levels in Kazakhstan and some recovery in Uzbekistan uranium output.

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## 10.6 Milling

Milling, which is generally carried out close to a uranium mine, extracts the uranium from the ore. Most mining facilities include a mill, although if mines are close together, one mill may process the ore from several mines. Milling produces a uranium oxide concentrate which is shipped from the mill. It is sometimes referred to as *yellowcake* and generally contains around 80% uranium. The original ore may contain as little as 0.1% uranium.

In a mill, uranium is extracted from the crushed and ground-up ore by leaching, in which a strong acid or a strong alkaline solution is used to dissolve the uranium. The uranium is then removed from this solution and precipitated. After drying and usually heating, it is packed in 200-L drums as a concentrate.

The remainder of the ore, containing most of the radioactivity and nearly all the rock material, becomes tailings, which are emplaced in engineered facilities near the mine (often in the mined-out pit). Tailings contain long-lived radioactive materials in low concentrations and toxic materials such as heavy metals; however, the total quantity of radioactive elements is less than in the original ore, and their collective radioactivity will be much shorter-lived. These materials need to be isolated from the environment.

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## 10.7 Environmental Aspects of Uranium Mining

In many respects, uranium mining is much the same as any other mining. Projects must have environmental approvals prior to commencing and must comply with all environmental, safety, and occupational health conditions applicable. Increasingly, these are governed by international standards, with external audits.

Once approved, open pits or shafts and drives are dug and waste rock and overburden is placed in engineered dumps. Tailings from the ore processing must be placed in engineered dams or underground. Finally, the whole site must be rehabilitated at the end of the project. Meanwhile air and water pollution must be avoided. In the case of ISL mining, there is much less disturbance, simply multiple boreholes, and rehabilitation is simpler.

Uranium is radioactive, though with the major isotope U-238, having a half-life equal to the age of the earth, is certainly not strongly radioactive. U-235 has a half-life one-sixth of this and emits gamma rays as well as alpha particles. Hence, a lump of pure uranium would give off some gamma rays, but less than those from a lump of granite. Its alpha radioactivity in practical terms depends on whether it is as a lump (or in rock as ore) or as a dry powder. In the latter case, the alpha radioactivity is a potential, though not major, hazard. It is also toxic chemically, being comparable with lead. Uranium metal is commonly handled with gloves as a sufficient precaution. Uranium concentrate is handled and contained so as to ensure that people do not inhale or ingest it.

At an early stage of a mine feasibility study, environmental studies of the site begin. These escalate in detail and progressively focus on issues of concern in relation to the proposal, in consultation with the governmental authorities. Depending on the government jurisdiction, an environmental effects or impact statement is published and may be made available for public comment. After consideration of comments and in the light of judgments by a wide range of authorities, approval may then be given for the project to proceed.

In most respects, conventional mining of uranium is the same as mining any other metalliferous ore, and well-established environmental constraints apply to avoid off-site pollution. From open cut mining, there are substantial volumes of barren rock and overburden waste. These are placed near the pit and used in rehabilitation or shaped and revegetated where they are.

However, uranium minerals are always associated with more radioactive elements such as radium and radon in the ore. Therefore, although uranium itself is not very radioactive, the ore which is mined, especially if it is very high-grade such as in some Canadian mines, is handled with some care for occupational health and safety reasons. Mining methods, tailings and run-off management, and land rehabilitation are subject to government regulation and inspection. Mining operations are undertaken under relevant national health and radiation protection codes of practice. These set strict health standards for exposure to gamma radiation and radon gas. Standards apply to workers and members of the public.

Solid waste products from the milling operation are tailings. They comprise most of the original ore and contain most of the radioactivity in it. In particular, they contain all the radium present in the original ore. At an underground mine they may be first cycloned to separate the coarse fraction which is used for underground fill. The balance is pumped as a slurry to a tailings dam, which may be a worked-out pit.

When radium undergoes natural radioactive decay, one of the products is radon gas. Radon occurs in most rocks and traces of it are in the air we all breathe. However, at high concentrations it is a health hazard. Because radon and its decay products (*daughters*) are radioactive and because the tailings are now on the surface, measures are taken to minimize the emission of radon gas. During the operational life of a mine the material in the tailings dam is usually covered by water to reduce surface radioactivity and radon emission (though with lower-grade ores will not pose a hazard at these levels).

On completion of the mining operation, it is normal for the tailings dam to be covered with some 2 m of clay and topsoil to reduce radiation levels to near those normally experienced in the region of the ore body, and for a vegetation cover to be established.

Run-off from the mine stockpiles and waste liquors from the milling operation are collected in secure retention ponds for isolation and recovery of heavy metals or other contaminants. The liquid portion is disposed of by natural evaporation or recirculation to

the milling operation. Process water discharged from the mill contains traces of radium and some other metals which would be undesirable in biological systems downstream. This water is evaporated and the contained metals are retained in secure storage. During the operational phase, such water may be used to cover the tailings while they are accumulating.

With ISL operations, the ore body stays in the ground and uranium is recovered by circulating oxygenated and acidified groundwater through it, using injection and recovery wells. The main environmental consideration with ISL is avoiding pollution of groundwater away from the ore body and leaving the immediate groundwater no less useful than it was initially.

Apart from tailings, other solid wastes at a mine include equipment which is not able to be sold at the end of the operation. This is usually buried with the tailings.

At the conclusion of mining, tailings are covered permanently with enough clay and soil to reduce gamma radiation levels and radon emanation rates to levels near those naturally occurring in the region and enough rock to resist erosion. A vegetation cover is then established. Apart from groundwater considerations discussed above, rehabilitation of ISL mines is very straightforward, making this a technique with remarkably low environmental impact. Upon decommissioning, wells are sealed or capped, process facilities removed, any evaporation pond revegetated, and the land can readily be returned to its previous uses.

At the concentrations associated with uranium (and some mineral sands) mining, radon is a potential health hazard, as is dust. Precautions taken during the mining and milling of uranium ores to protect the health of the workers are listed below:

- Good forced ventilation systems in underground mines to ensure that exposure to radon gas and its radioactive daughter products is as low as possible and does not exceed established safety levels.
- Efficient dust control because the dust may contain radioactive constituents and emit radon gas.
- Limiting the radiation exposure of workers in mine, mill, and tailings areas so that it is as low as possible, and in any event does not exceed the allowable dose limits set by the authorities. In Canada, this means that mining in very high-grade ore is undertaken solely by remote control techniques and by fully containing the high-grade ore where practicable.
- The use of radiation detection equipment in all mines and plants.
- Imposition of strict personal hygiene standards for workers handling uranium oxide concentrate.

At any mine, designated employees (those likely to be exposed to radiation or radioactive materials) are monitored for alpha radiation contamination and personal dosimeters are worn to measure exposure to gamma radiation. Routine monitoring of air, dust, and surface contamination is undertaken. If uranium oxide is ingested it has a chemical toxicity similar to that of lead oxide. Similar hygiene precautions to those in a lead smelter are therefore taken when handling it in the drying and packing areas of the mill. The usual radiation safeguards are applied at an ISL mining operation, despite the fact that most of the radioactivity remains well underground, and there is hence minimal increase in radon release and no ore dust.

## 10.8 Uranium Production

Although there remain some uncertainties about the amounts of uranium mined in the former Soviet Union in the period between 1945 and 1990, it is nevertheless now possible to produce a rough picture of cumulative world production since the end of World War II. Table 10.2 summarizes historical production, with estimates of production in countries where data are not fully available.

Cumulative world uranium production is estimated to be 2.56 million tU since 1945. Canada and the United States are clearly the largest two producing countries historically. There are more uncertainties related to the production numbers in the former Soviet Union and China, particularly prior to 1991. The patterns of production have been markedly different in the West and the remainder of the world.

Within the West, the period can be divided into four *ages of uranium*, as detailed below.

- *A Military era, from 1945 to the late 1960s.* The generation of electricity from nuclear fuel was incidental to the arms race as production rose rapidly during the 1950s to satisfy the military requirement for highly enriched uranium (HEU) and plutonium. Uranium demand from this source fell sharply from 1960 onward and, in response, production halved by the mid-1960s.
- *A period of rapidly expanding civil nuclear power, lasting from the late 1960s to the mid-1980s.* Uranium production picked up again as reactor orders expanded. Many new mines were quickly brought into production, frequently underwritten by long-term contracts agreed with electricity utilities in North America, Japan, and Western Europe. Production peaked in 1980 and stayed above annual reactor requirements until 1985.

**TABLE 10.2**

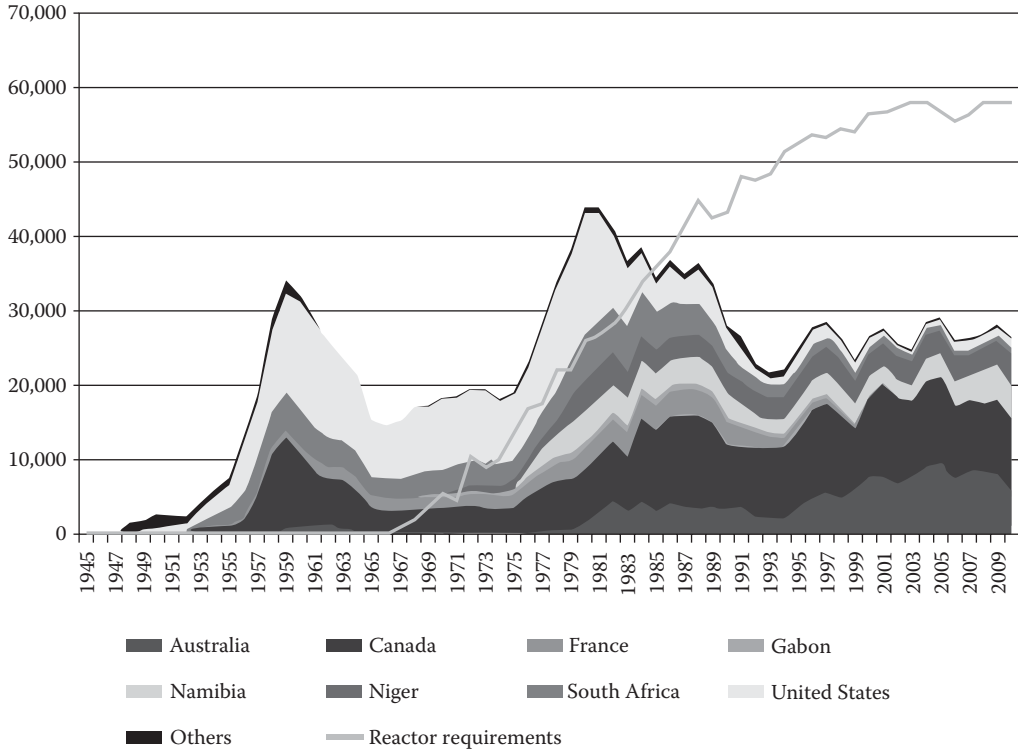
Cumulative Uranium Production, 1945–2012

Country/Area	Cumulative Production (tU)
Australia	183,974
Canada	466,144
China	42,385
Czech Republic	110,457
France	77,009
Germany	217,101
Kazakhstan/Uzbekistan	339,668
Namibia	111,753
Niger	123,018
Russia	152,865
South Africa	162,047
Ukraine	61,850
United States	367,133
Others	144,483
World total	2,559,887

Source: World Nuclear Association, *The Global Nuclear Fuel Market, Supply and Demand 2013–2030*, London, 2013. With permission.

- An age dominated by an inventory over-hang, extended by supply from the former Soviet Union (NIS), lasting from the mid-1980s up to about 2002. By 1985, nuclear construction programs had been cut back severely. However, many utilities had signed uranium contracts in anticipation of building additional nuclear plants. Honoring these resulted in a build-up of utilities' uranium inventories. As these were being run down, mines closed or cut production. Utilities satisfied much of their requirements without recourse to new production. This period was extended by the arrival on Western markets of uranium from the former Soviet Union in the early 1990s. Markets formerly almost completely segregated reached the first stage of integration, although the material flow was effectively one-way only.
- A strong market reaction to the perception that secondary supplies are beginning to run out and that primary production needed to rise sharply to fill more of the gap still evident with reactor requirements. This started in 2003 with a strong upward movement in world uranium prices and continued into 2007 (the spot market price rose by a factor of 10 between 2003 and mid-2007).

Figure 10.1 summarizes the position in the West since 1945. Production was substantially ahead of reactor requirements until 1985, but has since fallen. Since 1985, requirements have exceeded production by more than 450,000 tU.



**FIGURE 10.1**  
Western world uranium production and demand 1945–2010.

Within the former Soviet Union and Eastern Europe, significant reductions in uranium production occurred much later than in the West. Fissile material production was considered vital in the Soviet Union with central plans seeking to maximize production irrespective of civil reactor demand. This resulted in the accumulation of large quantities of materials in various forms. Uranium production began to fall sharply only from the late 1980s onward, largely as a consequence of the end of the arms race, but also of the exhaustion of some resources and the economic disruptions during the collapse of the Soviet Union.

The opening-up of Western markets for nuclear fuel to producers in the former Soviet Union also occurred at this time. The additional outlet for material was certainly welcome by producers there, but application of market financial criteria indicated that much production from these countries was uneconomic. Even though uranium exports from the former Soviet Union to the West have been subject to trade restrictions, it is believed that approximately 200,000 tU has so far reached Western markets from there.

Table 10.3 shows that Kazakhstan has now become easily the world's largest producer of uranium, taking over the position formerly held by Canada. Kazakhstan produced 36% of world supply from mines in 2012, followed by Canada (15%) and Australia (12%).

Uranium production and demand has been heavily impacted by the Fukushima.

*Accident in Japan in 2011.* Prior to this event, uranium producers were expanding or preparing to expand for projected uranium shortfalls in the coming years. Following the accident, nuclear plans in Japan and around the globe faced increased uncertainty which eventually led to reduced demand projections and thus many mining projects being

**TABLE 10.3**

Uranium Production by Country (Tonnes U), 2010–2012

	Change (%)			
	2004	2005	2006	2010–2012
Australia	5,900	5,983	6,991	+18
Brazil	148	265	231	+56
Canada	9,783	9,145	8,998	–8
China	827	885	1,500	+81
India	400	400	385	–4
Kazakhstan	17,803	19,451	21,317	+20
Malawi	846	1,077	1,101	+30
Namibia	4,496	3,258	4,495	0
Niger	4,198	4,351	4,667	+11
Russia	3,562	2,993	2,872	–19
South Africa	583	582	465	–20
Ukraine	850	890	960	+13
United States	1,660	1,537	1,596	–4
Uzbekistan	2,400	2,500	2,400	0
Others	207	178	366	+77
Total	53,663	53,494	58,344	+9

Source: World Nuclear Association, *The Global Nuclear Fuel Market, Supply and Demand 2013–2030*, London, 2013. With permission.

delayed or deferred indefinitely. The reduced demand affected the uranium junior mining sector in particular. Many small firms found themselves unable to secure funding to develop their new mines, and this again led to delays and cancellations of a number of projects in the development pipeline.

As shown in Table 10.3, even with the deteriorating market conditions from 2010 to 2012, uranium production increased through this period as new production centers gained efficiencies in ramping up and achieved production near their nameplate capacities. This ramp up has been led by Kazakhstan, due to the continued expansion of their ISL operations.

The new mines that have entered production since 2010 include Honeymoon in Australia, Langer Heinrich Stage three in Namibia, Novokonstantinovskoe in Ukraine, and Willow Creek in the United States. However, Husab in Namibia is the only significant project that advanced during this time to the *under development* stage. Other projects were delayed or deferred indefinitely. For example, Imouraren in Niger, and Talvivaara in Finland have been delayed but with development ongoing, while Trekkopje in Namibia has been deferred indefinitely. The most important project to be delayed was the Olympic Dam Expansion where different possible mine configurations are being studied. No major mine closures have occurred since 2010 but it should be noted that the Australian open pit mine, Ranger, was fully mined in 2012, although stockpiles remain to be processed. Ezulwini in South Africa has been shut down and is proposed to be re-developed in the coming years.

During the 1990s, the uranium production industry was consolidated by takeovers, mergers, and closures. In 2012, the eight mining companies with over 2000 t output (marketable share of production) accounted for 82% of world mine production (Table 10.4).

With the onset of significant ISL production from Kazakhstan, which is based in a large number of smaller mines, production has also become less concentrated in a few major mines throughout the world. Table 10.5 shows that 51% of production came from the 10 largest mines—in the recent past, this has approached 70%.

**TABLE 10.4**

Uranium Production by Company, Tonnes U, 2012

	Production (tU)	World Share (%)
Kazatomprom	8,863	15
AREVA	8,641	15
Cameco	8,437	14
ARMZ/U-One	7,629	13
Rio Tinto	5,435	9
BHP Billiton	3,386	6
Paladin Energy	3,056	5
Navoi	2,400	4
Others	10,498	18
World total	58,344	100

Source: World Nuclear Association, *The Global Nuclear Fuel Market, Supply and Demand 2013–2030*, London, 2013. With permission.

**TABLE 10.5**

Uranium Production from the Top 10 Mines, Tonnes U, 2012

Mine	Country	Main Owner	Mine Type	Production (tU)	Percentage of World Production
McArthur River	Canada	Cameco	Underground	7,520	13
Olympic Dam	Australia	BHP Billiton	By-product/ underground	3,386	6
Ranger	Australia	ERA/Rio Tinto	Open pit	3,146	5
Arlit	Niger	AREVA	Open pit	3,065	5
Tortkuduk	Kazakhstan	Katco JV-AREVA	ISL	2,661	5
Rossing	Namibia	Rio Tinto	Open pit	2,289	4
Karatau/Budenovskoye 2	Kazakhstan	Kazatomprom/U-One	ISL	2,135	4
Priargunsky	Russia	ARMZ	Underground	2,011	3
Langer Heinrich	Namibia	Paladin	Open pit	1,955	3
South Inkai	Kazakhstan	Kazatomprom/U-One	ISL	1,870	3
Total from top 10 mines				30,037	51
World total				58,344	

Source: World Nuclear Association, *The Global Nuclear Fuel Market, Supply and Demand 2013–2030*, London, 2013. With permission.

## 10.9 Uranium Market and Prices

All mineral commodity markets tend to be cyclical, that is, prices rise and fall substantially over the years, but with these fluctuations superimposed on long-term decline in real prices. In the uranium market, very high prices in the late 1970s gave way to very low prices in the early 1990s, the spot prices being below the cost of production for most mines. In 1996, spot prices recovered briefly to the point where most mines could produce profitably, though they then declined again and only started to recover strongly late in 2003.

The reasons for fluctuation in mineral prices relate to demand and perceptions of scarcity. The price cannot indefinitely stay below the cost of production, nor will it remain at very high levels for longer than it takes for new producers to enter the market and anxiety about supply to subside.

Today some 432 operable reactors with combined capacity of 372 GWe require 65,000 t of uranium from mines (or the equivalent from stockpiles or secondary sources) each year. The capacity is growing slowly, and at the same time the reactors are being run more productively, with higher capacity factors and reactor power levels. However, these factors increasing fuel demand are offset by a trend for increased efficiencies, so demand is dampened; over the 20 years from 1970 there was a 25% reduction in uranium demand per kWh output in Europe due to such improvements, which continue today. Each GWe of increased capacity will require about 200 tU/year of extra mine production routinely and about 2.5-times this for the first fuel load.



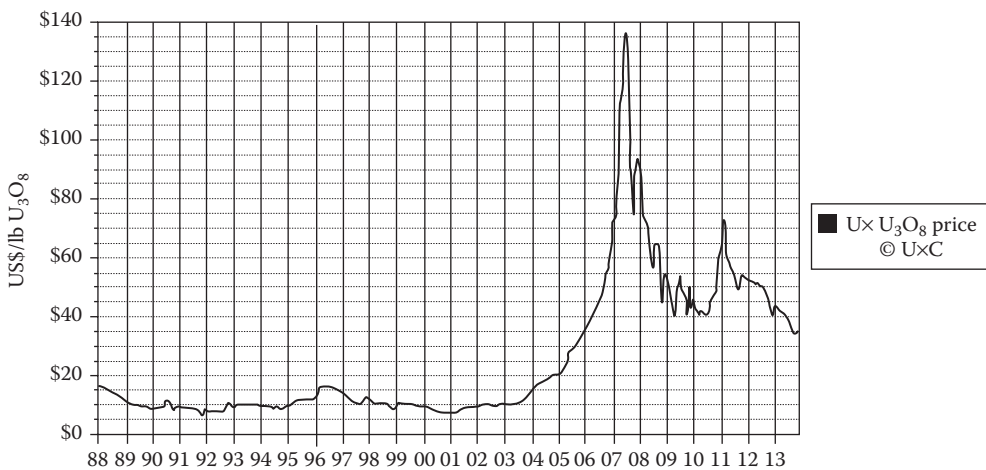
Because of the cost structure of nuclear power generation, with high capital and low fuel costs, the demand for uranium fuel is much more predictable than with probably any other mineral commodity. Once reactors are built, it is very cost-effective to keep them running at high capacity and for utilities to make adjustments to load trends by cutting back on fossil fuel use. Demand forecasts for uranium thus depend largely on installed and operable capacity, regardless of economic fluctuations.

Looking 10 years' ahead, the market is expected to grow slightly. Demand thereafter will depend on new plants being built and the rate at which older plants are retired. Licensing of plant lifetime extensions and the economic attractiveness of continued operation of older reactors are critical factors in the medium-term uranium market. However, with electricity demand by 2030 expected (by the OECD's International Energy Agency) to rise by at least two-thirds from that of 2010, there is plenty of scope for growth in nuclear capacity in a greenhouse gas conscious world.

Mines in 2012 supplied some 58,344 tU or about 90% of utilities' annual requirements (see above). The balance is made up from secondary sources or stockpiled uranium held by utilities, but those stockpiles are now substantially depleted.

The perception of imminent scarcity drove up the spot price of uranium for uncontracted sales to US\$138 per pound  $U_3O_8$  by mid-2007, but prices subsequently fell back significantly to US\$75 per pound by the autumn. Since then, prices have moved downward sharply to the US\$40 per pound level by late 2012. Most uranium however is supplied under long-term contracts, and the prices in new contracts have, in the past, reflected a premium above the spot market. Note that at the prices which utilities are likely to be paying for current delivery, only one-quarter of the cost of the fuel loaded into a nuclear reactor is the actual ex-mine (or other) supply. The balance is mostly the cost of enrichment and fuel fabrication, plus an allowance for used fuel management.

The spot prices shown in Figure 10.2 apply to marginal trading from day to day and usually represent 10%–20% of supply. Most trade is 3–10-year term contracts with producers selling direct to utilities, but with the price often related to the spot price.



**FIGURE 10.2**  
Uranium spot prices 1988 to date.

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## 10.10 Secondary Supplies of Uranium

As well as existing and likely new mines, nuclear fuel supply may be from secondary sources, including

- Recycled uranium and plutonium from spent fuel as mixed oxide (MOX) fuel
- Re-enriched depleted uranium tails
- Ex-military weapons-grade uranium
- Civil stockpiles
- Ex-military weapons-grade plutonium

Major commercial reprocessing plants are operating in France and the United Kingdom, with capacity of over 4000 t of used fuel per year. The product from these re-enters the fuel cycle and is fabricated into fresh MOX fuel elements. About 200 t of MOX is used each year, equivalent to less than 2000 t of uranium from mines.

Arguably, the biggest secondary supply of uranium available is the energy value locked up in the inventory of used reactor fuel round the world, regarded as a waste. The effective amount of uranium available can be greatly extended by the design of a fuel cycle and associated reactor that can use both the U235 and the U238 contained in natural uranium mined from the earth. Rather than using only 0.5% of the uranium, advanced fuel cycles and advanced reactors will use 75% or more of the uranium. Thus, projected supplies will allow uranium to be used as a major source of energy for thousands of years, making nuclear energy sustainable for the foreseeable future. We are essentially wasting a valuable energy resource, that is, uranium, by throwing away (not recycling or re-using) over 99% of its potential energy and then creating unwanted and un-needed problems of storage and disposal for generations to come.

Assuming there is spare enrichment capacity, it can also be economically feasible to send depleted uranium through enrichment plants again, in order to *create* additional enriched uranium. This is known as *re-enrichment* and, given the magnitude of world depleted uranium stocks (about 1.7 million tonnes), this could conceivably be an important part of secondary supplies in the future.

Military uranium for weapons is enriched to much higher levels than that for the civil fuel cycle. Weapons-grade is about 97% U-235, and this can be diluted about 25:1 with depleted uranium (or 30:1 with enriched depleted uranium) to reduce it to about 4%, suitable for use in a reactor. From 1999 to 2013, the dilution of 30 t of such material is displacing about 10,600 t/year of mine production as a result of the 1994 *Megatons to Megawatts* agreement between the United States and Russia. The United States and Russia also agreed to dispose of 34 t each of military plutonium by 2014. Most of it is likely to be used as feed for MOX plants, to make about 1500 t of MOX fuel which will progressively be burned in civil reactors.

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## 10.11 Conversion

The product of a uranium mill is not directly usable as a fuel for a nuclear reactor. Additional processing, generally referred to as *enrichment*, is required for most types of reactors. This process requires uranium to be in gaseous form and the way this is achieved is to convert it to uranium hexafluoride, which is a gas at relatively low temperatures.

At a conversion facility, uranium is first refined to uranium dioxide, which can be used as the fuel for those types of reactors that do not require enriched uranium. Most is then converted into uranium hexafluoride, ready for the enrichment plant. It is shipped in strong metal containers. The main hazard of this stage of the fuel cycle is the use of hydrogen fluoride.

Light water reactors (LWRs) require a process that involves transforming natural uranium concentrates into  $UF_6$ . The UK's gas-cooled reactors (AGRs) also require  $UF_6$  conversion. Heavy water reactors (HWRs), which are mainly of the CANDU design, require conversion from natural uranium concentrates directly to  $UO_2$ . For the UK's Magnox gas-cooled reactors, conversion from natural uranium concentrates to uranium metal and fuel fabrication is handled domestically in dedicated facilities.

Russia has produced two main reactor designs, known as VVERs and RBMKs. The former is a type of LWR, whereas the latter are light water-cooled graphite reactors. These reactors predominately exist in Russia, other countries that were part of the former Soviet Union, and Eastern Europe. Both designs use enriched uranium and therefore require  $UF_6$  conversion.

Worldwide requirements for conversion services, averaged over an extended period, will be equal to aggregate demand for uranium requirements, as outlined above, after allowing for the few reactors which do not require conversion. Accordingly, those factors, which affect annual worldwide uranium demand, must be considered when examining conversion demand, as well as any factors that are indigenous to the conversion market.

Countries operating CANDUs or other HWRs with requirements for  $UO_2$  conversion are Argentina, Canada, China, India, Korea, Pakistan, and Romania. The key to future growth in demand is the magnitude of the Indian nuclear program, which relies heavily on HWRs.

Worldwide, five major suppliers meet the majority of the demand for  $UF_6$  conversion services. Together, they have a combined nameplate capacity of 76,000 tU per annum. However, it should be noted that it is unfeasible for production plants to continually sustain 100% of nameplate capacity, and therefore production figures of significantly less than this should be expected.

With regard to  $UO_2$  conversion supply, Cameco's plant is by far the largest supplier, with a licensed annual capacity of 2800 tU. In addition, smaller plants exist to meet the local needs in India, Argentina, and Romania. Cameco Corporation owns and operates uranium refinery and conversion facilities located respectively at Blind River and Port Hope. The Blind River plant refines natural uranium concentrates ( $U_3O_8$ ) into uranium trioxide ( $UO_3$ ) and was commissioned in 1983. The intermediate product is shipped to the Port Hope plant (commissioned 1984) where further processing produces natural  $UF_6$ .

Many of the secondary sources of uranium described above also displace demand for  $UF_6$  conversion. These include inventories of  $UF_6$  and low enriched uranium (LEU), Russian and US ex-military HEU and plutonium, uranium and plutonium recovered by civil spent fuel reprocessing, and  $UF_6$  supply from the re-enrichment of tails. In addition, *underfeeding* of enrichment facilities can also affect the  $UF_6$  market.

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## 10.12 Thorium

Thorium is a naturally occurring, slightly radioactive metal discovered in 1828 by the Swedish chemist Jons Jakob Berzelius, who named it after Thor, the Norse god of thunder. It is found in small amounts in most rocks and soils, where it is about three times more abundant than uranium. Soil commonly contains an average of around 6 ppm of thorium.

**TABLE 10.6**  
Worldwide Thorium Resources

Economically Extractable	
Country	Reserves (Tonnes)
Australia	300,000
India	290,000
Norway	170,000
United States	160,000
Canada	100,000
South Africa	35,000
Brazil	16,000
Other countries	95,000
World total	1,200,000

*Source:* US Geological Survey, Mineral commodity summaries, January 1999. With permission.

Thorium occurs in several minerals, commonest being the rare earth-thorium-phosphate mineral, monazite, which contains up to about 12% thorium oxide (but average 6%–7%). There are substantial deposits in several countries (Table 10.6).

Thorium-232 decays very slowly (its half-life is about three times the age of the earth), but other thorium isotopes occur in the decay chain of it and uranium. Most of these are short-lived and hence much more radioactive than Th-232, though on a mass basis they are negligible.

When pure, thorium is a silvery white metal that retains its luster for several months. However, when it is contaminated with the oxide, thorium slowly tarnishes in air, becoming gray and eventually black. Thorium oxide ( $\text{ThO}_2$ ), also called thoria, has one of the highest melting points of all oxides (3300°C). When heated in air, thorium metal turnings ignite and burn brilliantly with a white light. Because of these properties, thorium has found applications in lightbulb elements, lantern mantles, arc-light lamps, welding electrodes, and heat-resistant ceramics. Glass containing thorium oxide has a high refractive index and dispersion and is used in high-quality lenses for cameras and scientific instruments.

Thorium, as well as uranium, can be used as a nuclear fuel. Although not fissile itself, thorium-232 (Th-232) will absorb slow neutrons to produce uranium-233 (U-233), which is fissile. Hence, like uranium-238 (U-238), it is fertile.

In one significant respect, U-233 is better than uranium-235 and plutonium-239: its higher neutron yield per neutron absorbed. Given a start with some other fissile material (U-235 or Pu-239), a breeding cycle similar to but more efficient than that with U-238 and plutonium (in slow-neutron reactors) can be set up. The Th-232 absorbs a neutron to become Th-233, which normally decays to protactinium-233 and then U-233. The irradiated fuel can then be unloaded from the reactor, the U-233 separated from the thorium, and fed back into another reactor as part of a closed fuel cycle.

Over the past 30 years, there has been interest in utilizing thorium as a nuclear fuel because it is more abundant in the earth's crust than uranium. Also, all of the mined thorium is potentially usable in a reactor, compared with the 0.7% of natural uranium, so about 40 times the amount of energy per unit mass may theoretically be available (without recourse to fast breeder reactors).

# 11

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## *Uranium Enrichment*

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Nathan (Nate) Hurt and Kenneth D. Kok

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## 11.1 History of Uranium Enrichment

### 11.1.1 Beginning

Physicists all over the world were excited by the announcement in January 1939, by the German chemists Otto Hahn and Fritz Strassman, that the nucleus of the uranium atom produced barium when bombarded with neutrons. It was deduced by Otto Frisch and Lise Meitner that the barium resulted from the splitting or fissioning of the uranium nucleus. The German work was quickly confirmed by four laboratories in the United States (Columbia University, Carnegie Institute of Washington, Johns Hopkins University, University of California) and others around the world. Those well recognized for their work at that time were Niels Bohr and E. Fermi. The possible military applications of *atomic power* were soon being discussed in every country. In the United States, studies of

the possible approaches and problems were undertaken with government support in 1940. These studies produced promising results and culminated in June 1942 with the decision by the government to undertake an all-out effort to develop the bomb.

During the next 6 months, the Manhattan Engineering District of the US Corps of Engineers was organized to direct the work. Two routes were undertaken: the uranium route at Oak Ridge, Tennessee, in new facilities (code-named Y-12 and K-25); and the plutonium route in new facilities in Oak Ridge (code-named X-10) and at Hanford, Washington. Laboratories at several universities had been for some time studying possible approaches for separating uranium isotopes (gas centrifuge, gaseous diffusion, thermal diffusion, photochemical, electromagnetic). The choice was narrowed down in late 1942 to two methods: the electromagnetic process and the gaseous diffusion process.

The first uranium isotope separation operations took place at Y-12 in Fall 1943 using the electromagnetic process. The uranium fuel used in the Hiroshima bomb came from that operation. Within another year the gaseous diffusion process at K-25, a backup gamble, was proven to be a workable process and, being a continuous rather than batch operation, it was much less expensive. Y-12 was therefore shut down in December 1946.

There was another enrichment plant that was built near K-25, the S-50 thermal diffusion plant, developed by the US Navy. It operated only 1 year, from September 1944 until September 1945, 1 month after the end of World War II. These events marked the beginning of uranium enrichment.

### 11.1.2 Evolution of Uranium Enrichment

During World War II, the interest of the United States was in producing very highly enriched uranium (>80%) for bombs. However, in the early 1960s, the interest shifted to producing moderately enriched uranium (3%–5%) as fuel for plants generating electricity. Reactor developers had found that when lower level enriched uranium was used as reactor fuel, the cost of the electricity produced could be cheaper. The demand for more enrichment capacity in the 1950s and the successful operation of the gaseous diffusion plant in Oak Ridge, Tennessee, led to the construction and operation of two additional plants (Paducah, Kentucky, and Portsmouth, Ohio).

There also have been gaseous diffusion plants built in other countries, some still in operation, but the gaseous diffusion process is gradually being replaced by the gas centrifuge process. Another method of enrichment, laser isotopic separation, has generated interest in recent years.

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## 11.2 What Is Uranium Enrichment?

Enriched uranium is a critical component in civilian nuclear power generation, naval power generation, and military nuclear weapons. One of the forms in which uranium exists in nature—the  $U^{235}$  isotope—is fissionable, that is, can be caused to split and release tremendous amounts of energy. To build a weapon or make it usable as reactor fuel, the desirable  $U^{235}$  isotope must be enriched from the very low concentrate (0.7%) found in nature where the balance (99.3%) is  $U^{238}$ . Enrichment is therefore the term used to describe processes by which the  $U^{235}$  concentration is increased above its naturally occurring level.

### 11.2.1 Grades of Enrichment

Slightly enriched uranium (SEU) has a  $U^{235}$  concentration of 0.9%–2.0% and is being used in some heavy water reactors. Low enriched uranium (LEU) has a  $U^{235}$  concentration of <20%. The biggest use of LEU is in power-generating reactors at enrichment between 3% and 5%. HEU has a  $U^{235}$  concentration >20%. Uranium enriched to weapon-grade and fuel for naval fleet weapons has  $U^{235}$  concentrations of  $\geq 85\%$ .

Enrichment calculations are based on the use of a simple mass balance as shown in the series of equations shown below:

$$F = P + T$$

$$Fx_F + Px_P + Tx_T$$

where:

$F$  = mass of feed

$P$  = mass of the product

$T$  = mass of the tails

$x_F$  = fraction of the  $U^{235}$  in the feed

$x_P$  = fraction of the  $U^{235}$  in the product

$x_T$  = fraction of the  $U^{235}$  in the tails

The fraction of  $U^{235}$  in the feed is known and the fraction of  $U^{235}$  in the product and the tails are specified; thus to determine the amount of feed required and the amount of tails produced the system is solved as follows:

$$F = \frac{P(x_P - x_T)}{(x_F - x_T)}$$

and

$$T = \frac{P(x_P - x_F)}{(x_F - x_T)}$$

Using the equations above, the mass of feed uranium  $UF_6$  and the quantity of tails  $UF_6$  produced in order to obtain a product of one metric ton of enriched  $UF_6$  for various tail assays and enrichments is shown in Table 11.1.

**TABLE 11.1**

The Amount of Feed Required to Produce 1 Metric Ton of Enriched Product

U <sup>235</sup> Fraction			Mass of UF <sub>6</sub> —MT		
Feed	Product	Tails	Feed	Product	Tails
0.0072	0.03	0.0025	6.0	1.0	5.0
0.0072	0.03	0.0015	5.1	1.0	4.1
0.0072	0.05	0.0025	10	1.0	9.0
0.0072	0.20	0.0025	42	1.0	41
0.0072	0.93	0.0025	197	1.0	196

### 11.2.2 Separative Work Unit

The term separative work unit (SWU) is used to quantify the output of the enriching plants and to describe the supply and demand for enriching service. It is not a quantity of material. It is the measure of the effort spent to separate a given quantity into two streams, one having a higher concentration of  $U^{235}$ , the other lower.

The equation used to calculate SWU for a cascade with enriching and stripping sections is as follows:

$$E = P \left( \frac{x_P - x_F}{x_F - x_T} \varnothing_T + \varnothing_P - \frac{x_P - x_T}{x_F - x_T} \varnothing_F \right)$$

Where  $\varnothing$  is the separation potential and is defined as:

$$\varnothing_T = (2x_T - 1) \ln \frac{x_T}{1 - x_T}$$

$$\varnothing_F = (2x_F - 1) \ln \frac{x_F}{1 - x_F}$$

$$\varnothing_P = (2x_P - 1) \ln \frac{x_P}{1 - x_P}$$

Using these equations the separative work required to produce uranium enriched to the levels shown in Table 11.1 are listed in Table 11.2.

---

## 11.3 Methods of Uranium Enrichment

### 11.3.1 Thermal Diffusion

The thermal diffusion enrichment process was developed by Dr. Philip H. Abelson for naval applications. He was asked to oversee the engineering and operations of a thermal diffusion plant, S-50, at K-25 in Oak Ridge, Tennessee, during World War II as a back-up to the other enrichment processes, which were experiencing problems.

**TABLE 11.2**

Separative Work Needed to Produce Enriched  $U^{235}$  from Natural Uranium

U <sup>235</sup> Fraction		Separative Work
Enrichment	Tails	
0.03	0.0025	3.8
0.03	0.0015	5.0
0.05	0.0025	7.8
0.20	0.0025	41
0.93	0.0025	214



Thermal diffusion utilizes the transfer of heat across a thin liquid layer to accomplish isotope separation. The process is based upon the fact that lighter molecules diffuse toward a hot surface and heavier molecules diffuse toward a cold surface.

The S-50 plant built contained 2142 diffusion columns that were 48' high. Each column consisted of a 1 1/4" nickel pipe in the center to contain superheated steam and a slightly larger copper pipe with a precisely fixed 0.01" annular spacing for the liquid UF<sub>6</sub>. Both pipes were surrounded by a 4"-diameter galvanized pipe for the cooling water.

The plant operated for 1 year but was shut down and abandoned in favor of the gaseous diffusion process. Clusius and Dickel in Germany experimented with thermal diffusion in 1938 for other isotopes, and the Japanese tried to use the process for uranium enrichment without success during World War II. There is no known interest in thermal diffusion at this time.

### 11.3.2 Electromagnetic Isotope Separation

Calutrons are electromagnetic isotope separators that operate like analytical mass spectrometers. The term *calutron* is a tribute to the work of E. O. Lawrence and his team of scientists who developed the process at their University of California cyclotron laboratory and assisted in its transformation to a production-scale process at the electromagnetic plant located at the Y-12 site in Oak Ridge, Tennessee. The Y-12 calutron process was replaced shortly after the end of World War II by the gaseous diffusion process located at the K-25 plant (also in Oak Ridge). The gaseous diffusion operation had much larger production capabilities and was far less labor intensive.

A calutron is a mass spectrometer with much higher material throughput than a typical mass spectrometer used for analytical purposes. Because of this high throughput, it is very useful for producing milligram to multikilogram quantities of enriched isotopes for a wide variety of applications, ranging from nuclear weapons to nuclear medicine, and for research to industrial-scale quantities.

In a calutron, the feed material, composed of an element with multiple isotopes, is vaporized by heating in a vacuum. It must be in a chemical form (elemental or compound) that has sufficient vapor pressure between room temperature and approximately 1000°C to form adequate amounts of vapor for the process. This vapor passes into an electron plasma ionization chamber which ionizes the atom, ideally, to a +1 charge. The ions are then extracted by high voltage to form an ion beam. The entire process takes place within an intense magnetic field which bends the ion beam in a semi-circular path. The light ions are bent by the magnetic field into a tighter radius than the heavier ions.

With the proper combination of accelerating voltage and magnetic field suited to the range of masses being separated, discrete ion beams of each isotope in the feed material can be formed and spatially separated sufficiently to be physically collected.

Although the enrichment of uranium by calutrons was discontinued in 1946, 72 of the 1152 calutrons remained in operation for the separation of isotopes for research in medicine, agriculture, industry, and biology. Eight calutrons were still in operation in 1997.

Because of its use in the Manhattan Project, the details of the electromagnetic isotope enrichment process were highly classified. After the discontinuation of its use for enrichment of uranium, much of the related technology was declassified and made available to the rest of the world through conferences and technical publications. Many countries developed their own electromagnetic isotope enrichment capability, but much smaller in scale. While the individual separators were similar in size and design, the number of separators, and thus the total production capability, was much smaller. Russia pursued a course similar

to the United States in their atomic weapons development program, and after developing improved techniques, converted the facility to stable isotope enrichment. During the first Gulf War in the early 1990s, it was discovered that Iraq had attempted to use this method as a part of its atomic weapons development program. The end of the war put an end to those efforts. From that lesson the process remains a nuclear proliferation concern.

### 11.3.3 Gaseous Diffusion

#### 11.3.3.1 Background of Gaseous Diffusion

After the discovery by Otto Hahn and Fritz Strassman that uranium atoms could be split when bombarded with neutrons, physicists all over the world rushed to their laboratories to confirm this finding. One was a laboratory at Columbia University. John Dunning, an experimental physicist, joined in early theoretical discussions with Enrico Fermi, Leo Szilard, Edward Teller, and occasionally Eugene Wigner. Dunning later worked with Eugene T. Booth in studying methods of diffusion. They settled on gaseous diffusion as being the best bet. British physicists came to the same conclusion at about the same time.

Dunning and Booth then began to develop the idea of gaseous diffusion. In January 1941, they could demonstrate a very-small-scale enrichment of  $U^{235}$ . The demonstration brought in support for further work on gaseous diffusion and, over the next 3 years, their program grew and expanded to the development of the process used for the K-25 plant in Oak Ridge. In December 1942, the decision was made by the management of the Manhattan District to use the electromagnetic process and have gaseous diffusion as a backup. Construction was then started on facilities for both processes in Oak Ridge, Tennessee. The electromagnetic plant was operated until 1946 because gaseous diffusion proved to be a more efficient method. The unit cost of enrichment by gaseous diffusion was less than 10% the cost of electromagnetic enrichment. Thus, gaseous diffusion became the sole process for the enrichment of uranium in the United States.

In 1945, enrichment capacity was increased by the addition of a second building in Oak Ridge, and in the early 1950s three more buildings were added to the same site. In 1952, a plant was built in Paducah, Kentucky. In 1956, yet another plant near Portsmouth, Ohio, went into production.

The end of operating the complex at full capacity to produce highly enriched uranium for the US nuclear weapons stockpile was reached in 1964. The original plant at Oak Ridge (commonly referred to as the *U* because of the mile-long building that was folded into the shape of a giant letter *U*) was shut down. The Portsmouth plant continued to produce highly enriched uranium, up to 97 wt% for the nuclear fleet reactor fuel.

The advent of commercial nuclear power reactors spawned a new mission from 1965 to 1985 for the diffusion plants—production of enrichment between 3% and 5% for domestic and foreign reactors. The productive capacity of the Oak Ridge, Paducah, and Portsmouth complex in the 1960s was 10 million kilogram SWUs per year. By 1971, the capacity had been increased to 17.2 million SWU per year.

A major program was started in early 1972 to increase the productive capacity of the diffusion complex through efficiency improvements and equipment upgrading. At the completion of the program in 1981, the capacity of the complex when at full power was 27.3 million SWUs per year. In 1974, plans were started for additional diffusion production capacity by adding a new building at Portsmouth. The diffusion equipment was to be larger than that in operation at that time. However, in 1977 this program was canceled in favor of a gas centrifuge enrichment plant to be built at the Portsmouth site.

In 1985, after 8 years of design and construction and billions of dollars spent, Washington stopped the centrifuge plant and all R&D in favor of the development of a process using lasers (atomic vapor laser isotopic [AVLIS] process) that the Lawrence Livermore National Laboratory (LLNL) argued would be cheaper than any centrifuge present or future. At the same time, June 5, 1985, Washington, faced with a falling share of the market and increasing power costs, decided to operate only the Portsmouth and Paducah plants, and shut down all Oak Ridge diffusion operations.

In 1992, a government-owned corporation, United States Enrichment Corporation (USEC) was formed and given the responsibility for the US enrichment programs, including operation of the Paducah and Portsmouth plants. In 1998, USEC was privatized. In 2002, the highly enriched operations at Portsmouth were discontinued, and in 2004 all diffusion enrichment at Portsmouth was shut down. The Paducah facility was shut down in 2013 leaving no operating gaseous diffusion-based enrichment operational in the United States.

France and China are the only other countries operating gaseous diffusion enrichment plants. The British and Russians previously operated gaseous diffusion plants. The British plant was shut down several years ago and the Russian plant was converted to a centrifuge plant. The French are underway with plans for a centrifuge plant. When USEC and the French have centrifuges operating, we will see the last of gaseous diffusion, largely because the centrifuge process requires so much less (<5%) electrical power than does diffusion.

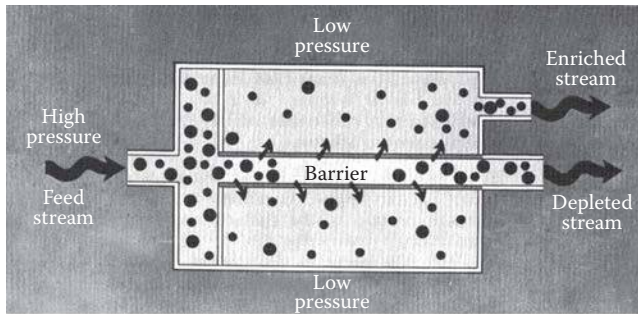
### 11.3.3.2 Gaseous Diffusion Process

The fundamental principle on which the gaseous diffusion process is based was discovered by a Scotsman, Graham, in 1829. Graham showed that the average velocities of gas molecules at a given temperature depended upon their masses. Therefore, in a gas made-up of molecules containing different isotopes, molecules containing the lighter isotope will, on average, have velocities a little faster than those containing the heavier isotope. It is this small but useful velocity difference which the gaseous diffusion process utilizes.

If the gaseous mixture is forced under pressure against a suitable porous membrane or *barrier*, more of the lighter molecules will flow through the pores due to their higher velocity. To maximize the amount of separation achieved, the barrier material must meet exacting standards so that *diffusive* flow occurs, that is, pore sizes must be such that individual molecules collide only with the pore walls rather than with each other. Diffusive flow requires not only small barrier pores (less than two-millionths of an inch in diameter) but also uniformity of pore size.

Because the amount of separation that can be realized by one pass through barrier material is dependent on the difference in mass of the molecules, very small gaseous mixtures of  $U^{235}$  and  $U^{238}$  must undergo this diffusion process many times for meaningful separation. It is the achievement of the gaseous diffusion industry that this delicate and demanding physical process was harnessed on a huge production scale to enrich uranium in a reliable manner.

The gaseous compound used to carry out the diffusion is  $UF_6$ . Various physical and chemical properties of  $UF_6$  make it the only known suitable compound. It is a solid at room temperature; consequently, the diffusion plant had to operate at temperatures and pressures to maintain  $UF_6$  in gaseous form. Although it is a stable compound,  $UF_6$  is extremely reactive with water, very corrosive to most common metals, and incompatible with organic materials (e.g., lubricating oils). This chemical activity dictates the use of metals such as nickel and aluminum, and means that the entire process flow system must

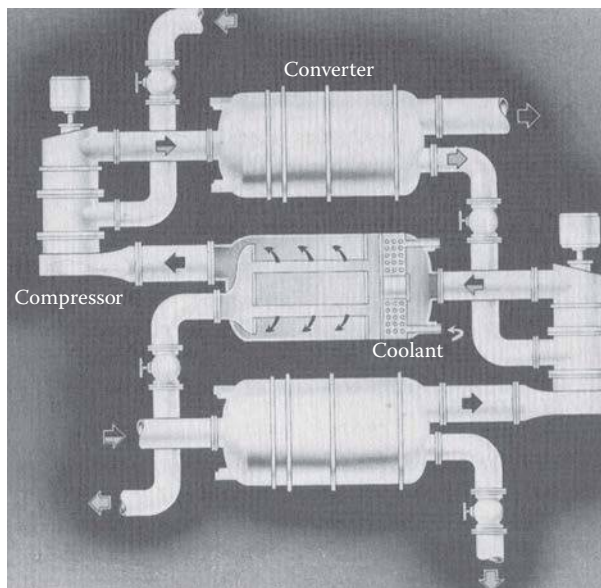


**FIGURE 11.1**  
Gaseous diffusion stage.

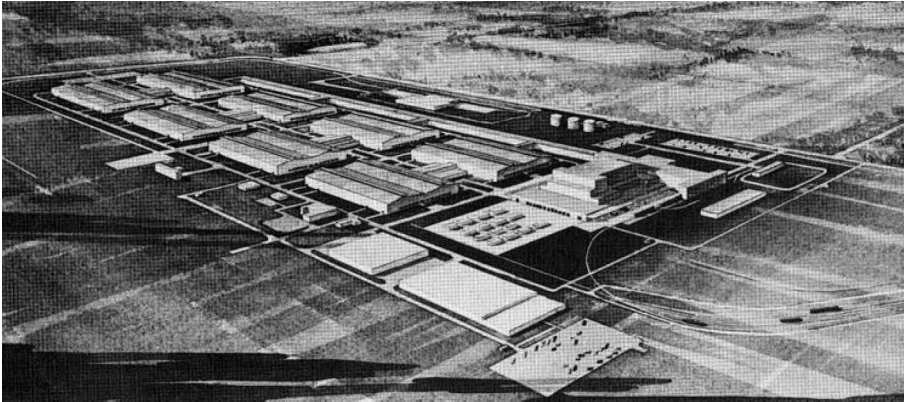
be leak-tight and clean. The corrosiveness of the process gas also makes barrier production more difficult because the barrier quality must be maintained over many years of operation.

The basic concept of gaseous diffusion is illustrated in Figure 11.1. It shows what takes place in any single stage in the process system. Gaseous  $\text{UF}_6$  is introduced into the *diffuser* or *converter* under pressure and made to flow along the inside of barrier tubes. About one-half of the gas diffuses through the barrier and is fed to the next higher stage; the remaining undiffused portion is recycled to the next lower stage. The diffused stream is slightly enriched with respect to  $\text{U}^{235}$ , and the undiffused stream is depleted of  $\text{U}^{235}$  to the same degree (Figure 11.2).

Single stages are connected together (Figure 11.3) so as to accomplish significant enrichment. Figure 11.3 also indicates the basic equipment components vital to the process. Axial flow compressors are used in the larger stages to compress the gas to maintain the inter-stage flow. The compressors are electrically driven. A gas cooler is incorporated in each



**FIGURE 11.2**  
Gaseous diffusion stage arrangement.



**FIGURE 11.3**  
Conceptual drawing of the DOE-planned Portsmouth centrifuge plant (1980).

converter to remove the heat of gas compression. Groups of stages are coupled in this way to make up operating units, which in turn make up the entire *cascade*.

Viewed as a whole,  $UF_6$  feed with a  $U^{235}$  assay of 0.71% is introduced into the middle of the cascade. From there,  $U^{235}$  selectively moves toward the top of the cascade and is withdrawn as *product* at various points depending upon the desired enriched level. Similarly, the  $U^{238}$  selectively moves toward the bottom of the cascade and is withdrawn as deleted or *tails* material. This depleted material contains about 0.2%  $U^{235}$  and has been stored at all three plants since operations began for possible future use.

### 11.3.4 Gas Centrifuge

#### 11.3.4.1 Background of Gas Centrifuge Enrichment

The idea of using centrifugal or gravitational forces to separate a mixture of gases of different molecular weights is very old. Experiments to this end were conducted in Germany as early as 1895. The extension of the method to separate isotopes had been suggested soon after researchers had discovered their existence. In 1919, the physicists Lindemann and Aston suggested that centrifuges might be used to separate isotopes, but it was not successfully accomplished until 1935 by Jesse Beams at the University of Virginia. He succeeded in separating the isotopes of chlorine and, with the 1939 German discovery of nuclear fission, went to work quickly on uranium. By 1941, he had enriched gram amounts of uranium to 5%  $U^{235}$ .

In mid-1942, the National Academy of Sciences, being asked for their assessment of what processes should be used in the Manhattan Project, recommended gas centrifuge and gaseous diffusion. Beams had more advanced hardware and experimental data than the Columbia Diffusion Team or the California Electromagnetic Team. A pilot plant was designed, constructed, and operated by industrial teams at the Baytown, New Jersey, refinery of Standard Oil. There were 24 commercial centrifuges of diameter 7.2", 42" long rotors, with an overall height four times that long because of the complex bearing and gas systems. Some ran successfully, some failed. The Manhattan experts toured the facility and talked to the engineers but decided the mechanical complexities of the many thousands of machines needed were too great. So the centrifuge program was stopped.

In 1958, there was a revival of the gas centrifuge brought about by the work of the Austrian Gernot Zippe. He had been working in the USSR on an R&D effort on gas centrifuge and

had come up with a revolutionary mechanical design. Beams heard of this and talked Zippe into coming to the United States in 1958 to work with him.

In 1960, under an AEC contract, R&D of the centrifuge began at the K-25 site in Oak Ridge by a team of engineers from Union Carbide (operating contractor to AEC), University of Virginia, and a little later Garrett AI Research of Torrance, California. Over the next two decades, significant advances led to much longer, faster, and more efficient machines and to a full-scale demonstration that was convincing enough for President Carter to announce in his energy message in April 1977 that the next US uranium endeavor would be a gas centrifuge plant instead of an *add-on* gaseous diffusion facility as originally planned. It was later determined that the centrifuge plant would be built at Portsmouth, Ohio, adjacent to the gaseous diffusion plant (Figure 11.3).

Development work continued to further increase the output and reduce the cost of the machines while construction of the plant proceeded. However, in June 1985, the Department of Energy (DOE) made the decision (after an expenditure of nearly \$3 billion) to shut down the centrifuge plant and concentrate on the development of the AVLIS process for uranium enrichment at LLNL.

The Energy Act of 1992 established a government corporation, USEC, for the enrichment operations that had been under the DOE. In 1998, USEC became completely private. USEC took over the continuing development of AVLIS. In 1999, USEC discontinued all work on AVLIS and began an investigation of alternative methods of enrichment. In 2002, the decision was to reactivate the centrifuge program. An R&D program was initiated to improve upon the machine using latest materials and technologies. The centrifuge plant will be located at the site of the past Portsmouth, Ohio, diffusion plant and will use very large buildings that were put up for the DOE centrifuge program in the 1980s. A lead cascade test program began at the site in August 2007. The purpose of the lead cascade is to validate the feasibility of closed-loop cascade operations and demonstrate the capability of the technology to produce LEU at commercial product assay levels.

USEC has released the information that its centrifuge rotor will be 40' high, about 2' in diameter, and made of tightly woven carbon fibers. The rotational speed remains a secret. The machine capacity will be 350 SWUs per year. The announced total centrifuge plant output will be 3.8 million SWUs per year. Complete plant operation was scheduled to be achieved in 2012. Financial and technical problems significantly delayed the program. With the assistance of DOE the lead cascade was completed and successfully operated in 2013. However, the program is still under the cascade license which does not allow for commercial operation.

There will be two other enrichment plants in the United States. One is under construction in Eunice, New Mexico, by Louisiana Energy Services (a US company formed by URENCO Enrichment Company). The announced capacity of this plant is 3 million SWUs per year starting in 2010. URENCO is a European private sector consortium with plants in Germany, the United Kingdom, and the Netherlands. URENCO was the first commercial entity to enrich uranium on a production scale (centrifuges started in 1977) and is currently the world's largest single provider of centrifuge-enriched uranium. The other centrifuge plant is being planned by Enrichment Technology Company, a US division of Areva, a French company. The announced planned capacity of this plant is 3 million SWUs per year starting in 2013. The location site has not been determined.

The URENCO USA facility began operations on June 11, 2010. Construction of the project will continue until the plant reaches the planned 5700 tSW/a capacity. URENCO USA is the first enrichment facility to be built in the United States in 30 years and the first ever using centrifuge enrichment technology. At the end of December 2014, capacity at

URENCO USA stood at 3700 tSW/a. It is anticipated that at full capacity the facility can produce sufficient enriched uranium for nuclear fuel to provide approximately 10% of America's electricity needs.

Other countries using (or planning to use) centrifuge technology are Russia, Japan, China, North Korea, and Brazil. The first centrifuge plant in France is under construction with full operation scheduled for 2009. Iran has had a gas centrifuge plant under construction for several years. The existence of the site became publicly known in August 2002. According to the International Atomic Energy Agency (IAEA), the Iranian centrifuges are based upon first-generation URENCO designs.

#### 11.3.4.2 Gas Centrifuge Process

A gas centrifuge comprises an evacuated casing containing a cylindrical rotor which rotates at high speed in an almost friction-free environment and into which gaseous  $\text{UF}_6$  is fed. It works with the small difference in molecular weight between  $\text{UF}_6$  molecules containing the  $\text{U}^{235}$  atom and those containing the  $\text{U}^{238}$  atoms. The difference is exploited by imposing an artificial gravitational field on the gaseous mixture of the two isotopes, creating a radial concentration gradient which then becomes an axial gradient with countercurrent flow on the inner part in one direction and along the wall in the opposite direction. The degree of enrichment in a single centrifuge machine is dependent upon the mass difference of the isotopes being separated, the length of the rotator, and the speed of rotation. The principle of a gas centrifuge is illustrated in Figure 11.4.

The  $\text{UF}_6$  introduced near the center of the rotor accelerates to approximately the speed of the rotor. Centrifugal force causes the heavier  $\text{U}^{238}$  molecules to move closer to the wall of the rotor, producing partial separation of the  $\text{U}^{235}$  and  $\text{U}^{238}$  isotopes. This separative effect is increased by an axial countercurrent flow of gas within the rotor. The enriched  $\text{UF}_6$  is removed through a scoop at the top and the depleted through a scoop at the bottom.

Because the desired enrichment is not obtained in a single centrifuge, several machines must be connected in a series, known as a *cascade* (Figure 11.5).

Only a small amount of gas will flow through a single centrifuge. Therefore, many machines must be connected in parallel to achieve the total flow necessary for a large capacity plant.

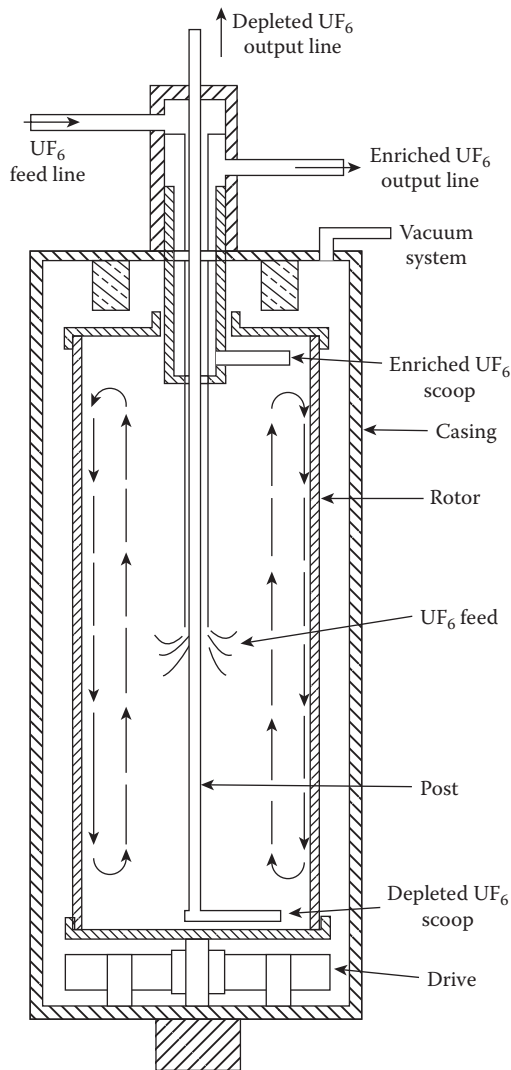
Even though the separation achieved per machine is much greater than in a diffusion stage, it is much smaller than needed for most applications. Many machines must therefore also be connected in series to achieve the total enrichment required. In a centrifuge cascade, the largest number in parallel will be at the feed point with decreasing numbers as the enrichment levels increase toward the desired product enrichment.

The separation factor for a centrifuge varies with the fourth power of the speed, thus the desire to operate at the highest speed possible. This in turn means that the rotor must be made of a material that has the highest strength-to-density ratio possible. USEC has released the information that the rotor in the *American Centrifuge* is made of tightly woven carbon fiber.

#### 11.3.5 Laser Isotope Separation

##### 11.3.5.1 Laser Isotope Background

The invention of lasers in the early 1960s made possible the laser isotope separation (LIS) approach. The first US research began in the 1960s at the Los Alamos National Laboratory (LANL); lasers were used to excite the  $\text{U}^{235}$   $\text{F}_6$  molecules in a  $\text{UF}_6$  stream, a process they called *molecular laser isotope separation* (MLIS). Their work continued for a decade or more,

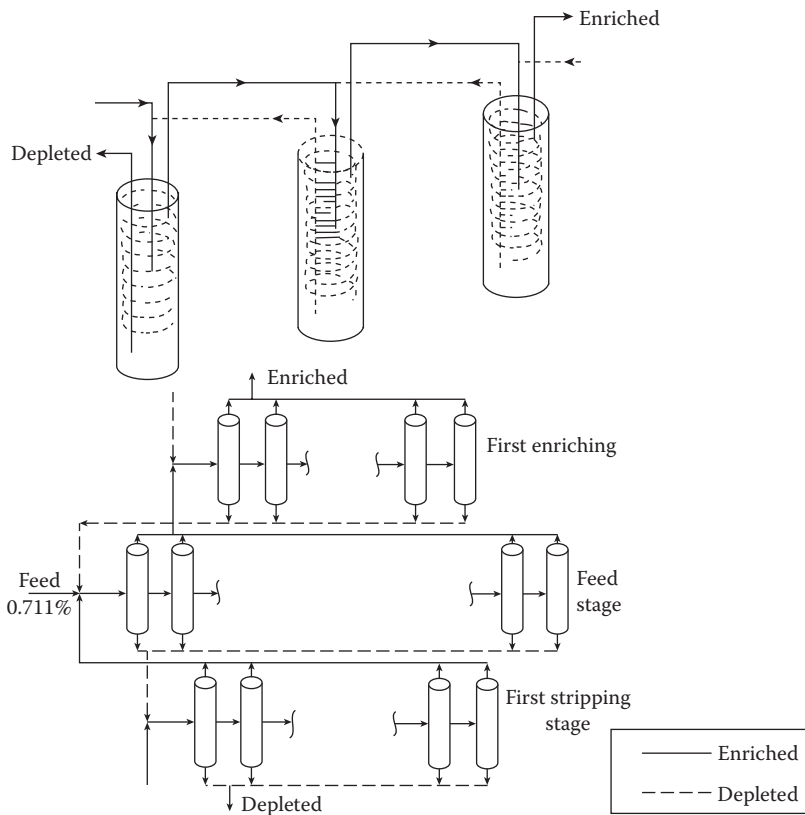


**FIGURE 11.4**  
Cutaway drawing of a gas centrifuge.

showed promise, but was phased out during the late 1970s. Not long after the MLIS work was underway, the LLNL started research on a complementary approach using lasers to excite  $U^{235}$  atoms in a stream of uranium metal vapor (AVLIS). Their work progressed over the next 30 years, being finally phased out in 1999. There was one private venture by Exxon Nuclear Company. They worked on laser enrichment technology for approximately 7 years before phasing out.

In France, research started soon after on a program (SILVA). It was reported in 1975 that enriched uranium had been produced in small quantities. In Germany, a small-scale research program was started in 1971. Israel was doing research on laser separation in 1973 but kept it under wraps. However, convincing reports were that Israel had a cheap but sophisticated laser process and later received a German patent. Japan initiated its research program in 1975 and reported in 1977 that they had collected a microgram of uranium at





**FIGURE 11.5**  
Configuration of gas centrifuges.

a concentration of 35%  $U^{235}$  in a successful experiment. Italy stated research work in 1977. The Soviet Union was known to be working on LIS and in 1979 published articles on uranium including LIS. However, the most active programs were in the United States.

In 1978, the US Congress clamped a secrecy label on laser separation of uranium retroactive to 1976 due to concerns that, because of costs of construction and operations being much lower for a laser separation plant than for diffusion and centrifuge plants, enrichment plants could be available to small nations and lead to proliferation. Funds were cut from the Federal energy budget for laser enrichment.

The Energy Research and Development Agency (ERDA), the forerunner to the DOE, through the late 1970s to 1981 supported the study of three new experimental processes for uranium enrichment. Two were based upon laser separation, and one on plasma separation. Jersey Nuclear-Avco Isotopes Incorporated (subsidiary of Exxon) and the LLNL worked on atomic uranium vapor. LLNL referred to it as AVLIS. The LANL and a group at Exxon Research Laboratories (not connected with Jersey-Avco) worked on molecular  $UF_6$ . TRW Incorporated pursued research work on a plasma separation process. Union Carbide Nuclear Division (UCC-ND) supported each in their efforts. In 1981, the AVLIS process at LLNL was selected as the process to be developed further and the other processes were subsequently phased out.

Throughout the first half of the 1980s, there was a continued erosion of the US share of the enriched uranium market as a result of stiff competition from French EURODIF and URENCO. The US monopoly in the early 1970s dropped to 47% in the next decade.

At prodding from Congress, the DOE began to address the issue of how to best provide technology for future competition. In mid-1984, the process was begun with the appointment of a process evaluation panel which was given the task of recommending to the Secretary of Energy which of the two advanced technologies (advanced centrifuges or AVLIS) was to be the replacement for the diffusion operations. There followed a string of debates between teams from each side before the panel, which culminated in the selection of AVLIS as the favored program for the future. The announcement by the Secretary of Energy was that concentration thereafter was to be on the development of AVLIS as the sole advanced enrichment technology and included shutting down as soon as possible the gaseous diffusion plant at Oak Ridge, Tennessee, and the immediate discontinuation of the gas centrifuge plant program at Portsmouth, Ohio, and development work on centrifuge processes.

The advantages of AVLIS over gaseous diffusion and gas centrifuge were seen as:

- Requirement of only a single step to enrich uranium compared with the sequential steps required in gaseous diffusion and gas centrifuge
- Ability to extract nearly all of the  $U^{235}$  from the feedstock
- Ability to reprocess the depleted  $UF_6$  resulting from the diffusion operations
- Lower cost plant to construct and operate

DOE supported the development work of AVLIS at LLNL until 1994 when the USEC became an entirely private company. At that time, DOE turned the program over to USEC for continuation. In the first half of 1999, USEC gave up on AVLIS and started to search for a replacement.

Upon the closure of AVLIS, the only remaining laser process on the world stage was (separation of isotopes by laser excitation [SILEX]), a molecular separation process developed by the Australian company Silex Systems Limited. The French had ceased work on their laser program, SILVA, in 2003.

In 1996, USEC secured the rights to evaluate and develop SILEX but relinquished those rights in 2003, having earlier decided to reopen the advanced centrifuge program.

In early 2006, General Electric (GE) announced the signing of an exclusive agreement with Silex Systems Limited to license the technology and develop the next-generation low-enriched uranium manufacturing process in the United States. The transaction will require regulatory controls and government approvals. GE has had meetings with the US Nuclear Regulatory Control (NRC) and the license application was submitted in 2009 and approved by the USNRC in September of 2012. The first phase would be a test loop in GE's existing fuel manufacturing facility in Wilmington, North Carolina.

The Silex fuel technology exclusively licensed to GE in 2006 was later licensed to Global Laser Enrichment (GLE) which is made up of GE, Hitachi, and Cameco. GLE now has an NRC license to construct and operate a laser enrichment plant in Wilmington, North Carolina. Also, GLE and DOE are currently negotiating a commercial contract to establish a uranium tails processing facility based on the Silex process at the closed down gaseous diffusion enrichment plant at Paducah, Kentucky.

#### **11.3.5.2 Laser Separation Processes**

The gaseous diffusion and gas centrifuge techniques exploit the small mass differences between  $U^{235}$  and  $U^{238}$  in the gaseous form of  $UF_6$ . AVLIS is based on an entirely different concept.  $U^{238}$  and  $U^{235}$  isotopes have different electron energies, so that they absorb

different colors of light. Lasers are tuned to emit a carefully chosen combination of colors that will be absorbed by only  $U^{235}$ . The laser-excited  $U^{235}$  atom emits an electron and becomes a positively charged ion. The  $U^{235}$  ions are then separated from neutral  $U^{238}$  using electromagnetic fields.

The AVLIS process consists of a laser system and a separator system. The latter contains a vaporizer and a collector. The working medium is metallic uranium that is melted and vaporized to form an atomic vapor stream. The vapor stream flows through the collector where it is illuminated by precisely tuned laser light. The selected atoms become charged by photoionization and are removed from the vapor stream by an electronic field.

The laser system is a laser-pumped laser, that is, one laser system is used to energize a second, which finally produces the light used in the separation process. This allows the separation of the requirements for efficiency and color precision. The large pump lasers are reasonably efficient, whereas the dye lasers (which convert the pump laser light to process light) are tunable, reliable, and commercially available.

When the laser light illuminates a stream of uranium vapor, the  $U^{235}$  vapor absorbs the light but the  $U^{238}$  does not. The now excited  $U^{235}$  ejects an electron, thus becoming a positively charged atom or ion which is deflected by an electromagnetic field to the product collector. The  $U^{238}$  remains uncharged and passes through the collector section to the tails collector.

The vaporization of the uranium is accomplished by means of an electron beam that creates an atomic  $U^{235}/U^{238}$  vapor stream.

MLIS uses  $UF_6$  as its feedstock, thereby fitting more readily into the conventional fuel cycle than AVLIS. There are two steps involved in the MLIS process: excitation with infrared lasers and then dissociation with an ultraviolet laser. Gaseous  $UF_6$  mixed with a carrier gas (argon) is expanded through a nozzle that cools the gas to low temperatures. The  $UF_6$  is irradiated by infrared lasers, which selectively excite the  $U^{235}F_6$ , leaving the  $U^{238}F_6$  unexcited. Photons from an ultraviolet laser then preferentially disassociate the excited  $U^{235}F_6$  to form  $U^{235}F_5$  and free fluorine atoms. The  $U^{235}F_5$  formed in this manner precipitates from the gas as a solid powder which can be filtered from the gas stream.

### 11.3.6 Plasma Separation

The plasma (fluid or vapor composed of charged particles) method of separation is based upon the fact that the cyclotron frequency of an ion in a magnetic field is a function only of the ion mass and the magnetic field strength. Accordingly, by exciting plasma comprising uranium ions and electrons at the cyclotron frequency of the  $U^{235}$  ions, it is possible to *pump energy* into the  $U^{235}$  selectively and thereby increase the orbit diameter of the  $U^{235}$  ion relative to that of the  $U^{238}$  ion. Collection schemes are based on the difference in orbit diameter between the  $U^{235}$  and  $U^{238}$  ions as they traverse the length of the plasma orbiting around the field lines of a high-strength solenoidal magnet. The excitation of the  $U^{235}$  is accomplished by placing in this plasma a radiofrequency antenna tuned to the  $U^{235}$  cyclotron frequency. The separation performance of a single separator is adequate enough to cover the isotopic gradient of interest so that staging will not be required.

### 11.3.7 Aerodynamic Processes

There have been two aerodynamic separation methods developed and tested. One, in Germany, is the separation nozzle, referred to as the *Becker jet nozzle*. The other is the vortex tube separation technique developed in South Africa. In both processes, the separation

results mainly from centrifugation induced in a gas stream deflected by a duly shaped stationary wall.  $\text{UF}_6$  is used in both cases. Enhancement of centrifugation is achieved by dilution of the  $\text{UF}_6$  with a light gas (hydrogen or helium). Addition of the light gas in molar excess is to increase the  $\text{UF}_6$  flow velocity attainable at a given pressure ratio. Negative features are high energy requirements and removal of waste heat problems. Neither method has been in use for some time.

### **11.3.8 Chemical Exchange**

The French did investigative work on chemical exchange in the last half of the 1970s but found the separation effect for uranium to be too small for practical application. There is no current use of this method for uranium.

# 12

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## *Nuclear Fuel Fabrication*

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McLean T. Machut

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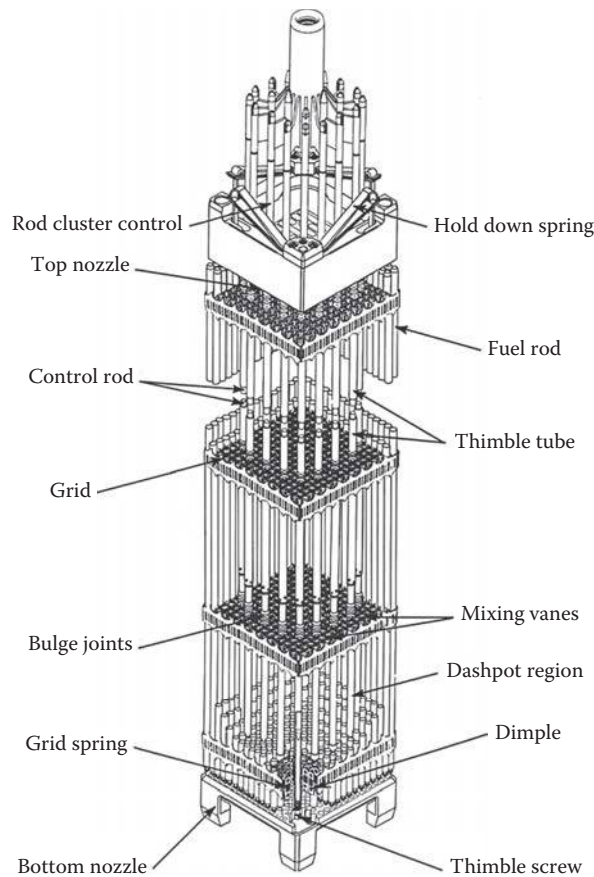
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### 12.1 Introduction

The purpose of this chapter is to provide a general insight into the manufacture of fuels used in nuclear reactors. The primary focus will be on uranium dioxide (UO<sub>2</sub>) fuels for light water reactors (LWRs), including both the pressurized water reactor (PWR) and the boiling water reactor (BWR). Many of the details relating to the fuel for these reactors are also presented in Sections 1.2 and 1.3 of this handbook. Some of the information from those chapters will be repeated for clarity.

A fuel assembly (also called a fuel bundle) consists of an array of fuel rods (also called fuel pins) which consist of cylindrical cladding tubes containing a stack of UO<sub>2</sub> fuel pellets and welded closed at each end. These rods are supported by a structural cage (or skeleton) consisting of grids, guide tubes (also, guide thimbles), and end fittings (also known as tie plates or nozzles). Grids provide support, maintain separation between fuel rods, and in some cases provide mixing of the coolant. End fittings provide the support for the guide tubes, interface with the reactor core plates, and can act as a debris filter.



**FIGURE 12.1**  
Typical fuel assembly, present generation of pressurized water reactors.

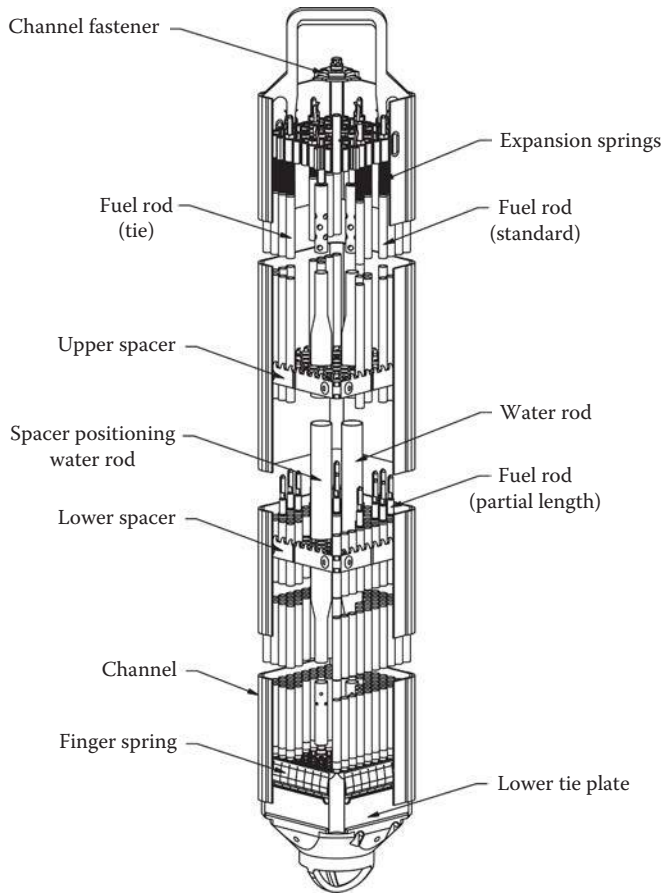
In PWR fuel, guide tubes provide structure and allow for insertion of instrumentation and control rods. A current-generation PWR fuel assembly is shown in Figure 12.1.

BWR fuel uses a square fuel channel (or shroud) which surrounds and supports the fuel rods and grids while separating flow channels. A BWR fuel assembly is shown in Figure 12.2. Figure 12.3 shows a cross section of a BWR fuel module that consists of four assemblies with a cruciform shaped control blade in between them.

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## 12.2 Fuel Fabrication

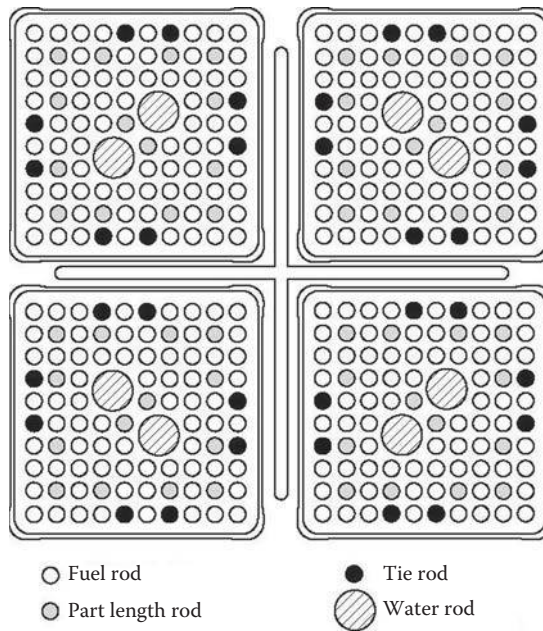
The fuel fabrication process is shown schematically in Figure 12.4 from conversion of enriched uranium hexafluoride ( $UF_6$ ) gas to the final assembled product. The enriched material is received from an enrichment facility in the form of  $UF_6$ . Figure 12.5 shows the original gaseous diffusion plant in Oak Ridge, Tennessee. This facility is no longer in



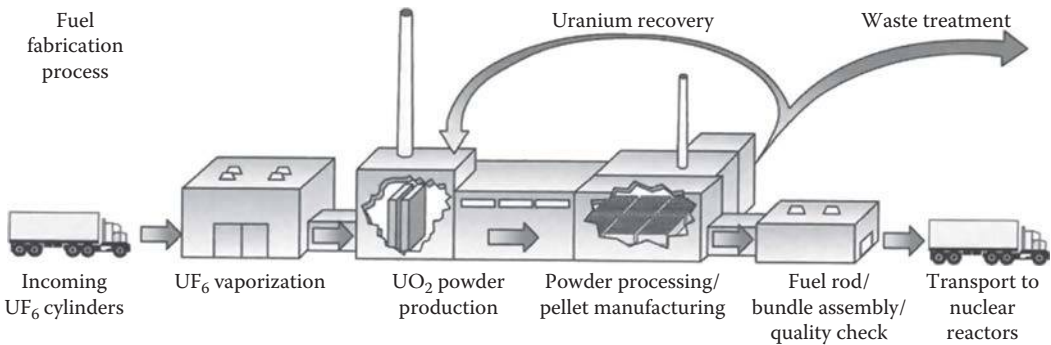
**FIGURE 12.2**  
GE14 fuel assembly.

operation, but many cylinders of enriched  $\text{UF}_6$  were shipped from here to fuel fabrication facilities. Today, there are more modern technologies in use for the enrichment of uranium, including banks of centrifuges and laser separation.

The  $\text{UF}_6$ , which is a solid at room temperature, in depleted and natural forms is shipped in large cylinders and in smaller containers after being enriched to a maximum of 5% U-235 by weight. At the fuel fabrication facility, the  $\text{UF}_6$  must be converted to uranium dioxide,  $\text{UO}_2$ , which is the basic material that is used for fuel in LWRs. Then the  $\text{UO}_2$  powder is formed into cylindrical pellets, sintered, and ground into final shape. A rigorous inspection process ensures that pellets are free from surface imperfections and other defects which could later manifest themselves as fuel failures during operation. The finished pellets are then loaded into the fuel cladding, and the ends are welded closed to make up the fuel rods. The process is summarized in Table 12.1. The key safety consideration during this process is maintaining the material in a criticality-safe configuration at all times.



**FIGURE 12.3**  
Fuel module (cell).



**FIGURE 12.4**  
Fuel fabrication process (schematic).

### 12.2.1 Fuel Fabrication Facilities

Fuel fabrication facilities are located in many parts of the world, as shown in Table 12.2, where the countries and the annual fuel production capacities are listed. These facilities range from sites which can handle all steps including  $UF_6$  processing,  $UO_2$  powder blending, pelletizing lines, and specialized component manufacture, to sites that are responsible for final assembly only.

Whether the components are sourced from other facilities or manufactured within, when the final assembly is complete, the fuel is ready to be shipped to a nuclear power plant and inserted into a reactor during a scheduled refueling outage. Fuel fabrication facilities used for commercial power reactors are large, industrial-scale manufacturing





**FIGURE 12.5**  
Oak Ridge Gaseous Diffusion Plant in Oak Ridge, Tennessee.

**TABLE 12.1**  
Process Flow Chart for UO<sub>2</sub> Pelletizing

Enriched UO <sub>2</sub> Pellets	
<input type="checkbox"/>	UF <sub>6</sub> receiving
<input checked="" type="checkbox"/>	Vaporization, hydrolysis, defluorination, reduction, and inspection (chemical properties)
<input checked="" type="checkbox"/>	Sieve, homogenization, and inspection (powder visual properties, chemical and physical properties)
<input checked="" type="checkbox"/>	Blend and poreformer/U <sub>3</sub> O <sub>8</sub> addition, precompaction, granulation, and inspection (enrichment and physical properties)
<input type="checkbox"/>	Spheroidization and dry lubricant addition
<input type="checkbox"/>	Press and sinter pellet
<input checked="" type="checkbox"/>	Inspection (density, pellet chemistry)
<input checked="" type="checkbox"/>	Grind and inspect (diameter, length, chamfer, surface finish, perpendicularity, visual integrity, and pellet chemistry)
<input type="checkbox"/>	Transfer to rod load

Notes: • Inspection/QA processes,  Manufacturing/assembly processes.

facilities. For example, the Global Nuclear Fuel facility in Wilmington, North Carolina, has 600,000 ft<sup>2</sup> of manufacturing space. AREVA, Inc. and Westinghouse have similarly large fuel fabrication facilities in Richland, Washington, and in Columbia, South Carolina, respectively. Many of the specific details related to the manufacturing process are proprietary to the companies that perform this work, and those details will not be included in this general description.

### 12.2.2 Fuel Pellet Manufacturing

#### 12.2.2.1 Uranium Conversion

The enriched UF<sub>6</sub> is converted to UO<sub>2</sub> in two basic steps. Initially, UF<sub>6</sub> is vaporized by slowly heating the transportation cylinder and then hydrolyzed in a gas phase by reacting it with steam to form uranium oxyfluoride (UO<sub>2</sub>F<sub>2</sub>) powder. The UO<sub>2</sub>F<sub>2</sub> powder is

TABLE 12.2

Light Water Reactor Fuel Fabrication Facilities Located World Wide<sup>a</sup>

Country	Owner/Controller	Plant Name/Location	Nominal Capacity (MTU/Year)
Brazil	FEC (INB)	Resende	100
China	CNNC	Yibin	400
France	FBFC (AREVA)	Romans-sur-Isère	1400
Germany	Advanced Nuclear Fuels	Lingen	650
India	DAE Nuclear Fuel Complex	Hyderabad	25
Japan	Global Nuclear Fuel LLC, Ltd.	Yokosuka City	750
	Mitsubishi Nuclear Fuel Co., Ltd.	Tokai-Mura	440
	Nuclear Fuel Industries, Ltd.	Kumatori	284
Kazakhstan	Ulba Metallurgical Co. (90% Kazatomprom)	Tokai-Mura	200
		Ust-Kamenogorsk	2800
South Korea	KEPCO Nuclear Fuel Co. Ltd.	Taejon	550
Pakistan	Pakistan Atomic Energy Commission	Kundian	20
Russia	JSC TVEL	ELEMASH, Elektrostal	1020
		NCCP, Novosibirak	1000
Spain	ENUSA	Juzbado	400
Sweden	BNFL/Westinghouse Atom	Västerås	600
United Kingdom	Springfields Fuels, Ltd.	Springfields, Lancashire	330
United States	AREVA, Inc.	Richland, Washington	1100 <sup>b</sup>
	Westinghouse	Columbia, South Carolina	1150
	Global Nuclear Fuel—Americas, L.L.C.	Wilmington, North Carolina	1200

<sup>a</sup> Data extracted from [www.wise-uranium.org/efac.html](http://www.wise-uranium.org/efac.html). Closed plants omitted.

<sup>b</sup> Includes capacity transferred from Lynchburg, Virginia.

then defluorinated with additional steam and reduced to  $\text{UO}_2$  by hydrogen in a rotary kiln. HF is the by-product of this reaction process. After cooling, the powder is discharged into large storage containers that maintain the powder in a criticality-safe geometry.

### 12.2.2.2 Powder Preparation

Powder preparation for pressing consists of four phases: homogenization; blending; pre-compaction and granulation; and spheroidization and lubrication.

- *Homogenization*— $\text{UO}_2$  powder from the conversion process is homogenized in a large, criticality-safe mixer with an orbital-arm screw to achieve uniformity of ceramic characteristics in batches that may exceed 2000 kg.
- *Blending*—Using a conical mixer, enrichment and additive blending of batches of more than 500 kg are performed in this operation. Dry, recycled  $\text{U}_3\text{O}_8$  and pore-former are added at this stage, as necessary. These additions are made to enhance the properties of the powder for further operations. The  $\text{U}_3\text{O}_8$  is generated from material that is recovered during processing of the powder. The fine  $\text{U}_3\text{O}_8$  is added

back to the in-process powder in controlled amounts to achieve an acceptable pellet structure and desired density.

- *Precompaction and granulation*—The powder is precompacted in a rotary press, and then forced through a screen (granulation) to form granules. This step improves the flowability of the powder to aide in the pelletizing process. The outlet granulated powder is discharged into bicone-shaped containers.
- *Spheroidization and lubrication*—In the first step, the entire bicone is tumbled to spheroidize the powder granules through abrasion and the adhesion of fine powder particles, further improving the powder flow behavior. In some processes, a lubricant is also added to enhance the flowability of the powder during future handling steps.

After these steps, the powder is transferred to storage drums for transfer to pelletization. At each step in the process, careful attention is paid to the physical, chemical, and nuclear properties of the powder. The quality assurance process ensures that the final material will provide for safe operation of the reactor in which the fuel is used.

Some reactor fuel also contains material called a *burnable poison*. This burnable poison material is added to the powder during the blending process. A material such as gadolinium has a large neutron cross section or ability to absorb neutrons when it is in the reactor core. Once it has absorbed a neutron, it is transmuted to a material that no longer acts as a poison. This poison allows more fissile material to be initially loaded into the reactor core and, as the poison material is burned away, more reactivity will be available for ongoing fission reactions. The end result is that the energy production can be more evenly distributed between fuel assemblies, which helps core designers increase the neutron utilization of the core while continuing to meet safety limits. In some cases, the operating time between refueling for the reactor may be extended through the use of such burnable poisons, allowing for a reduction in the number of outages over a period of time. An alternate scheme for reactivity control is through use of integral burnable absorbers, where a coating containing the neutron poison Boron-10 is applied to the pellet prior to fuel rod loading. The transmutation of B-10 produces additional helium gas during operation, which often requires the use of annular *blanket* pellets to avoid excessive fuel rod pressures.

### 12.2.2.3 Fuel Pelletization

The blended and prepared powder is pressed into *green* pellets using a mechanical die. Sintering is then performed in a reducing atmosphere at temperatures of  $>1750^{\circ}\text{C}$  to achieve a high density, which is verified by gamma densitometry. The stoichiometry, grain size, porosity, density, and other properties of the fuel pellet are carefully controlled during this step. In the early 1970s when pellets were made to  $\sim 92\%$ TD (theoretical density), it was determined that during the early stages of irradiation the fuel pellets continued to become more dense, which is known as *resinter densification*. This created a larger pellet-to-cladding gap that could cause creep collapse of the rod or power spikes due to gaps in the fuel stack and potentially lead to fuel failures. Modern fuels are made to  $95\%–97\%$ TD so this effect is lessened.

All pellet processes are typically performed dry, although some manufacturers have used wet grinding processes with an additional drying step, with strict control over the contacting materials to preclude contamination with water or hydrocarbons. The pellets

are ground to strict diameter requirements. A typical fuel pellet is shown in Figure 12.6. The pellets used in the current generation of PWRs typically range from 8.1 to 9.7 mm in diameter and those used in BWRs are typically 8.2 to 10.6 mm, depending on reactor design and the array of fuel rods in the bundle. Following grinding, the pellets are inspected for diameter, length, end squareness, chamfer, surface finish, chips, pits, and surface inclusions. Trays of completed fuel pellets are shown in Figure 12.7. Finally, the pellets are transferred to storage, where they are held until they are needed in the fabrication of the fuel rods.



**FIGURE 12.6**  
Single fuel pellet prior to insertion into a fuel rod.



**FIGURE 12.7**  
Stacks of completed nuclear fuel pellets being moved in a fuel fabrication facility.

**TABLE 12.3**

## Process Flowchart for Fuel Rod Assembly

- 
- First weld end plug and release (lot verification, U/T results review, visual checks, end plug parallelism, and weld diameter)
  - Rod load and inspection (weight, cleanliness, column length, zone position, and scale calibration)
  - Final end plug weld and inspection (weld integrity, U/T, rod pressure, visual checks, end plug parallelism, and weld diameter)
  - Rod inspection—scanner (enrichment zones, zone grams U, gaps between pellets, and zone interface position)
  - Fuel rod QA release (fuel rod process verification, weld records review, traceability records review, and certification and release entry)
  - To bundle assembly
- 

Notes: • Inspection/QA processes,  Manufacturing/assembly processes.

### 12.2.3 Fuel Rod Fabrication

Details of the fuel rod fabrication process are shown in Table 12.3. The primary components for a fuel rod are the cladding and the fuel pellets. A number of minor components are also required for the fuel rod. These could include a spring to keep the stack from forming a gap during shipping and handling and the end plugs that are welded to the cladding tubing. Some of these end plugs have specially designed features to assist in assembly loading or gripping of fuel rods. Most of these small parts are machined either in house or by a third-party supplier.

#### 12.2.3.1 Fuel Rod Cladding Materials

Most LWR fuel rod cladding is made of Zircaloy (and its derivatives), which is an alloy of primarily zirconium and tin. Other alloying elements include niobium, iron, chromium, and nickel. Zircaloy was chosen because it has a very low cross section for thermal neutrons. Naturally occurring zirconium contains about 1%–5% hafnium. The hafnium must be removed because it has a very high thermal neutron cross section and is often used in making control rods for reactors. The separation process used in the United States is a liquid–liquid extraction process. It is based on the difference in solubility of the metal thiocyanates in methyl isobutyl ketone. In Europe, a process known as *extractive distillation* is used to purify zirconium. This method employs a separation solvent that interacts differently with the zirconium and hafnium, causing their relative volatilities to change. This enables them to be separated by a normal distillation process. The separated zirconium is then alloyed with the required constituents.

BWRs utilize Zircaloy-2, which normally consists of 98.25 weight % zirconium with 1.45% tin, 0.10% chromium, 0.135% iron, 0.055% nickel, and <0.01% residual hafnium. Some BWRs also use a cladding liner made with iron to improve pellet-to-cladding mechanical interaction (PCMI) during power maneuvers.

The zirconium alloys used in PWR nuclear components consist of both zirconium–tin alloys and zirconium–niobium alloys. PWRs have historically used Zircaloy-4, which consists of 98.23 weight % zirconium with 1.45% tin, 0.21% iron, 0.1% chromium, and 0.01% hafnium.

Zirconium has a high affinity for hydrogen. Absorption of hydrogen leads to hydrogen embrittlement, which can lead to failure of the zirconium tubing used as cladding for

nuclear fuel. Zirconium also reacts with oxygen. During operation a passive oxide layer is formed on the Zircaloy components, giving the material corrosion resistance. If this layer gets too thick, the mechanical integrity is diminished to the point that the layer spalls away and the corrosion rate increases exponentially, a condition known as breakaway oxidation. Developments of advanced cladding alloys have seen significant progress in recent years and have reduced fuel cladding corrosion rates versus Zircaloy-4. Westinghouse calls its advanced alloy *Optimized Zirlo* and AREVA identifies theirs as *M5*. For example, *M5* has also been shown to have very low hydrogen pickup when compared to other alloys in the industry, which improves behavior of the fuel under severe accident scenarios such as a LOCA (loss-of-coolant accident).

### **12.2.3.2 Fuel Rod Loading**

Each fuel rod carries a unique identifying mark as a quality assurance function in material control and traceability. One manufacturer uses the lower end plug of each rod and another uses a barcode on the cladding. The end plug is welded to the fuel cladding at an automated weld and inspection station, which performs 100% dimensional and flaw inspections for each weld. The weld parameters used are qualified to match the dimensions of the tube and the specific materials configuration of the tube. The end plug identification and tube bar code are both read at first weld, at which point a traceability linkage is established and maintained via a computer system.

After the lower end plug is welded to the cladding, the fuel pellets are loaded into the rod. Some manufactures mechanically push the pellet stack into the rod, whereas others use a vibratory loading process. All pellet and rod processing is performed dry and in a strictly controlled temperature and humidity environment to avoid contamination by hydrogenous impurities.

### **12.2.3.3 Fuel Rod Welding and Inspection**

The loaded rod is then moved to the final weld and inspection station, where a plenum spring is first inserted. The rod is evacuated, backfilled with helium to ~20 atmospheres, and the upper end plug is welded to the rod.

Resistance welding is now routinely used throughout the nuclear industry as a means of attaching end plugs to fuel rods. This process eliminates virtually all of the end plug weld failure mechanisms experienced over the years. This new process replaces the two-step TIG flush weld process. After final welding, each finished rod is scanned for fuel enrichment and U-weight content. The pure  $\text{UO}_2$  rods are scanned on an active scanner, which uses a neutron source to slightly irradiate the fuel so that gamma counting can accurately determine the U-weight content and verify the correct enrichment. The completed fuel rod is then ready for assembly into the cage.

## **12.2.4 Structural Component Manufacture and Cage Assembly**

Structural components of the fuel assemblies that reside in the active fuel region are also typically made of stainless steel, while the fuel cladding is a zirconium alloy to limit neutron absorption. Components residing outside the active region are made from stainless steels, such as 316L, or from nickel alloys such as Alloy 718 which have high strength and good corrosion resistance. Not all of the components are made at the fuel fabrication plant;

some components are made by affiliate or third-party metal fabricators and the pieces are sent to the fuel fabrication plant for final assembly of the fuel elements.

The grids are manufactured from sheets of zirconium or nickel alloys depending on their location and function. The sheets are stamped and cut to form strips that are then assembled and carefully welded together to create the final grid shape. Guide tubes are cold-rolled and annealed and are typically sourced from a third party or affiliate. Heavy components such as the end nozzles are cast and then machined. Small components such as guide tube connections and hold-down springs are typically machined. The lower end fitting, guide tubes, and grids are assembled on a dimensionally precise jig table and welded to create a cage for final assembly.

### 12.2.5 Final Fuel Bundle Assembly and Quality Assurance

In the fuel bundle assembly process outlined in Table 12.4, fuel rods are loaded into the cage and the upper end fitting is then attached to fabricate the finished bundle (Figure 12.8). Some manufactures manually push the rods through the grids, whereas others pull them through using specialized tooling. Figure 12.9 shows the final inspection of a completed fuel assembly. The key to this whole process is the strict and detailed adherence to quality assurance practices and standards during each step in the manufacturing process. The mass of materials and their geometric configurations are critical to the proper operation of the nuclear reactor core. Quality assurance procedures and systems built into the process ensure that the correct hardware is accumulated and assembled into the correct locations by bar coded material identification and automated checks for compliance with the engineering design requirements using automated computerized systems. The shop floor control and process history software links the assembly activities directly to the verified engineering data contained in the computer system databases.

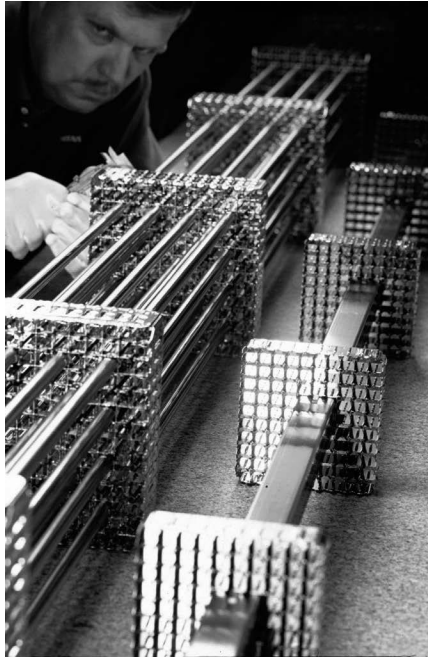
To minimize the potential for an in-reactor fuel rod fretting failure, particular attention is given to preventing the introduction of debris into the bundle during assembly. In addition to a complete bundle helium leak check, final dimensional and visual inspections are performed prior to packaging. Finally, an independent review of all material and processing records for the components of each bundle is performed, concluding with the preparation of the final product quality certification and records package prior to releasing the fuel bundles for shipment. The finished assemblies are packed and held in a storage area until shipped.

**TABLE 12.4**

Process Flowchart for Fuel Bundle Assembly

- 
- Accumulate components and inspection (cleanliness, integrity, and spring envelope)
  - Bundle assembly and inspection (cleanliness, component orientation, upper tie plate perpendicularity, envelope dimensions, and maximum rod gap)
  - Vacuum helium detection leak check
  - Pack and inspect packaging
  - Bundle QA certification and QA release (fuel record search, inspection checklist review, bundle matrix design and as-built assembly comparison, bundle average weight, 48B shipping document preparation, and product quality certification [PQC])
  - Ship
- 

Notes: • Inspection/QA processes,  Manufacturing/assembly processes.



**FIGURE 12.8**  
Assembly of nuclear fuel elements.



**FIGURE 12.9**  
Inspection of a completed fuel assembly.



---

### 12.3 Fuel Facility Licensing

Nuclear fuel cycle facilities are regulated by many state and federal agencies. In the United States, facilities are licensed by the Nuclear Regulatory Commission (NRC), under Title 10 of the Code of Federal Regulations, Part 70 (10 CFR 70), as special nuclear material facilities. In addition, facilities are required to have environmental and nuclear material permits. The NRC licenses fuel cycle facilities under the following federal regulations:

- 10 CFR Part 30—Rules of general applicability to domestic licensing of by-product material
- 10 CFR Part 40—Domestic licensing of source material
- 10 CFR Part 70—Domestic licensing of special nuclear material
- 10 CFR Part 73—Physical protection of plants and materials
- 10 CFR Part 74—Material control and accounting of special nuclear material

The licensing process begins with rigorous evaluation and safety checks to ensure the facilities will be constructed and operated under the regulatory requirements set forth by the NRC in the regulations listed above. Doing this will ensure the safety of the public, the employees, and the environment. Facilities are inspected regularly by NRC inspectors based at NRC headquarters and the NRC regional offices. The number of inspections varies, as does the focus. Key items of interest to the regulators are nuclear criticality control, chemical safety, fire safety, and radiation safety.

License violations are subject to monetary fines based on the significance of the violation. Licenses must also be renewed on a regular schedule. These renewals allow a detailed evaluation of the implementation of the regulatory requirements.

---

### 12.4 Other Fuel Fabrication Facilities

Other types of reactor fuels that are fabricated include heavy water reactor fuels, research reactor fuels, naval fuels, and mixed oxide (MOX) fuels. Heavy water reactor fuels fabrication is much the same as light water fuels with the exception that the assembly configuration is different and the uranium can be of natural enrichment due to the increased moderation provided by the heavy water. Naval fuel fabrication and configuration processes are classified, although it is generally understood that the fuel is metallic and highly enriched, similar to some research reactors.

#### 12.4.1 MOX Fuel

MOX fuel is a next-generation fuel material that is manufactured for use in LWRs using excess plutonium, either from weapons programs or as a recycled product from the reprocessing of LWR fuels. The primary difference in fabrication of MOX fuel from that of  $\text{UO}_2$  is the need for all operations to be carried out in a radiological-controlled hot-cell environment due to the toxicity of the reprocessed fuel. This adds considerable complexity to all processes involved, requiring automation, remote operations, and robotics.

MOX fuel has been used in commercial reactors in France, Japan, Germany, and several other countries for over 40 years. Reprocessing used fuel and recycling the plutonium for power production reduces the total volume of nuclear waste produced by a factor of 5–10 and also reduces the thermal load on final repositories. Much of this reprocessing is performed at the La Hague plant in northern France with a total annual capacity of 1700 MTHM. The primary facilities for fabricating MOX fuel assemblies are the MELOX plant in France with annual capacity of 195 MTHM, and Sellafield in the United Kingdom with annual capacity of 128 MTHM.

The use of weapons programs plutonium aims to decrease the amount of plutonium available for use as nuclear warheads, while recovering useful energy in the process. At the end of the Cold War, the United States and Russia began to cooperate to counter the proliferation of weapons of mass destruction. In 2000, the two countries each arranged to dispose of 34 metric tons of surplus military-grade plutonium by recycling it as fuel for civil nuclear applications. The United States Department of Energy is constructing an MOX Fuel Fabrication Facility in Aiken, South Carolina, with the aim of receiving weapons-grade material for processing by 2019.

#### 12.4.2 Research Reactor Fuels

Research reactor fuels have been manufactured for more than 50 years. In the period up until the 1990s, most research reactors used fuel that was highly enriched (greater than 20% U-235 by weight, sometimes as high as 93%). This fuel was normally uranium metal clad with aluminum. The fuel was in the form of plates that were assembled into elements with seven or more plates in a single element. Due to the unique nature of research reactor fuel and small quantities required, the fuel is fabricated in specialized facilities rather than large-scale industrial facilities. Today, only a few reactors still use highly enriched fuel due to nonproliferation initiatives. Research reactors today use fuel that is enriched to less than 20% <sup>235</sup>U. Many research reactors are pool-type, which typically operate at low temperatures (less than the boiling point of water). Research reactors may produce the equivalent of 1–2 MW of power compared to commercial power reactors which produce on the order of 1000 MW at much higher temperatures.

# 13

## *Spent Fuel Storage*

Kristopher W. Cummings

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### 13.1 Introduction

With the operation of nuclear power plants also comes the creating of and responsibility for storing and eventually disposing of the high-level nuclear waste in the form of spent fuel assemblies. Every 12–18 months the reactor undergoes a refueling outage, where approximately 30%–40% of the fuel assemblies in the core are discharged and replaced with fresh fuel assemblies. The spent fuel discharged from the core during the refueling outage must be safely stored with appropriate cooling, shielding, and criticality safety. The discharged fuel is placed in spent fuel storage racks in the spent fuel pool (SFP) until it is completely filled, and then the spent fuel assemblies are transferred to dry storage casks on the nuclear plant premises (but outside the reactor buildings). This chapter addresses the storage of spent fuel in the SFP and dry storage casks at commercial nuclear reactors.

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### 13.2 Wet Storage

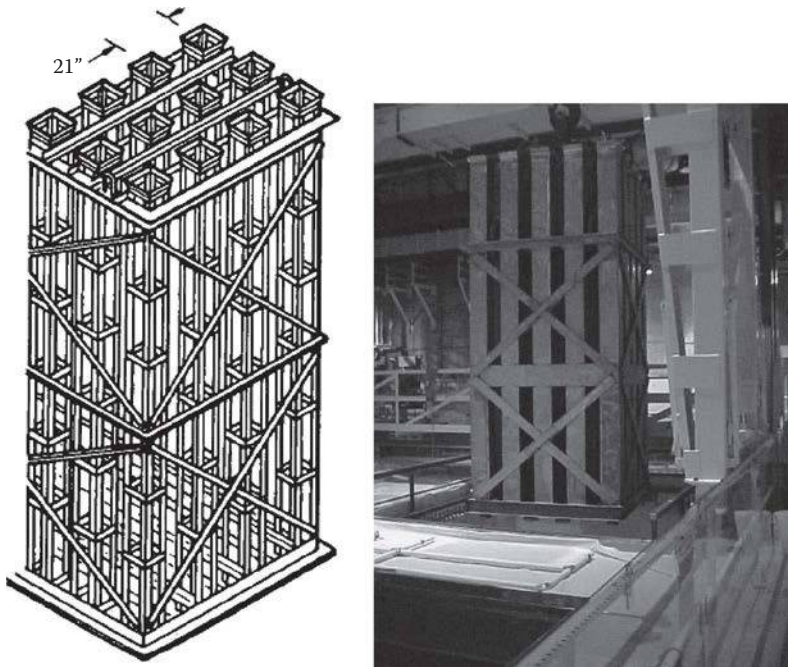
The nuclear power plant was originally designed with an SFP capacity that would hold approximately one full core plus one additional discharge of spent fuel in a widely spaced, open design of spent fuel rack. This design was based on the assumption that spent fuel would be shipped offsite for reprocessing to allow for further discharges from the reactor (NUREG-0575). These original racks contained no neutron absorbers but maintained reactivity control and cooling by spacing the fuel assemblies relatively far apart (21 in.) in a low-density configuration as shown in Figure 13.1 (NUREG/CR-0649,SAND77-1371). The racks were qualified to store fuel of the maximum enrichment used in the core without crediting the reactivity decrease from the destruction of fissile uranium and creation of fission products during operation in the reactor.

With the decision in the mid-1970s to not pursue reprocessing for anti-proliferation reasons, additional spent fuel storage was required. Because the plants continued to discharge spent fuel and the Department of Energy (DOE) did not accept spent fuel in accordance with the Nuclear Waste Policy Act (NWPA), nuclear power plants were required to increase their on-site spent fuel storage by replacing the low-density racks with high-density close-packed racks. Most, if not all nuclear power plants have now undergone re-racking of the SFP at least once, if not several times, to gain additional spent fuel storage capacity.

#### 13.2.1 PWR Spent Fuel Storage Racks

The re-racking of a pressurized water reactor (PWR) SFP includes installation of high-density storage racks and removal of the existing low-density storage racks. Two types of racks must be used in the PWR SFP: flux-trap racks and non-flux-trap racks. The flux-trap racks are qualified for storage of fresh fuel and fuel which has not been fully irradiated in the reactor, whereas the non-flux-trap racks are qualified for storage of fully-irradiated fuel and therefore are used to store the majority of the fuel in the SFP.

The flux-trap racks are composed of stainless steel boxes separated by a gap, with fixed neutron absorber panels centered on each side. The steel walls define the storage cells, and stainless steel sheathing supports the neutron absorber panel and defines the boundary of the flux-trap water gap used to augment reactivity control. Stainless steel channels connect the storage cells in a rigid structure and define the flux-trap between the neutron absorber panels.



**FIGURE 13.1**  
Original low-density spent fuel storage rack design (schematic and photo).

Neutron absorber panels are installed on all exterior walls facing all other racks. Figure 13.2 shows an elevation view of a typical flux-trap type rack used in PWR storage racks.

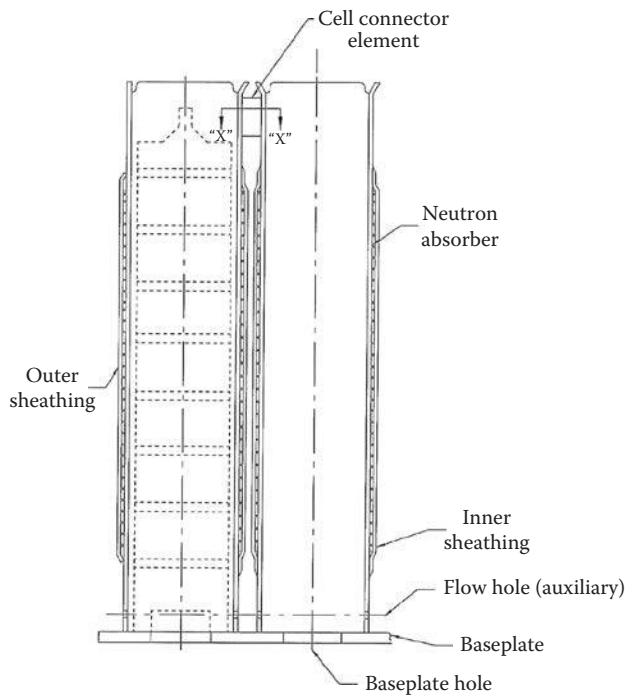
The non-flux-trap racks are composed of stainless steel boxes with a single fixed neutron absorber panel, attached by stainless steel sheathing centered on each side. The stainless steel boxes are arranged in an alternating pattern such that the connection of the box corners form storage cells between the stainless steel boxes. Neutron absorber panels are installed on one side between neighboring non-flux-trap racks.

### 13.2.2 BWR Spent Fuel Storage Racks

Because of the small assembly cross section, and the presence of burnable absorbers ( $Gd_2O_3$ ) in boiling water reactor (BWR) fuel, BWR spent fuel racks can be qualified for maximum enrichment fuel in the close packed array racks with no flux traps. The BWR racks are composed of stainless steel boxes with a single fixed neutron absorber panel, attached by stainless steel sheathing centered on each side. The stainless steel boxes are arranged in an alternating pattern such that the connection of the box corners form storage cells between the stainless steel boxes. Neutron absorber panels are installed on one side between neighboring non-flux-trap racks.

### 13.2.3 Criticality Safety Considerations

The high-density spent fuel storage racks (PWR or BWR) are analyzed in accordance with the applicable codes and standards listed in ANSI standards (ANS-8.17-1984), the *Code of Federal Regulations* (CFR, Title 10, Part 50), Kopp (1998), and US Nuclear Regulatory Control (USNRC) guidance documents (NUREG-0800, 1978, 1981, ISG-DSS-2010-001).



**FIGURE 13.2**  
Elevation view of a PWR flux-trap style spent fuel storage rack.

If no credit for soluble boron in the SFP water is taken, the effective neutron multiplication factor ( $k_{\text{eff}}$ ) of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95% probability, 95% confidence level, if flooded with unborated water. If credit is taken for soluble boron in the SFP water, then the  $k_{\text{eff}}$  of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95% probability, 95% confidence level, if flooded with borated water. The  $k_{\text{eff}}$  must also remain below 1.0 (i.e., subcritical) at a 95% probability, 95% confidence level under the assumed loss of soluble boron in the pool water, that is, assuming unborated water in the SFP (CFR, Title 10, Part 50). Finally, reactivity effects of abnormal and accident conditions are also evaluated to assure that under these conditions the reactivity will be maintained less than 0.95.

The maximum  $k_{\text{eff}}$  is determined from the computer code-calculated  $k_{\text{eff}}$ , the calculational bias, and the applicable uncertainties (bias uncertainty, calculational uncertainty, fuel and rack manufacturing tolerances, depletion uncertainty, burnup uncertainty) using the following formula:

$$\text{Max } k_{\text{eff}} = \text{Calculated } k_{\text{eff}} + \text{biases} + [\sum_i (\text{Uncertainty}_i)^2]^{1/2}$$

In the geometric models used for the calculations, each fuel rod and its cladding are described explicitly, and reflecting or periodic boundary conditions are used in the radial direction which has the effect of creating an infinite radial array of storage cells. Only the active fuel height of the fuel assembly is modeled (the top and bottom nozzle and other hardware ignored) with a water reflector on each end.

### 13.2.3.1 PWR Racks

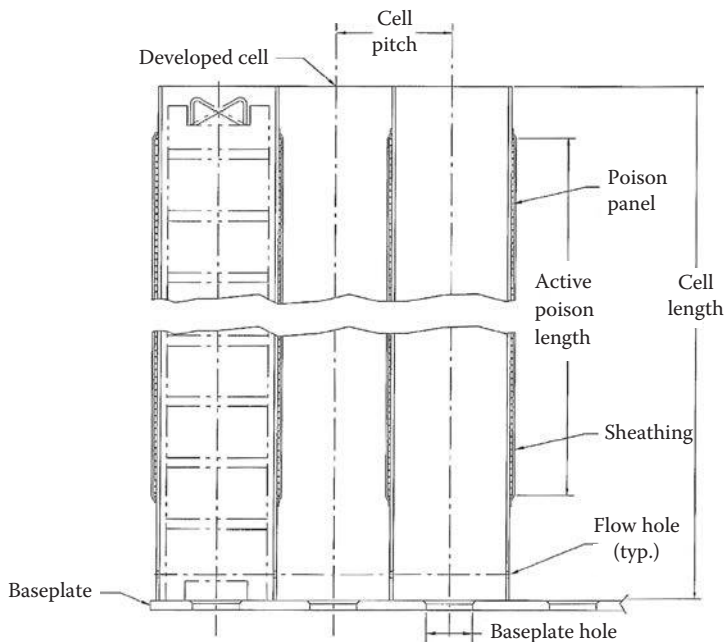
The PWR flux-trap style racks are designed to accommodate fresh or spent fuel assemblies with a maximum initial enrichment of 5.0 wt%. No burnup requirement is dictated for the flux-trap racks. These racks are present in the SFP to accommodate new fuel to be loaded into the reactor during the refueling outage or to accommodate the entire core in the event of a shutdown and subsequent defueling of the entire core.

The PWR non-flux-trap style racks, as shown in Figure 13.3, are designed to accommodate fuel assemblies with a maximum initial enrichment of 5.0 wt% which have accumulated a given minimum burnup or fuel of initial enrichment and burnup combinations within the acceptable domain. A representative example of the burnup versus enrichment curve for a typical PWR non-flux-trap rack is provided in Figure 13.4.

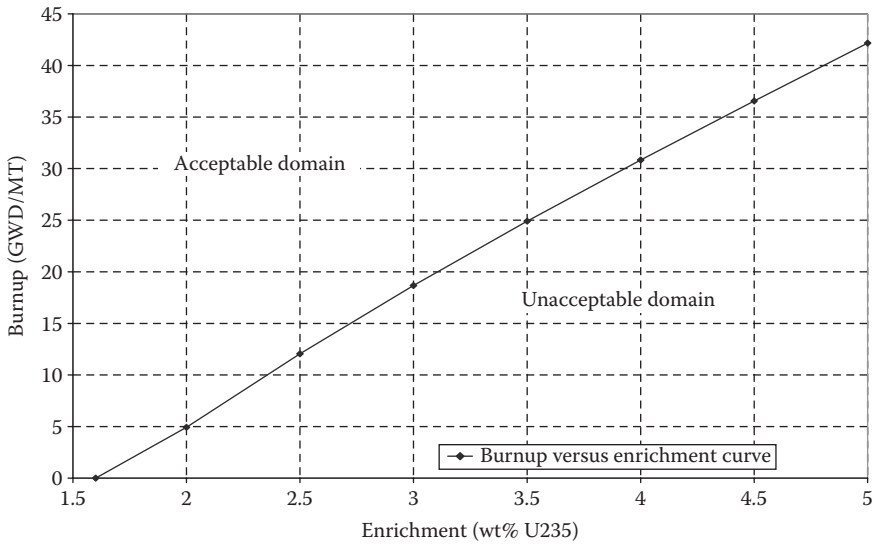
Additionally, accident and abnormal conditions are considered such as misplacement of a fuel assembly outside the rack, misloading of a fresh, unburned assembly into the center of the non-flux trap racks, variation of water temperature, drop of an assembly onto the top of the rack, and lateral rack movement during a seismic event. For accident conditions, soluble boron in the SFP may be credited to ensure that the reactivity does not exceed 0.95.

### 13.2.3.2 BWR Racks

In BWR fuel, there are a wide variety of fuel assembly designs, including distributed fuel enrichments and gadolinium loadings, varying in the axial and radial directions. It is conventional practice to describe the fuel assembly in terms of the  $k_{inf}$  in the standard cold core geometry (SCCG) for each planar (axial) region of unique enrichment or geometric design. Frequently, the anticipated  $k_{inf}$  in the SCCG is provided by the fuel vendor but may be independently calculated for various void contents in the core.

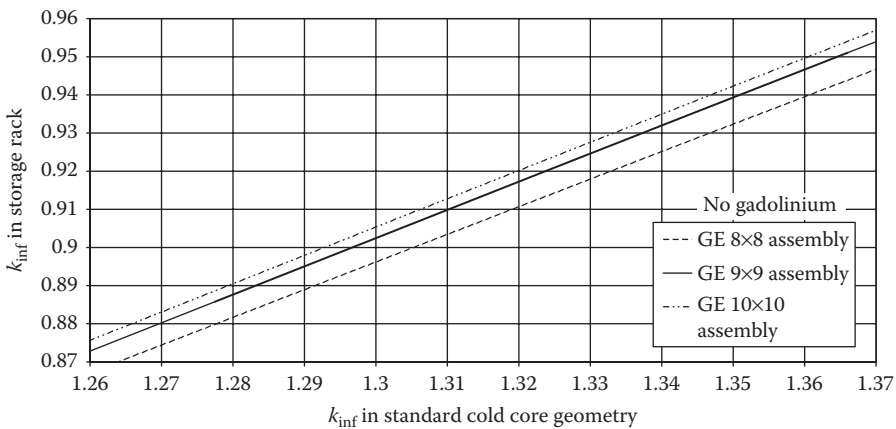


**FIGURE 13.3**  
Elevation view of a PWR non-flux-trap spent fuel storage rack.



**FIGURE 13.4**  
Typical burnup versus enrichment curve for a non-flux trap PWR storage rack.

The general methodology is to perform in-core depletion calculations, then to restart the calculations with the assemblies in the SCCG and in the storage rack geometry of the specified design. Based on these calculations, the  $k_{inf}$  in the storage rack configuration is correlated to the  $k_{inf}$  in the SCCG over the entire range of possible burnups for each assembly in the SFP. Figure 13.5 illustrates this correlation for several types of General Electric assemblies in a specific rack design. From this correlation, the maximum allowed  $k_{inf}$  in the SCCG may be determined to ensure that the  $k_{eff}^*$  in the storage rack remains  $\leq 0.95$ . The established SCCG  $k_{inf}$  limit, determined from the above analysis, is compared with



**FIGURE 13.5**  
Correlation of the  $k_{inf}$  in the SCCG to the  $k_{inf}$  in the BWR fuel storage rack.

\* Reactivity allowances for manufacturing tolerances, uncertainty in depletion calculations, and other correction factors are added to the rack  $k_{inf}$  to determine  $k_{eff}$ .



the peak  $k_{inf}$  in the SCCG, as calculated or as provided by the fuel vendor to ensure the acceptability of a specific fuel design to be safely placed in the storage rack.

#### 13.2.4 Structural Considerations

The spent fuel storage racks must be able to withstand all loadings under normal, seismic, and accident conditions. The structural integrity of the racks must be shown to comply with the USNRC Standard Review Plan (USNRC, NUREG-0800) and the OT Position Paper (USNRC 1978).

For closely spaced racks, demonstration of kinematic stability under seismic conditions is verified by including all rack modules in one comprehensive simulation using a whole pool multi-rack (WPMR) 3D analysis. During dynamic rack motion, hydrodynamic energy from the water surrounding the racks is withdrawn from or added to the moving rack. In the WPMR analysis, all rack modules are modeled simultaneously and coupling effect due to the multibody motion addressed. Additionally, more conservative single rack analyses are performed to confirm rack stability under the most conservative conditions, such as fuel loading eccentricities or interim re-racking configurations.

The WPMR method allows for appropriate determination of stress and displacement criteria through the following steps:

1. Preparation of 3D dynamic models suitable for a time history analysis of the racks which includes the assemblage of all racks in the SFP, all fluid coupling interactions, and mechanical couplings appropriate for a nonlinear simulation.
2. 3D dynamic simulations of various physical conditions such as coefficient of friction and extent of cells containing fuel assemblies. Resultant displacements and stresses are determined.
3. Stress analyses of high stress areas for limiting conditions are completed to demonstrate compliance with ASME Code Section III, Subsection NF limits on stress and displacement.

The SFP itself must be capable of holding the significant weight associated with the racks, spent fuel, and water under normal and accident conditions. The key SFP structural members include four reinforced concrete walls and a base slab that are lined with a stainless steel liner. The pool walls are supported by reinforced concrete columns and a mass foundation. The SFP may also be supported by a system of caissons that extend below the pool slab through the supporting soil and into the underlying bedrock. Floor slabs adjacent to the SFP walls provide horizontal bracing.

The presence of high-density racks, compared with the original low-density open-frame racks originally considered for the SFP, will increase mechanical and thermal loads on the SFP structure (i.e., walls and floor). The loads on the pool slab are increased due to the presence of a greater number of racks and fuel. Pool structural loading includes the following discrete load components defined in accordance with the applicable governing document (USNRC, NUREG-0800):

1. Dead loads
  - a. Weight of concrete and steel liner
  - b. Weight of racks and fuel assemblies stored in rack modules
  - c. Hydrostatic pressure from the presence of water in the SFP

2. Live loads
  - a. Live loads from the floor slab
3. Seismic-induced loads
  - a. Self-inertia loads
  - b. Rack pedestal seismic loads
  - c. Hydrodynamic inertia and sloshing loads
  - d. Hydrodynamic coupling pressures
4. Thermal loading
  - a. Normal operating temperatures
  - b. Accident condition temperatures

The evaluation of the pool structural integrity considers critical load combinations specified by NUREG 0800 Section 3.8.4 (NUREG 2007), which are similar in intent and bounding to the load combinations specified in ACI-349 in 1981, with additional guidance from Reg. Guide 1.142.

### 13.2.5 Thermal–Hydraulics Considerations

One of the primary purposes of the water in the SFP is to ensure that the freshly discharged fuel assemblies, which continue to emit a significant amount of heat, are properly cooled. The thermal hydraulic considerations for the SFP racks and fuel assemblies may be broken down into the following categories:

- The maximum bulk temperatures for the design basis core offload scenarios must not be exceeded. The bulk temperature limit is typically in the range 160°F–170°F.
- The local water temperature must be limited to ensure that localized boiling does not occur within the spent fuel storage rack while forced cooling is operating.
- The cladding temperature of the fuel must be limited to ensure that nucleate boiling is not possible while forced cooling is operating.
- During a loss-of-forced cooling scenario, corrective actions must be implemented within a minimum timeframe and makeup water must be available.

The SFP cooling system (SFPCS) cools the fuel in the SFP by transferring decay heat through a heat exchanger to the component cooling water (CCW) system. The SFPCS typically has two cooling loops, each with a single pump and heat exchanger. Each cooling loop has an SFP water flow rate of at least 1800 gallons per minute. Normal makeup water for the SFP is obtained from the demineralized water storage tank. Emergency makeup water is obtained from the refueling water storage tank (RWST) or the reactor makeup water storage tank (RMWST).

#### 13.2.5.1 Offload/Cooling Scenarios

To ensure that the spent fuel in the pool can be adequately cooled with the existing SFPCS, the following scenarios are postulated to occur and analyzed accordingly:

- A partial core offload into the SFP which becomes filled to capacity with the partial core offload. This scenario covers a standard refueling outage where only the assemblies to be discharged are transferred to the SFP.
- A full core offload into the SFP which becomes filled to capacity with the full core offload. This scenario covers a refueling outage where the entire core is transferred to the SFP.
- An abnormal full core offload (full core plus a partial core offload) into the SFP into the spent fuel which becomes filled to capacity with the abnormal full core offload. This scenario covers a shutdown of the reactor and transfer of the entire core soon after a previous refueling outage.

These scenarios cover any conceivable condition that the SFPCS would be expected to handle during the lifetime of the reactor.

### 13.2.6 Radiological Considerations

In addition to providing cooling of the discharged fuel assemblies, the SFP water also provides for radiological protection (shielding) of the highly radioactive fuel assemblies. More than 25 ft of water cover the fuel assemblies stored in the spent fuel racks, providing for only negligible doses (<1.0 mrem/h) at the surface of the SFP water. An assembly in transit above the racks during fuel movement during refueling or fuel shuffling is restricted by the fuel gantry crane to have no less than approximately 9 ft of water between the top of the assembly and the surface of the SFP water. Under transit conditions, a freshly discharged fuel assembly would result in a dose of <2 mrem/h at the pool water surface.

The SFP building is also equipped with high-efficiency particulate air (HEPA) and charcoal filters to collect and filter radioactive gases that might be released from a spent fuel assembly in the event of an accident. An accident of a dropped assembly is postulated to occur that causes a breach of all fuel rods in a single assembly and release of the radioactive gases trapped in the fuel rod. Under such conditions, the dose to an individual at the exclusion area boundary (EAB) would be much less than the acceptance criteria specified in Section 15.7.4 of the Standard Review Plan (NUREG 1981).

### 13.2.7 Material Considerations

Safe storage of nuclear fuel in the SFP requires that the materials utilized in the storage rack fabrication be of proven durability and be compatible with the pool water environment. This section provides a synopsis of the materials used in spent fuel storage racks.

#### 13.2.7.1 Structural Materials

The structural materials of the spent fuel storage racks must be able to withstand the temperatures and pressures present in the SFP. Additionally, in PWR SFPs, where boric acid (soluble boron) is used to augment criticality safety, the acidic environment must be considered. Therefore, the primary structural material in use in the majority of spent fuel racks is austenitic stainless steel.

### 13.2.7.2 Neutron Absorbing Materials

A variety of materials have been used as fixed neutron absorber materials in SFP racks. Boraflex was one of the first materials used in high-density storage racks. Boraflex is a composite material, similar to rubber, composed of  $B_4C$  particles trapped in a silicone elastomer matrix. Boraflex's main advantages were its cost and ease of installation. Unfortunately, time has shown Boraflex to be an unviable option. Boraflex degrades under exposure to radiation, causing embrittlement, shrinkage, cracking, and eventual washout into the fuel pool water.

The material in widest use is Boral<sup>®</sup>, a hot pressed composite of  $B_4C$  and aluminum powder sandwiched between thin 0.01-inch-thick aluminum plates. Boral has been shown to be dependable in nuclear applications with no noticeable degradation. In some instances, Boral has shown a tendency to exhibit blistering near the edges of the material where ingress of water causes generation of hydrogen.

The latest generation of neutron absorber material in use is Metamic and Boralcan. These newer materials are a combination of  $B_4C$  and aluminum mixed together to produce uniformly homogeneous materials. These materials offer the advantage of no aluminum cladding which eliminates blistering.

An alternate which has not gained wide acceptance in the United States is borated steel. Borated steel has a lower concentration of boron for a given thickness compared to Boral, therefore the spacing between assemblies must be larger to achieve the same results.

### 13.2.8 Summary

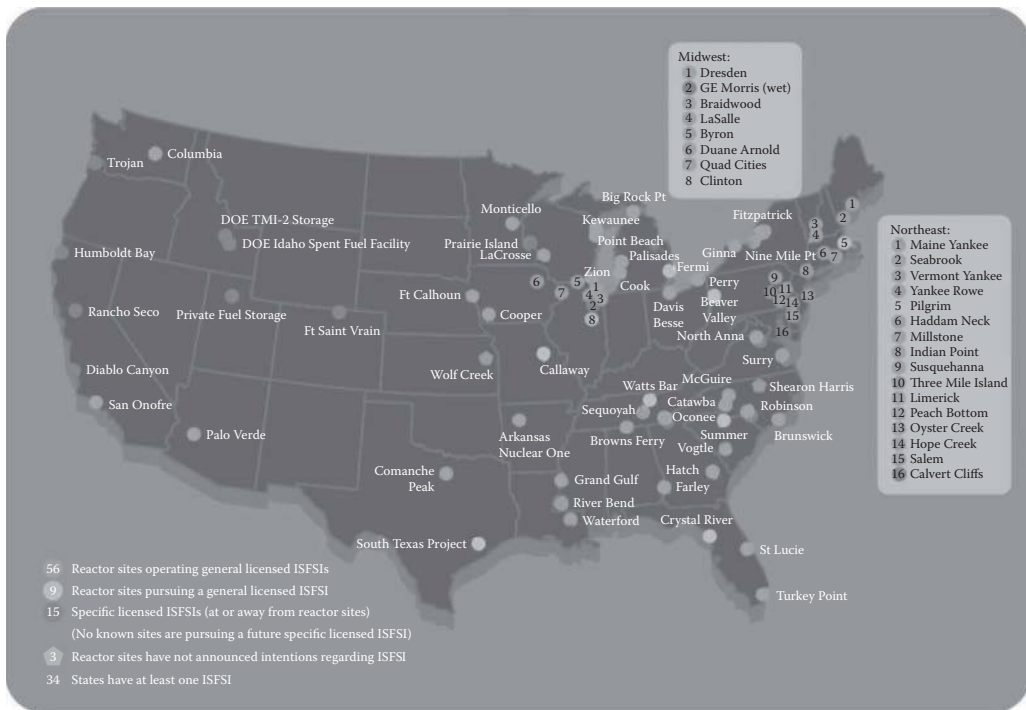
Most nuclear plants in the United States have now been re-racked at least once, with the entire pool filled with high-density storage racks. For most nuclear plants, there is no longer room to place additional racks in the pool, and the density of the storage racks is at a maximum. Therefore, the only remaining option available to nuclear plants is to remove the spent fuel from the SFP and store the fuel assemblies in dry storage casks, which is discussed in the next section.

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## 13.3 Dry Cask Storage

After high-density re-racking of the pools, the capacity of the SFP is at or near its maximum capacity. Once all of the rack storage locations have been filled with discharged fuel, the utility is required to remove fuel from the SFP to continue operating the plant. In one instance, this has involved shipping spent fuel assemblies from one plant that is nearing SFP capacity to another plant that has sufficient room to accommodate those assemblies. However, in most cases this is neither a viable nor cost-effective option. The majority of nuclear plants in the United States have opted to implement an independent spent fuel storage installation (ISFSI) or more commonly referred to as dry storage.

Dry storage of spent fuel assemblies began in 1986 at the Surry/North Anna site, being the first established ISFSI in the United States. As it became apparent that additional nuclear plants would also need dry storage, regulations for dry storage were codified in the Code of Federal Regulations under 10CFR72 (CFR, Title 10, Part 72). 10CFR72 sets down the regulatory requirements for dry storage casks and applicable regulatory acceptance



**FIGURE 13.6**  
Locations and types of independent spent fuel storage installations (ISFSIs).

criteria. These regulations are further supported by various NRC Guidance, most importantly NUREG-1536 “Standard Review Plan for Dry Cask Storage Systems” and additional Interim Staff Guidances (ISGs) distributed by the Spent Fuel Storage and Transportation (SFST) Division of the NRC.

Dry cask storage has currently been implemented at approximately 65 nuclear sites in the United States as shown in Figure 13.6, with over 2000 casks loaded and implemented as of the beginning of 2015 with an additional 150–200 casks expected to be loaded each year. Spent fuel storage is now implemented or in the process of being implemented at almost every nuclear plant in the United States. The decision to develop an ISFSI on the plant premises begins by deciding whether to develop a site-specific license or implement a previously approved dry storage cask under a general license.

### 13.3.1 General License versus Site-Specific License

NRC authorizes storage of spent nuclear fuel at an ISFSI under two licensing options: site-specific licensing and general licensing.

Under a site-specific license, an applicant submits a license application to NRC and the NRC performs a technical review of all the safety aspects of the proposed ISFSI. If the application is approved, the NRC issues a license that is valid for 40 years, which can be further renewed for additional time periods. A spent fuel storage license contains technical requirements and operating conditions (fuel specifications, cask leak testing, surveillance, and other requirements) for the ISFSI and specifies what the licensee is authorized to store at the site.

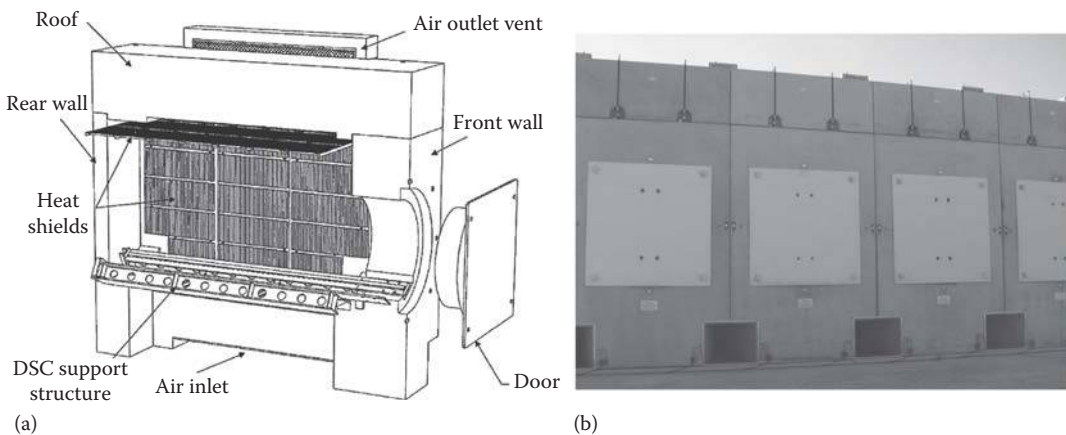
A general license authorizes a nuclear power plant licensee to store spent fuel in NRC-approved casks listed in 10CFR 72.214 (CFR, Title 10, Part 72) at a site that is licensed to operate a power reactor under 10 CFR Part 50. Licensees are required to perform evaluations of their site to demonstrate that the site is adequate for storing spent fuel in dry casks. These evaluations must show that the cask Certificate of Compliance conditions and technical specifications can be met, including analysis of earthquake intensity and tornado missiles. The licensee must also review their security program, emergency plan, quality assurance program, training program, and radiation protection program and make necessary changes to incorporate the ISFSI at its reactor site.

### 13.3.2 Types of Dry Storage Casks

While there are currently 17 approved spent fuel storage casks listed in 10CFR72.214 (CFR, Title 10, Part 72), the majority of commercial spent fuel is stored in concrete storage casks or modules in a vertical or horizontal configuration, respectively. For the vertical configuration, the fuel may be stored in either an aboveground storage cask or can be placed into a storage location constructed in the ground. Figure 13.7a and b shows a schematic and photograph of a horizontal dry storage module; Figure 13.8a and b shows a schematic and photograph of an aboveground vertical storage cask; and Figure 13.9a and b shows a schematic of an underground vertical storage cask.

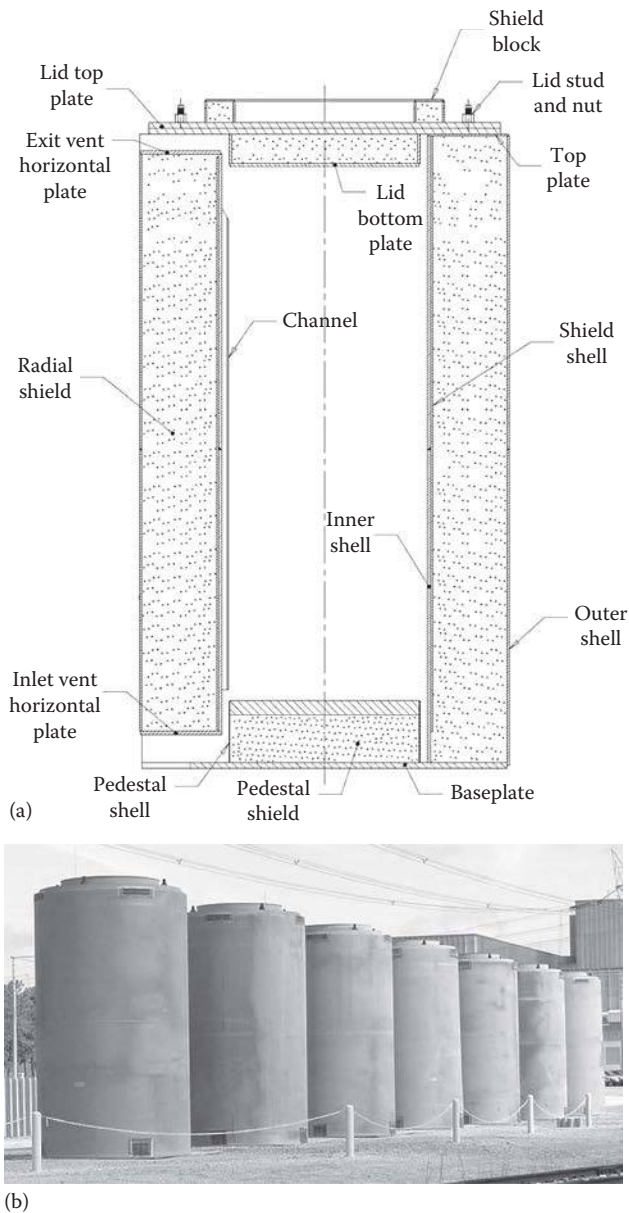
An aboveground vertical cask consists of a cylindrical storage overpack with the internal canister stored in an upright position. The storage overpack is constructed from steel and concrete and provides for physical protection of the internal canister and massive shielding to reduce the emitted radiation. Inlet and outlet vents are provided at the top and bottom of the storage overpack to facilitate removal of the residual spent fuel decay heat through natural convective cooling.

An underground vertical cask consists of a subterranean cylindrical cavity with the internal canister stored vertically, such that the top of the canister is at or near the surface. A foundation pad is constructed below the cylindrical cavities to support the loaded canisters, with an additional concrete pad at the top of the canister enclosure that provides



**FIGURE 13.7**

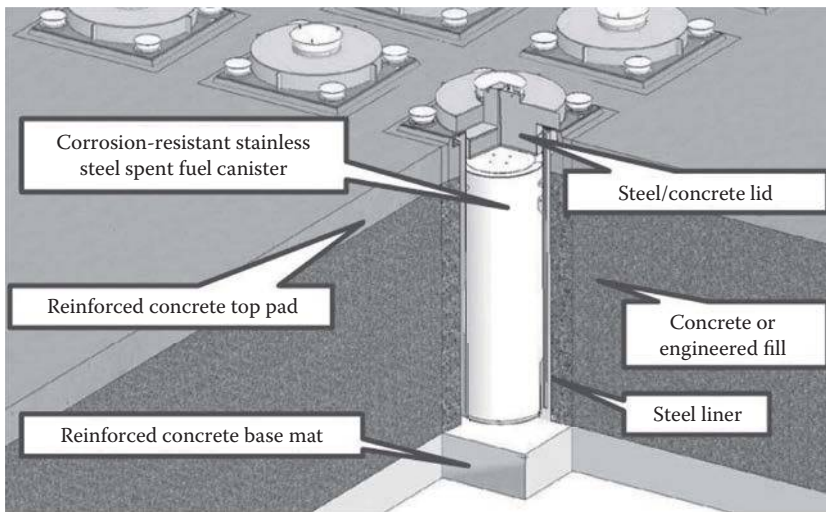
(a) Schematic of horizontal spent fuel storage cask. (From Final safety analysis report for the NUHOMS HD horizontal modular storage system for irradiated nuclear fuel, Revision 0, USNRC Docket No. 72-1030.)  
 (b) Photograph of horizontal spent fuel storage cask.

**FIGURE 13.8**

(a) Schematic of vertical spent fuel storage cask. (From Final safety analysis report for the HI-STORM 100, Revision 6, USNRC Docket No. 72-1014.) (b) Photograph of vertical spent fuel storage cask.

support for the loaded cask transporter. The space between the two support pads is filled with engineered fill and provides for shielding and physical protection of the stored canisters. A concrete and steel lid rests on top of the upper support pad and provides additional physical protection, shielding, and ventilation to the stored canister.

In the horizontal configuration, the canister is stored within a horizontal storage module, which consists of reinforced concrete walls, a steel support structure to hold the canister



(a)



(b)

**FIGURE 13.9**

(a) Schematic of underground spent fuel storage cask. (b) Photograph of underground spent fuel storage cask.

in a horizontal configuration, and a concrete door. The horizontal storage module serves the same purpose as the vertical overpack, to protect the spent fuel, reduce radiation, and provide for decay heat removal.

### 13.3.3 Types of Spent Fuel Storage Canisters

In the vertical and horizontal storage configurations, an internal canister stores the spent fuel assemblies in a safe subcritical configuration. The internal canister is a welded stainless steel pressure vessel that provides for criticality control, an inert helium environment, structural support of the fuel assemblies, and confinement of the highly radioactive



content contained therein. The canister contains a structural steel basket of varying geometries to store BWR or PWR spent fuel assemblies with additional fixed neutron absorbers composed of a metal matrix of aluminum and boron carbide.\* BWR canisters provide for storage of between 52 and 87 BWR fuel assemblies, whereas PWR canisters provide for storage of between 24 and 37 PWR assemblies. Figure 13.10 provides cross-sectional diagrams of several currently licensed canisters used for dry storage.

#### 13.3.4 Structural Considerations

A dry storage system must be designed to be sufficiently robust to be able to withstand normal and off-normal loads and the worst-case impact loads from accident events associated with handling, storage, and natural environmental phenomena. The key structural functions of the canister are to ensure that the canister basket does not deform such that the stored fuel remains in a subcritical geometric configuration and to provide for confinement of the enclosed radionuclides. The key structural functions of the storage overpack or horizontal module is to serve as a tornado-missile barrier to the enclosed canister, maintain air-cooling flow paths to the enclosed canister, ensure stability in the storage configuration over long periods of time, maintain the position of the radiation shielding and additionally for vertical overpacks, allow movement of the overpack with a loaded canister and withstand a nonmechanistic tip-over of the storage overpack.

The stress limits for the canister vessel are in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB (ASME, Subsection NB). The stress limits for the canister basket are in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG (ASME, Subsection NG). In both cases, stresses are identified by the type of stress, that is, bending, shear, and so on, and the classification of the stress, that is, primary, secondary, and so on.

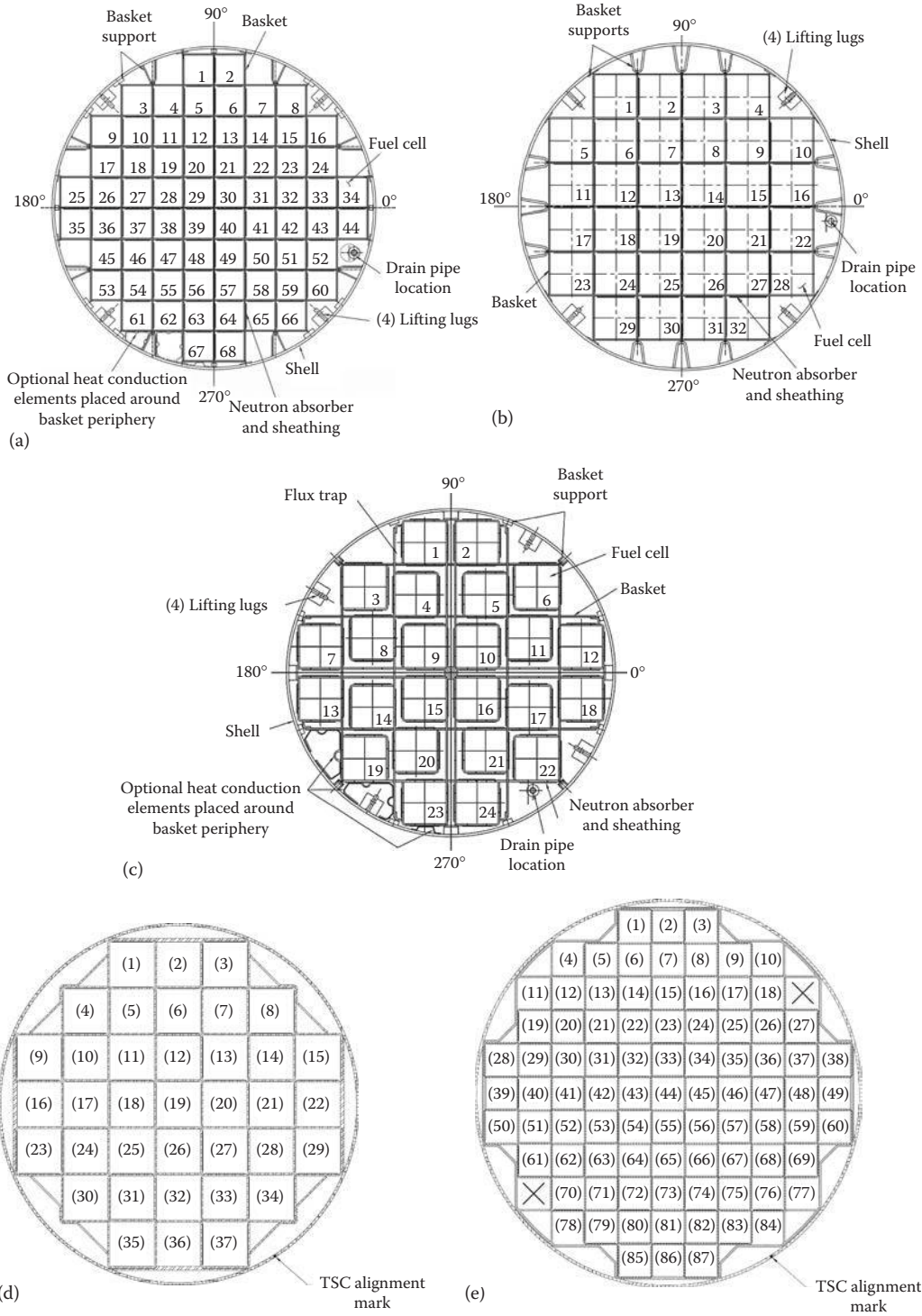
Specifically, the following event loads must be considered to ensure the safety of the storage system is maintained:

- Internal and external pressures
- Increased temperature
- Cask handling accident (horizontally oriented transfer cask or vertically oriented storage overpack)
- Flood
- Fire
- Tornado wind and missiles (i.e., automobile, steel pipe, telephone pole)
- Explosion
- Earthquake

Additionally, because the internal canister is expected to be used for transportation to a centralized interim storage facility or final repository for disposal, the canister is designed to withstand the various transportation scenarios postulated in 10 CFR 71.

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\* An alternative basket type has been developed and licensed which is constructed out of Metamic-HT, a metal matrix composite made by embedding nanoparticles of aluminum oxide and fine boron carbide powder on the grain boundaries of aluminum resulting in improved structural strength properties at elevated temperatures. This allows the basket to serve as both a structural component and a neutron absorber.



**FIGURE 13.10** (a–e) Cross-sectional view of various spent fuel storage canisters. (From Final safety analysis report for the HI-STORM 100, Revision 6, USNRC Docket No. 72–1014.)

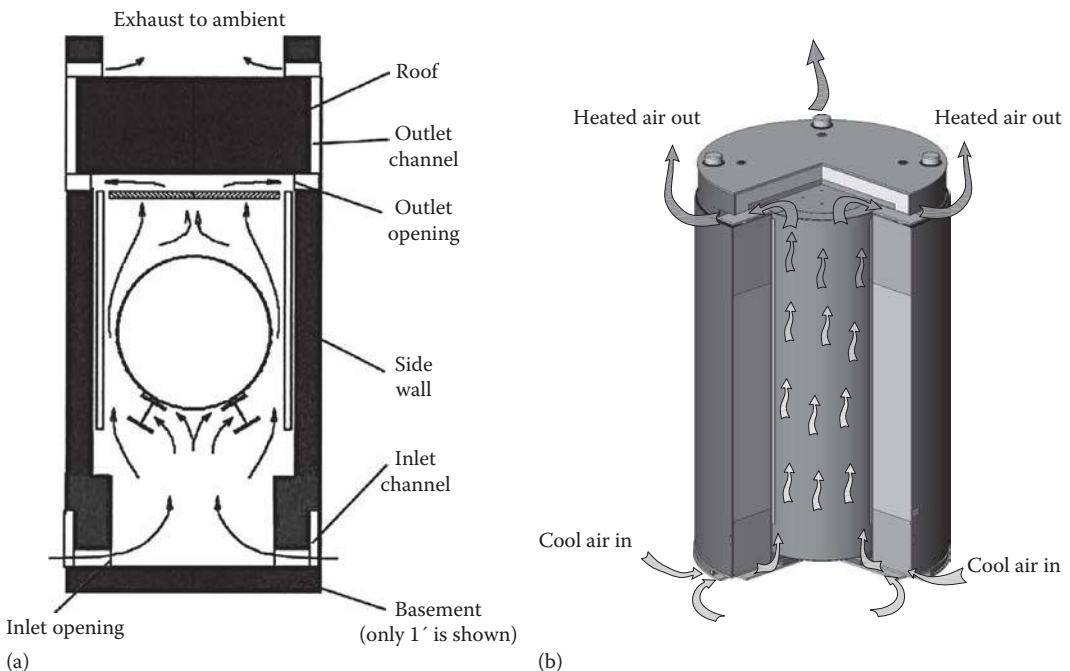
### 13.3.5 Thermal Considerations

All dry storage systems are designed to passively reject decay heat from the stored spent fuel over long periods of time. This heat rejection is primarily achieved through cool air entering the vents near the bottom of the dry storage system, traveling through the storage system cavity, past the internal canister, and finally exiting as hot air near the top of the overpack, as shown in Figure 13.11a and b. The safe storage of commercial spent nuclear fuel is ensured by demonstrating that the following acceptance criteria are met:

- The fuel cladding temperature is limited to 400°C for long-term storage and short-term operations.
- The fuel cladding temperature is limited to 570°C for accident and off-normal events.
- The maximum internal pressure remains below its design limitations.
- The cask and fuel materials are maintained within the minimum and maximum specified temperatures.

To ensure that these criteria are met, the total amount of decay heat stored in the system must be limited. Dry storage systems currently available allow as much as 45 kW of decay heat to be stored. The canister cavity is backfilled with helium gas to ensure an inert environment and to further facilitate heat transfer.

One primary difference between vertical and horizontal systems is in the primary method of heat transfer. For horizontal systems, the primary method of heat transfer is conduction



**FIGURE 13.11**

(a) Passive air cooling of horizontal dry storage systems. (From Final safety analysis report docket No. 72-1030.)

(b) Passive air cooling of vertical dry storage systems. (From Final safety analysis report docket No. 72-1014.)

and radiation of the heat through the materials of the basket to the canister walls. Therefore, high-heat conduction materials such as aluminum must be present within or surrounding the basket to facilitate this process. Vertical systems, however, rely on heat convection for heat transfer. An increased helium backfill pressure, in conjunction with low-pressure drop circulation passages in the canister basket design, induces a thermosiphon upflow through the basket structure. The decay heat absorbed by the helium during upflow is rejected to the canister shell during downflow of the helium in the peripheral downcomers.

### 13.3.6 Shielding Considerations

The majority of the shielding capability in a dry storage system comes from the surrounding overpack. For the case of underground systems, the ISFSI pad and the surrounding backfill provides additional shielding capabilities. NUREG-1536 (USNRC, 1997) outlines the dose rate acceptance criteria that must be met for a dry storage system at an ISFSI.

- The minimum distance from the spent fuel handling and storage facility to the controlled area boundary must be at least 100 m. The controlled area is the immediately surrounding area for which the licensee, that is, nuclear power plant, exercises authority.
- The radiation shielding of the dry cask storage system must be sufficient to ensure that the radiation dose requirements of 10 CFR 72.104(a) (CFR, Title 10, Part 72) are met at the controlled area boundary during normal and off-normal operations. This dose rate limit is 25 mrem/year to the whole body.
- The radiation shielding of the dry cask storage system must also limit the dose to an individual at or beyond the controlled area boundary to less than 5 rem to the whole body under any accident conditions.
- Dose rates must be consistent in “as low as reasonably achievable” (ALARA) program.

The principal source of radiation in the dry storage system is gamma rays and neutrons. The gamma source is the result of decay of radioactive fission products, secondary photons from neutron capture, and activation of fuel assembly components during in-core irradiation. The neutron source originates from spontaneous fission, alpha-neutron reactions in the fuel, secondary neutrons from subcritical fissions, and gamma-neutron reactions.

Gamma radiation shielding is provided by the steel structure of the canister and the steel and concrete of the storage overpack. Neutron radiation shielding is provided by the concrete in the storage overpack. To ensure that the acceptance criteria are met, radiation shielding analyses are performed to determine the allowable burnup and cooling time combinations that may be stored in the cask. Additional radiation dose reduction can be achieved by using a regionalized loading scheme, that is, placing high-radiation-emitting fuel assemblies in the center of the canister basket and low-radiation-emitting assemblies on the periphery of the basket.

### 13.3.7 Criticality Considerations

The criticality safety analysis must show that the effective multiplication factor ( $k_{eff}$ ) does not exceed 0.95, including all biases and uncertainties, evaluated with a 95% probability at the 95% confidence level, under all credible normal, off-normal, and accident events. At least two unlikely and independent changes or accidents must occur to modify the conditions that ensure criticality safety before a nuclear criticality accident is possible.

Criticality safety of a dry storage system is ensured by the following design parameters:

- The geometry of the fuel basket within the canister
- The use of fixed neutron absorbing materials within the basket
- A limit on the maximum enrichment of the fuel assemblies
- A limit on the minimum soluble boron concentration in the SFP water for loading/unloading operations where moderator (water) is in the canister

Additionally, criticality safety of the dry storage system includes the following conservative assumptions:

- All fuel is assumed to be fresh, unirradiated fuel; no credit for burnup of the fuel, such as destruction of  $^{235}\text{U}$  or build-up of neutron absorbing fission products.
- No credit for fuel-related burnable absorbers, such as Gadolinia in BWR fuel and integral fuel burnable absorber (IFBA) rods or Erbium in PWR fuel.
- Reduced credit (between 75% and 90%) for the  $^{10}\text{B}$  content in the fixed neutron absorbers.
- Internal moderator is assumed to be at the temperature and density that corresponds to the highest reactivity.

An essential part of the criticality safety analysis is to ensure that the computer code accurately predicts the effective multiplication factor. Therefore, the computer code is benchmarked against experimental data, using critical experiments that encompass the pertinent design parameters of the canister basket. The most important parameters are (1) the enrichment, (2) the geometrical spacing between fuel assemblies, (3) the boron loading of the fixed neutron absorbing panels, and (4) the soluble boron concentration in the water.

### 13.3.8 Confinement Considerations

The primary confinement boundary against the release of radionuclides is the internal canister. The confinement boundary consists of the canister shell, baseplate, lid, and all associated welds. The canister is designed to maintain its confinement function under all normal, off-normal, and accident conditions.

Current canisters are designed, welded, and tested to criteria specified by the USNRC such that there is no credible design basis scenario by which leakage of radioactive materials can occur. The criteria include the following:

- The canister is constructed from austenitic stainless steel.
- The canister welds are liquid penetrant and ultrasonically examined to ensure the weld is satisfactory.
- The canister is shown by structural analysis to maintain the confinement barrier even under a 45-g deceleration associated with a drop or tip-over.
- The canister materials and welds are performed under an NRC approved quality assurance program.
- The canister temperatures and pressures shown by thermal hydraulic analysis are demonstrated to be maintained within their design basis limits.

### 13.3.9 Loading Sequence

Specific steps are performed to prepare a dry storage system for fuel loading, to load the fuel, to prepare the system for storage, and to place it in storage at an ISFSI. For above-ground vertical systems, the canister transfer from the transfer cask to the vertical overpack may be performed in the cask receiving area, at the ISFSI, or any other location deemed appropriate by the utility. For horizontal systems and belowground vertical systems, the canister transfer from the transfer cask to the horizontal module or canister enclosure, respectively, must occur at the ISFSI, where the module/canister enclosure is located. The transfer cask and vertical overpacks may be transferred between the ISFSI and the fuel loading facility using a specially designed transporter, heavy haul transfer trailer, or any other load handling equipment designed for such applications.

The empty canister is raised and inserted into the transfer cask. The annulus is filled with plant water and the canister is filled with SFP water or plant demineralized water. An inflatable seal is installed in the upper end of the annulus between the canister and the transfer cask to prevent SFP water from contaminating the exterior surface of the canister. The transfer cask with the empty canister is then lowered into the SFP for fuel loading using the overhead crane. Pre-selected assemblies are loaded into the canister and a visual verification of the assembly identification is performed.

While still underwater, the canister lid is installed into the top of the canister. The transfer cask and loaded canister are raised to the SFP surface. When radiation dose rate measurements confirm that it is safe to remove the transfer cask from the SFP, the cask is removed from the SFP. The transfer cask is sprayed with demineralized water to help remove contamination as it is removed from the SFP.

The transfer cask is placed in the designated preparation area. The top surfaces of the canister lid and the upper flange of transfer cask are decontaminated. Temporary shielding, if utilized, is installed. Temporary shielding provides additional personnel shielding around the top of the transfer cask during canister closure operations. Dose rates are measured at the canister lid to ensure that the dose rates are within expected values.

The canister water level is lowered slightly, the canister is vented, and the canister lid is seal-welded using an automated welding system. Liquid penetrant examinations are performed on the root and final passes. An ultrasonic or multilayer liquid penetrant examination is performed on the canister lid-to-shell weld to ensure that the weld is satisfactory. The canister lid-to-shell weld is then pressure-tested followed by an additional liquid penetrant examination performed on the canister lid-to-shell weld to verify structural integrity.

Moisture is removed from the canister using a vacuum drying system or forced helium dehydration system. For canisters without high burn-up fuel, the vacuum drying system may be connected to the canister and used to remove all liquid water from the canister in a stepped evacuation process. A stepped evacuation process is used to preclude the formation of ice in the canister and vacuum drying system lines. The internal pressure is reduced to below 3 torr and held for 30 min to ensure that all liquid water is removed.

An alternative process using a forced helium dehydration system can also be utilized to remove residual moisture from the canister. Gas is circulated through the canister to evaporate and remove moisture. The residual moisture is condensed until no additional moisture remains in the canister. The temperature of the gas exiting the system demister is maintained below 21°F for a minimum of 30 min to ensure that all liquid water is removed.

Following canister moisture removal, the canister is backfilled with a predetermined amount of helium gas. The helium backfill ensures adequate heat transfer during storage

and provides an inert atmosphere for long-term fuel integrity. Cover plates are installed and seal welded over the canister vent and drain ports with liquid penetrant examinations performed on the root and final passes (for multipass welds). The cover plate welds are then leak-tested. The final closure ring or plate is then aligned and seal welded providing redundant closure of the canister confinement boundary closure welds.

Temporary shielding (if utilized) is removed and the remaining water in the annulus between the canister and the transfer cask is drained. The canister lid and accessible areas of the top of the canister shell are smeared for removable contamination and the transfer cask dose rates are measured. The transfer cask top lid is installed and the bolts are torqued.

At this point the loading operations are distinct based on whether a vertical or horizontal system is being loaded.

#### **13.3.9.1 Vertical System—Aboveground**

A mating device is positioned on top of the storage overpack. The loaded transfer cask is placed on top of storage overpack. The mating device helps guide the transfer cask during this operation. The canister is raised slightly, the transfer lid door locking pins are removed, and the doors opened. The transfer cask bottom lid is removed and the mating device drawer is opened. The canister is lowered into storage overpack. The transfer cask is removed from the top of storage overpack. The mating device with pool lid and vent duct shield inserts (if used) are removed, and the storage overpack lid is installed. The exit vent gamma shield cross plates, temperature elements (if used), and vent screens are installed. The storage overpack is secured to an on-site transporter (as applicable) and moved to the ISFSI pad. Once located at the storage pad, the inlet vent gamma shield cross plates are installed and the shielding effectiveness test is performed. Finally, the temperature elements and their instrument connections are installed.

#### **13.3.9.2 Vertical System—Underground**

The transfer cask is moved from the SFP to a location where the transfer cask can be moved outside the building. The transfer cask is secured to an on-site transporter (as applicable) and moved to the ISFSI pad. The top lid of the canister enclosure is removed and the mating device is located on top of the canister enclosure. The canister is transferred from the transfer cask to the canister enclosure in the same way as the aboveground system, with the exception of the on-site transporter performing the transfer capabilities. Optional temporary shielding may be installed to reduce streaming and operational dose to workers. Once transfer is completed and the canister is verified to be fully lowered, the transfer cask and mating device are removed and the top lid is installed. Finally, temperature monitoring elements are installed and connected and post-loading performance verification is completed.

#### **13.3.9.3 Horizontal System**

The transfer cask is lowered onto a transfer trailer in a vertical position so that the transfer cask lower trunions engage with the transfer trailer. The transfer cask is then downended into a horizontal configuration onto the transfer trailer. The transfer trailer with the loaded transfer cask is towed to the horizontal module at the ISFSI. The transfer trailer is positioned in front of the storage module with the transfer cask door and the storage module door removed. Using optical survey equipment and cask alignment marks the transfer cask is adjusted until it is aligned with the storage module. A hydraulic ram then pushes

the canister through the transfer cask into the storage module. The empty transfer cask is removed from the ISFSI and the door to the storage module is bolted in place. Temperature measurements are made to ensure adequate heat removal is achieved.

### 13.3.10 License Renewal

Since the first ISFSI was first licensed in 1986, and the first generic cask certificates were awarded in the 1990s with license periods of 20 years, there has been a need to “renew” the storage facilities and cask certificates for additional periods of time. This has been necessitated by the failure of the DOE to satisfy its contractual obligation to accept the spent fuel from utilities and take title to the fuel. Therefore, the spent fuel continues to be stored and maintained at each facility license under Part 72.

As with license renewal for the nuclear power plants, the possibility of aging effects and the changes to materials, and so on must be considered as time progresses. The primary method to address these issues is two part: (1) completion of time-limited aging analyses which predict the effects of aging, effect of the environment on materials, and so on over the expected license period; and (2) implementation of an aging management program which identifies parameters and components to be monitored with a subsequent periodic inspection of those components important to safety.

---

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# 14

## *Nuclear Fuel Recycling*

Patricia Paviet and Michael F. Simpson

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## 14.1 Objectives and Challenges of Recycling Used Nuclear Fuel

Over the past 50 years, the principal reason for recycling used nuclear fuel (UNF) has been to recover unused uranium (U) and plutonium (Pu) in the used fuel elements, thereby avoiding the wastage of a valuable resource. Most of the used fuel—about 95% uranium at 1% U-235 and up to 1% plutonium—can be recycled as fresh fuel, saving some 30% of the natural uranium otherwise required. It also avoids leaving the plutonium in the used fuel, where in a century or two the built-in radiological protection will have diminished, possibly allowing it to be recovered for illicit use (though it is unsuitable for weapons due to the nonfissile isotopes present). UNF recycling is performed in several countries using the plutonium and uranium recovery by the PUREX (originally “Plutonium Uranium EXtraction” but also found in the literature as “Plutonium Uranium Recovery by Extraction” or “Plutonium Uranium Reduction Extraction”) process (Table 14.1).

Even though the process is chemically and radiochemically well understood and proven to be commercially available, the prospect of recycling UNF comes with several concerns such as (1) this technology could contribute to the proliferation of plutonium, (2) the management of UNF and high-level waste (HLW) is a long-term process because

**TABLE 14.1**

World Commercial Reprocessing Capacity (tons per year)

**LWR fuel:**

France, La Hague	1700
UK, Sellafield (THORP)	900
Russia, Ozersk (Mayak)	400
Japan	14
Total approx	3000

**Other nuclear fuels:**

UK, Sellafield	1500
India	275
Total approx	1750

**Total civil capacity** **4750**

Source: OECD/NEA. 2004. Nuclear energy data. Nuclear engineering. *International Handbook*. <http://www.world-nuclear.org/info/inf69.html>.

it requires several decades of interim followed by a final disposal, and (3) the cost associated with recycling activities could be an economic disadvantage. Furthermore, the United States does not encourage the civil use of plutonium and, accordingly, does not itself engage in plutonium recycling for nuclear power. However, as pointed out by Lauvergeon (2009):

The renewed interest in nuclear energy and the international growth of nuclear electricity generation do not equate—and should not be equated—with increasing proliferation risks. Indeed, the nuclear renaissance presents a unique opportunity to enhance the culture of nonproliferation.

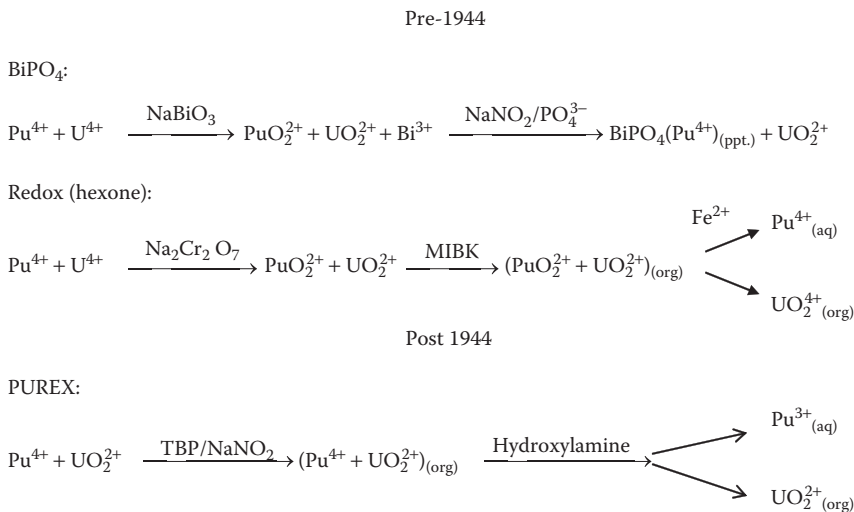
Finally, the alarming observations made by the Intergovernmental Panel on Climate Change (IPCC) issued in September 2013, warning that “continued emissions of greenhouse gases (GHG) will cause further warming and changes in all components of the climate system” pleaded in favor of a sustainable development of nuclear energy. The sustainability of nuclear energy implies strategic long-term planning for managing the nuclear waste which requires a study of applicable solutions for minimizing the quantity and the hazardous of final waste. One of the avenues appears to be the recycling of UNF with advanced R&D flowsheets proposed for actinides separation. Even though nuclear fuel reprocessing has been in standby for the past 30 years in the United States, one new project, called the MOX project, deserves special attention. In 1999, the National Nuclear Security Administration (NNSA) signed a contract with a consortium, now called Shaw AREVA MOX Services, LLC to design, build, and operate a Mixed Oxide Fuel Fabrication Facility (MFFF). This facility will be a major component in the United States’ program to dispose of 34 MT of surplus weapon-grade plutonium. If licensed, the MFFF would receive weapon-grade plutonium dioxide,  $\text{PuO}_2$ , from the US Department of Energy (DOE)’s Pit Disassembly and Conversion Facility (Paviet-Hartmann et al., 2010).

Separations developed in the 1960s based on molten salts, molten metal, including electrochemical processing in these media, have received considerable attention in recent years for their potential as alternative large-scale separation methods for UNF processing (Morss, Edelstein, and Fuger, 2006). Processes based on the volatility of certain actinide compounds have also received some attention and possess some interesting features. However, each of these concepts is far behind aqueous processing because of the number of years of experience that have been accumulated for the aqueous option, and the unknowns always attendant to developing new science and technology.

In the last decade, interest has grown in separating (partitioning) individual radionuclides to reduce long-term radioactivity in residual wastes and to be able to transmute separated long-lived radionuclides into shorter lived ones, mostly by fission. In 2005, this interest in more fully closing the fuel cycle grew and became more public, driven by concerns about long-term resource utilization and proliferation resistance (Mc Farlane, 2007). The main challenge that recycling is facing today is to develop, demonstrate, and deploy economic, proliferation-resistant nuclear fuel recycling processes appropriate for the power grids of developing countries and regions. With the expansion of worldwide energy demand, which may lead to a massive growth of nuclear energy, advanced nuclear technologies must solve key issues, such as waste accumulation, global proliferation of nuclear materials, continuous improvement of economy, and maintenance of an excellent safety record.

## 14.2 Commercial Reprocessing: History and Current Status

Precipitation remained the predominant separation technique in radiochemistry until the development of the atomic bomb during the Manhattan Project. The scientists working on this project were the first to separate plutonium from uranium and its decay and fission products (FPs) using kilogram-scale radiochemical separation. Reprocessing of spent nuclear fuels began during World War II to obtain weapons materials. Reactors then were production reactors because their primary purpose was the production of plutonium. At this time, several methods for extracting plutonium from irradiated uranium were studied and a precipitation technique, the bismuth phosphate process, was selected in view of the small quantities being handled. This technique did not recover uranium.  $\text{BiPO}_4$  was used as the carrier for the insoluble phosphates of Pu(III) and Pu(IV) (Figure 14.1) (Choppin, Khankhasayev, and Plendl, 2002, 4, 5). The bismuth phosphate process was abandoned because of high waste volumes and relatively low recovery. There was no interest in recovering the uranium from irradiated fuels for economic reasons. The redox process (Figure 14.1) (Choppin, Khankhasayev, and Plendl, 2002, 4, 5) was the first solvent extraction system adopted on a large scale which used methyl isobutyl ketone (MIBK or hexone) as a solvent and an aqueous nitrate solution as the feed material. The major experience with the redox process has been obtained using aluminum nitrate as the salting agent. Although several methods of extraction were actively studied, solvent extraction methods, in particular the PUREX process which uses tri-*n*-butyl-phosphate (TBP) as the solvent forged ahead and has come to dominate the reprocessing industry (Selvaduray, 1978). TBP has the advantages of being more stable, less flammable, and results in better separation than hexone. The PUREX process, using a solution of 30% TBP diluted in kerosene, dodecane, or hydrogenated polypropylene tetramer (HPT) as the organic phase, has been the most employed of all large-scale radiochemical separation techniques and remains in common use today (Choppin, Khankhasayev, and Plendl, 2002, 4, 5).



**FIGURE 14.1**

Fuel Reprocessing Chemistry. (From Choppin, G., M.K. Khankhasayev, and H.S. Plendl, eds. 2002. *Chemical separations in nuclear waste management, the state of the art and a look to the future*. DOE/EM-0591. Institute for the International Cooperative Environmental Research, Florida State University, 96. Columbus, OH: Battelle Press.)

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### 14.3 Irradiated Oxide Reactor Fuel: Composition and Chemical State

The chemical composition and the associated phases of irradiated nuclear fuel can be determined by the state-of-the art post-irradiation examination (PIE) as established at AEC, Cadarache, France, and at the Institute of Transuranic Elements (ITU) at Karlsruhe, Germany. A suite of shielded PIE equipment called the Irradiated Materials Characterization Laboratory (IMCL) is currently being installed at the Idaho National Laboratory (INL) (Christensen and Lehto, 2008, 8). Plans include to install in an individual shielded enclosure, a shielded electron probe micro-analyzer (EPMA), a focused ion beam instrument (FIB), and a micro-scale X-ray diffractometer (MXRD). This hot-cell environment where the instruments are installed side by side allows researchers to quickly pass samples between instruments. New PIE equipments at INL allow researchers to qualify and understand the behavior of new nuclear test fuels on a micro-scale. PIE will confirm the overall composition of irradiated fuel as provided by calculations readily from the established fission yields and cross-section data. Examples of calculations (IAEA, 1974) differentiate due to the type of fission, captured cross-sections, and irradiation time (burn-up). The calculated compositions are mainly applied to ceramic oxide fuels and changes in oxygen balance occurring with burn-up are established (Holleck and Kleykamp, 1970) where the redistribution of oxygen under the radial temperature gradient of the fuel element is considered in relation to its effect on oxygen to metal ratios and oxygen potentials.

The chemical state of reactor fuel material during irradiation is influenced by the compositional changes resulting from the fission process and by the redistribution of fuel- and FP phases (Findlay, 1974). Phase redistribution is of particular concern in fast-spectrum reactors, where the build-up of FPs is substantial and operating temperatures are high. The primary need of thermochemical assessment models for irradiated fuel is to establish the composition of irradiated fuel depending on its irradiation history. Therefore, all members of FP chains have to be considered, but simplification for short-lived isotopes is possible. The species that contribute principally on a mass basis are the stable and long-lived FP, and the importance of compositional changes as a result of burn-up and transmutation has to be considered. A successful computer program to address the needs was developed at Atomic Energy Research Establishment (AERE), Harwell, to specifically calculate chemical fuel compositions after 10% burn-up for a variety of idealized cases (IAEA, 1974), and the final concentrations listed in Table 14.2. Cases considered are the thermal fission of U-235, and the fast fission of U-235 and Pu-239.

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### 14.4 Evolution of Reprocessing Methods

The development of large-scale radiochemical separation technology was started in 1942 by the scientific staff of the Metallurgical Laboratory, a part of the Manhattan Engineering District organization that was created during World War II to develop nuclear weapons for military applications. The mission of the Separations Section of the Metallurgical Laboratory was to develop an industrial-scale chemical process to recover and purify plutonium from the uranium targets that were to be irradiated in the “neutron chain reactors” proposed for construction at Hanford, Washington (Thompson and Seaborg, 1956).

TABLE 14.2

Fission Product and Heavy Metal Concentration in Irradiated Fuel for 10% Burn-Up as Percentage of Initial Heavy Metal Content

Case #		1 Thermal	2 Thermal	3 Thermal	4 Fast	5 Fast	6 Fast
Initial Heavy Metal Concentration	235-U	100	100	100	100	0	0.50
	238-U	0	0	0	0	0	69.02
	239-Pu	0	0	0	0	100	23.94
	240-Pu	0	0	0	0	0	5.47
	241-Pu	0	0	0	0	0	0.93
	242-Pu	0	0	0	0	0	0.14
Fuel rating (W/g)		10	1000	1000	1000	200	200
Irradiation time(days)		10,150	102	102	86	514	476
Cooling time (days)		0	0	100	0	0	0
Final heavy metal content	235-U	88.38	88.22	88.38	86.89	0	0.28
	236-U	1.73	1.74	1.73	3.05	0	0.47
	237-Np	$1 \times 10^{-3}$	$1 \times 10^{-3}$	$1 \times 10^{-3}$	$4.1 \times 10^{-2}$	0	$2.6 \times 10^{-3}$
	238-U	0	0	0	0	0	64.50
	239-Pu	0	0	0	0	87.25	17.44
	240-Pu	0	0	0	0	2.70	6.36
	241-Pu	0	0	0	0	$4.9 \times 10^{-2}$	1.08
	241-Am	0	0	0	0	$1.2 \times 10^{-3}$	$6.4 \times 10^{-2}$
	242-Pu	0	0	0	0	$4.8 \times 10^{-4}$	0.22
Fission product concentration	Ge	$4.0 \times 10^{-4}$	$4.0 \times 10^{-4}$	$4.0 \times 10^{-4}$	$3.2 \times 10^{-2}$	0	$7.0 \times 10^{-5}$
	As	$1.2 \times 10^{-4}$	$1.4 \times 10^{-4}$	$1.2 \times 10^{-4}$	$1.9 \times 10^{-3}$	$1.0 \times 10^{-5}$	$4.0 \times 10^{-5}$
	Se	$4.2 \times 10^{-2}$	$4.2 \times 10^{-2}$	$4.2 \times 10^{-2}$	$8.6 \times 10^{-2}$	$5.0 \times 10^{-2}$	$4.8 \times 10^{-2}$
	Br	$1.8 \times 10^{-2}$	$1.8 \times 10^{-2}$	$1.8 \times 10^{-2}$	$3.1 \times 10^{-2}$	$1.8 \times 10^{-2}$	$1.8 \times 10^{-2}$
	Kr	0.35	0.37	0.37	0.39	0.19	0.20
	Rb	0.37	0.36	0.36	0.38	0.17	0.17
	Sr	0.78	1.20	1.0	1.18	0.37	0.39
	Y	0.48	0.57	0.52	0.56	0.20	0.21
	Zr	3.26	3.16	3.14	3.10	1.99	2.02
	Nb	$3.0 \times 10^{-3}$	0.14	0.11	0.14	$5.0 \times 10^{-2}$	$5.0 \times 10^{-2}$
	Mo	2.48	1.99	2.25	1.96	2.07	2.06
	Tc	0.61	0.60	0.62	0.57	0.58	0.58
	Ru	1.10	1.28	1.15	1.52	2.12	2.11
	Rh	0.29	0.16	0.27	0.18	0.52	0.52
	Pd	0.16	0.12	0.13	0.23	1.42	1.37
	Ag	$3.0 \times 10^{-3}$	$3.2 \times 10^{-3}$	$3.0 \times 10^{-3}$	$1.5 \times 10^{-2}$	0.16	0.16
	Cd	$7.7 \times 10^{-3}$	$7.6 \times 10^{-3}$	$7.7 \times 10^{-3}$	$1.9 \times 10^{-2}$	$8.1 \times 10^{-2}$	$8.3 \times 10^{-2}$
	In	$9.6 \times 10^{-4}$	$9.2 \times 10^{-4}$	$9.7 \times 10^{-4}$	$2.8 \times 10^{-3}$	$8.9 \times 10^{-3}$	$8.6 \times 10^{-3}$
	Sn	$8.6 \times 10^{-3}$	$8.9 \times 10^{-3}$	$8.6 \times 10^{-3}$	$3.7 \times 10^{-2}$	$5.3 \times 10^{-2}$	$5.1 \times 10^{-2}$
Sb	$3.3 \times 10^{-3}$	$6.2 \times 10^{-3}$	$4.9 \times 10^{-3}$	$5.4 \times 10^{-2}$	$3.6 \times 10^{-2}$	$3.4 \times 10^{-2}$	

(Continued)

**TABLE 14.2 (Continued)**

Fission Product and Heavy Metal Concentration in Irradiated Fuel for 10% Burn-Up as Percentage of Initial Heavy Metal Content

Case #	1 Thermal	2 Thermal	3 Thermal	4 Fast	5 Fast	6 Fast
Te	0.26	0.29	0.26	0.35	0.34	0.32
I	0.11	0.14	0.11	0.21	0.17	0.16
Xe	2.16	2.61	2.61	2.10	2.04	2.05
Cs	1.75	1.42	1.47	1.88	1.88	1.87
Ba	0.84	0.78	0.68	0.81	0.65	0.65
La	0.65	0.66	0.65	0.63	0.53	0.54
Ce	1.24	1.84	1.63	1.78	1.27	1.29
Pr	0.61	0.48	0.58	0.48	0.40	0.42
Nd	2.02	1.49	1.68	1.48	1.41	1.44
Pm	$2.9 \times 10^{-2}$	0.18	0.19	0.19	0.16	0.17
Sm	0.37	0.18	0.20	0.21	0.36	0.37
Eu	$1.8 \times 10^{-2}$	$1.8 \times 10^{-2}$	$1.8 \times 10^{-2}$	$2.7 \times 10^{-2}$	$6.9 \times 10^{-2}$	$6.9 \times 10^{-2}$
Gd	$4.3 \times 10^{-3}$	$3.5 \times 10^{-3}$	$4.3 \times 10^{-3}$	$4.9 \times 10^{-3}$	$4.5 \times 10^{-2}$	$4.3 \times 10^{-2}$
Tb	$1.0 \times 10^{-4}$	$1.1 \times 10^{-4}$	$1.1 \times 10^{-4}$	$3.4 \times 10^{-3}$	$4.5 \times 10^{-3}$	$4.4 \times 10^{-3}$
Dy	$1.0 \times 10^{-5}$	$1.0 \times 10^{-5}$	$1.0 \times 10^{-5}$	$4.0 \times 10^{-5}$	$1.5 \times 10^{-3}$	$1.5 \times 10^{-3}$

Source: IAEA. 1974. *Behaviour and chemical state of irradiated fuels*. STI/PUB/303 Vienna, Austria: International Atomic Energy Agency.

The underlying separation principles and many of the techniques developed and used at that time are essentially the same as those in use today.

Much of the basic chemistry of thorium and uranium was known in 1942, but the nuclear decay characteristics of most of the (FPs) were not. Furthermore, the chemistry of many of the FPs and transuranic (TRU) elements was not known in sufficient detail. Promethium, technetium, and all the TRU elements were new to science and much had to be inferred from an element's position in the periodic table. The chemical and physical effects of radiation imposed additional difficulties and uncertainties in the proposed processes, as they do even today.

The concentration of the plutonium to be produced in the reactors at Hanford was quite low, of the order of 300 g of plutonium per ton of irradiated uranium. (The only plutonium available for process development activities prior to the construction of the graphite reactor in Oak Ridge in 1943 was about a milligram of  $^{239}\text{Pu}$  created by extended cyclotron bombardment of uranium.) The separation processes had to have high recovery efficiencies, while also achieving high separation of the plutonium from the radioactive FPs that were produced in the uranium targets along with the plutonium. It was a monumental challenge to the Metallurgical Laboratory chemists, most of whom were new to this field of industrial-scale separations chemistry.

Several processes have been proposed for carrying out the needed separations. Some of them have been demonstrated at the bench scale and not necessarily with the levels of radioactivity that would be present in the actual wastes; a few others have been demonstrated on a very large scale at high levels of radioactivity and used on an industrial scale. For example, the PUREX process, an aqueous process employing solvent extraction techniques, was developed for, and has been used extensively in, reactor target processing for plutonium production for weapons in the United States, and for weapons and commercial fuel recycling in power reactors in France, the United Kingdom, Japan, and the former Soviet Union. Pyroprocesses that typically employ high-temperature, nonaqueous

systems have been used for many years in the nuclear as well as conventional industries, for example, for electrorefining of plutonium and americium—though they have had limited use for processing of HLW requiring remote operations.

Reprocessing technologies can be grouped into two categories: those based primarily on aqueous chemistry and those based on nonaqueous materials, such as molten salts and molten metals.

### 14.4.1 Aqueous Processes

#### 14.4.1.1 Bismuth Phosphate

Early work on precipitation processes for the separation of plutonium implies the bismuth phosphate process (Benedict, Pigford, and Levi, 1981; Cleveland, 1979; Etherington, 1958; Flagg, 1961; Hill and Cooper, 1958; Lawroski, 1955; Long, 1978; Morss et al., 2006; Perlman, Thompson, and Cunningham, 1961; Stoller and Richards, 1961; Thompson and Seaborg, 1956; Wick, 1980). The bismuth phosphate process is based on the ability of a  $\text{BiPO}_4$  precipitate to carry  $\text{Pu(IV)}$  but not  $\text{Pu(VI)}$ . By alternate precipitations with plutonium in the tetravalent and hexavalent states, a high degree of separation (95%–98%) of plutonium from uranium and FPs is achieved. The process (Figure 14.2) was designed to extract plutonium from aluminum-clad uranium metal fuel. The fuel was decladded by boiling it in caustic soda. After decladding, the uranium metal is dissolved in nitric acid and the solution goes

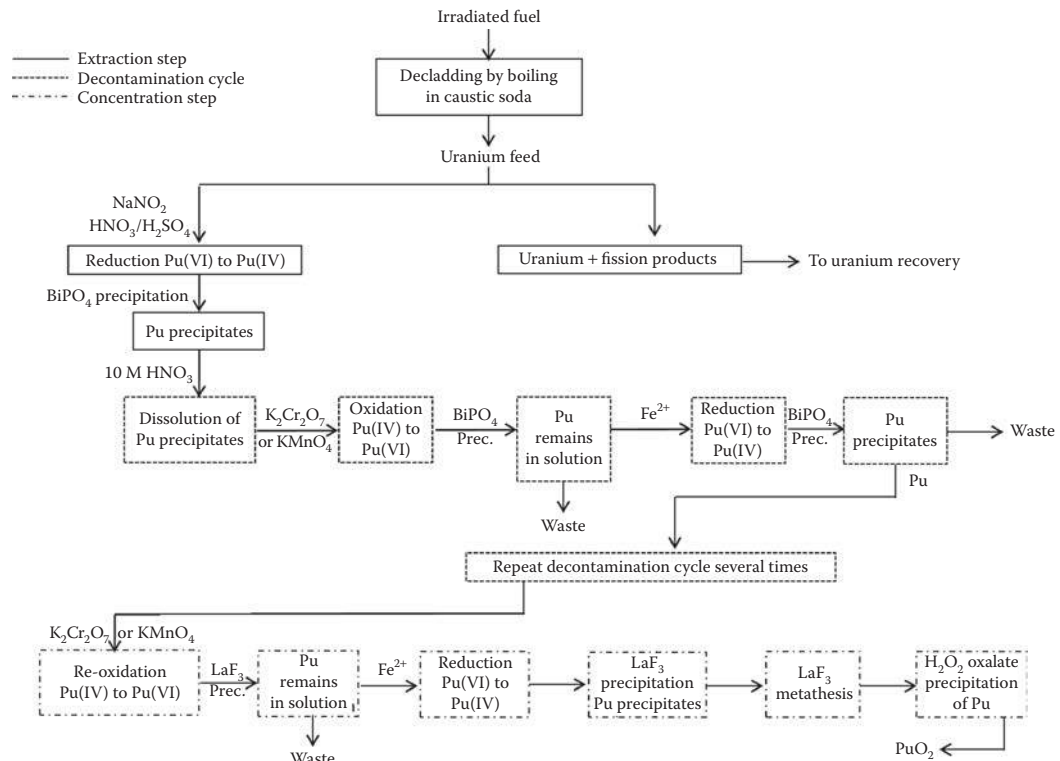


FIGURE 14.2  
Flowsheet of bismuth precipitation.



through an extraction step, where plutonium is separated from uranium and the FPs. The plutonium is reduced to the tetravalent state with  $\text{NaNO}_2$  and then coprecipitated with  $\text{BiPO}_4$ , the uranium being held in the solution by a complexing agent  $\text{SO}_4^{2-}$ . Under such conditions, U(VI) is sufficiently complexed by sulfate ion to prevent the precipitation of uranyl phosphate. The precipitate is dissolved in the  $\sim 10$  M nitric acid solution, and passed through several decontamination cycles to purify plutonium. Each cycle consists of a by-product precipitation in which the plutonium is oxidized to the hexavalent state by a strong oxidizer, such as  $\text{K}_2\text{Cr}_2\text{O}_7$  or  $\text{KMnO}_4$ , and stays in the solution. It can then be reduced to Pu(IV) by  $\text{Fe}^{2+}$  and is precipitated by  $\text{BiPO}_4$ . A few cycles of  $\text{BiPO}_4$  precipitation and dissolution followed by a final  $\text{LaF}_3$  precipitation considerably increases the concentration of Pu in the solution and final recovery can be accomplished by plutonium peroxide and then oxalate precipitations. Typical decontamination factors obtained with the process are shown in Table 14.3. The inability to operate the process continuously (separations were made batch wise) and its rejection of the still useful enriched uranium caused the discontinuation of its application in favor of solvent extraction processes, which overcame both of these limitations.

#### 14.4.1.2 Redox Process

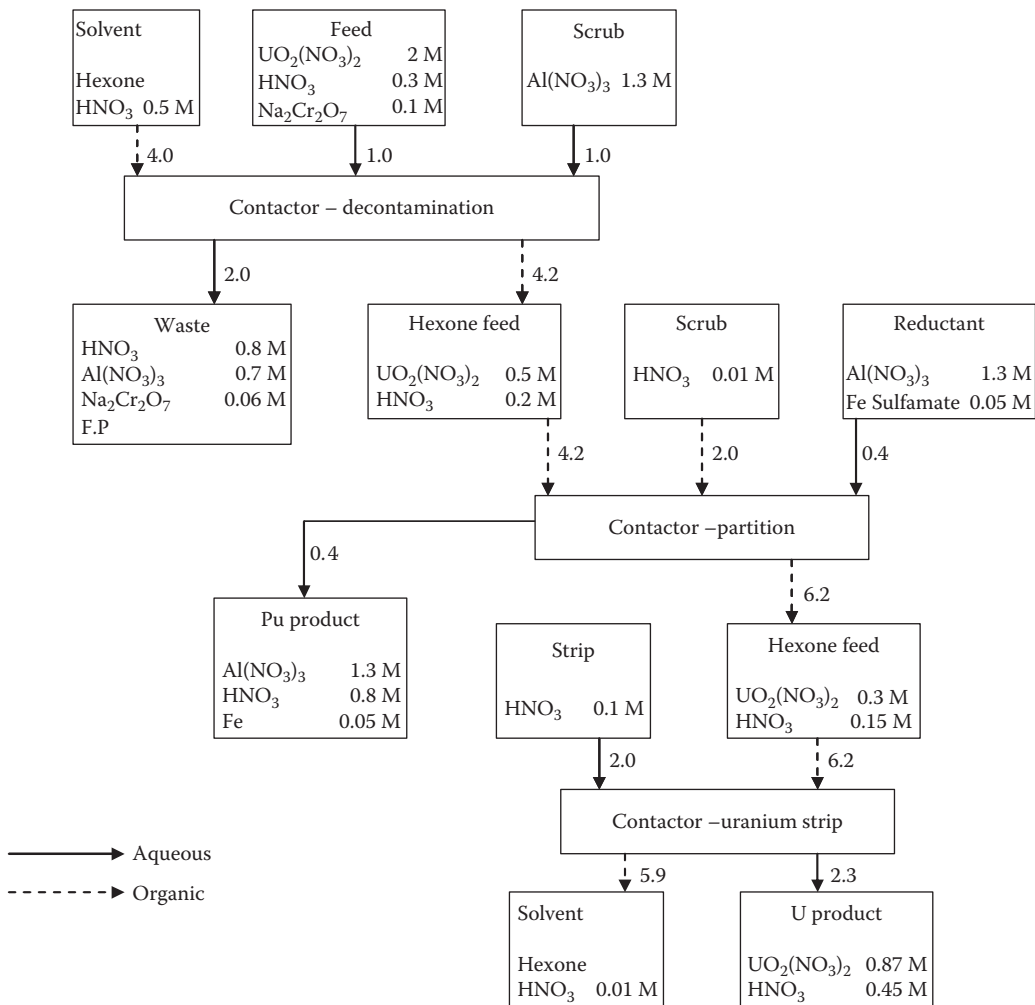
The first solvent extraction process used in the United States for large-scale separation of uranium and plutonium from irradiated fuel was the redox process (Figure 14.3). This process was developed by Argonne National Laboratory, tested in a pilot plant at Oak Ridge National Laboratory in 1948–1949 and installed by the General Electric Company at the Hanford plant in 1951 (Benedict, Pigford, and Levi, 1981, 459; Hanthorn, 1949; Isaacson, 1964; Zebroski and Feber, 1948). The redox process, based on MIBK (hexone) extraction, was developed to extract plutonium and uranium (and also neptunium) from oxidizing solutions (Benedict, Pigford, and Levi, 1981; Cleveland, 1979; Etherington, 1958; Flagg, 1961; Long, 1978; Morss et al., 2006; Selvaduray, 1978; Wick, 1980). This method was called the “redox process” because of its use of oxidation–reduction separations chemistry (Hill and Cooper, 1958).

The counterflow extraction process was carried out in stainless steel equipment and used column cascade extraction technology. The aqueous feed consisted primarily of nitric acid that contained the FPs and TRU elements as nitrate salts. The plutonium was oxidized to the hexavalent state with  $\text{Na}_2\text{Cr}_2\text{O}_7$ . The aluminum nitrate salting agent is added, after

**TABLE 14.3**  
Typical Decontamination Factors with the Bismuth Phosphate Process (Flagg, 1961)

Operation	Decontamination Factor	
	Individual	Overall
Extraction Step	7.5	7.5
First decontamination cycle	150	$1.1 \times 10^3$
Second decontamination cycle	90	$1 \times 10^5$
Concentration step	100	$1 \times 10^7$

*Note:* The inability to operate the process continuously (separations were made batch wise) and its rejection of the still useful enriched uranium caused the discontinuation of its application in favor of the solvent extraction processes, which overcame both of these limitations.



**FIGURE 14.3**

Redox process flowsheet. (Adapted from Benedict M., *Nuclear Chemical Engineering*, Pigford T.H., H.W. Levi, Eds, Second Ed., Chap 10—Fuel Reprocessing. McGraw-Hill, 1981. With permission.)

which the solution is contacted with hexone to extract the Pu(VI) and U(VI) in the organic phase, and, essentially, all of the radioactive FPs and the excess oxidant and its reduction product are rejected to waste in the first process step, minimizing oxidation and radiolysis problems of the solvent and radiolysis of downstream reagents needed for plutonium isolation (Cleveland, 1979, 486; Culler, 1956, 201–211).

Recovery of plutonium from the organic solvent was achieved by contacting the organic stream with a dilute nitric acid aqueous phase that was heavily salted with aluminum nitrate and contained a moderately strong reducing agent (Culler, 1956, 172–194). Plutonium may then be stripped from the hexone with an aqueous solution of Al(NO<sub>3</sub>)<sub>3</sub> and ferrous sulfamate (Fe(SO<sub>3</sub>NH<sub>2</sub>)<sub>2</sub>), which reduces Pu(VI) to the nonextractable Pu(III). U(VI) remained in the organic phase. The uranyl ion was stripped from the solvent with dilute nitric acid later in the process. Typical decontamination factors obtained with the process are shown in Table 14.4.

**TABLE 14.4**

Plant Scale Performance of Redox Process (Cleveland 1979: 486)

Separation of uranium from plutonium	>10 <sup>7</sup>
Separation of plutonium from uranium	10 <sup>6</sup>
Decontamination of fission products from plutonium	10 <sup>8</sup>
Decontamination of fission products from uranium	10 <sup>7</sup>
Plutonium recovery, %	99.8
Uranium recovery, %	99.9
Solvent recovery, %	99.2

The redox process has been supplanted by the PUREX process because of the low flash point and higher toxicity of hexone compared with the TBP-kerosene solutions utilized in PUREX. Furthermore, the reactivity of concentrated HNO<sub>3</sub> toward hexone prevented its use as a salting agent. A metal salt such as Al(NO<sub>3</sub>)<sub>3</sub> was used instead. The resulting high salt concentration of the aqueous waste seriously limited the extent to which it could be concentrated, and thus resulted in greater waste volumes. Moreover, at Hanford, the process-generated aqueous wastes were made basic and stored in underground tanks. The Al<sub>2</sub>O<sub>3</sub> that was generated as a by-product of this process complicates current efforts to remediate tank wastes (Morss et al., 2006, 2730–2731).

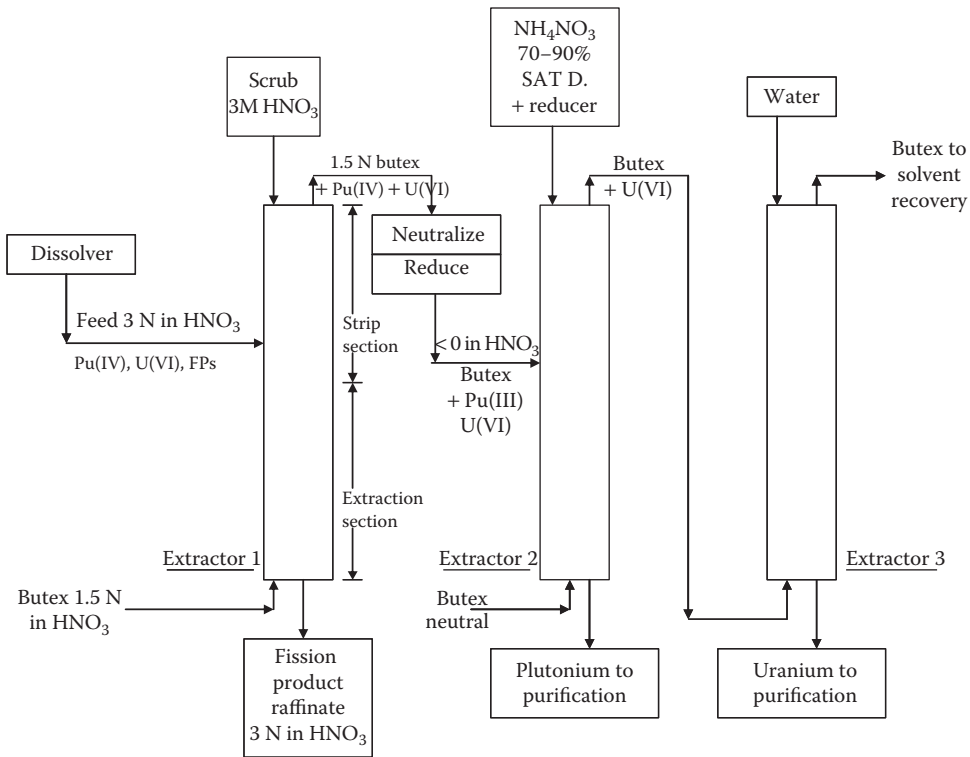
#### 14.4.1.3 Butex Process

The chemical aspects of the development of the Butex process (Figure 14.4) have been described by Spence (1957) and Howells et al. (1958). The process developed in the United Kingdom utilized two solvents: undiluted dibutyl carbitol (butex, C<sub>4</sub>H<sub>9</sub>OC<sub>2</sub>H<sub>4</sub>OC<sub>2</sub>H<sub>4</sub>OC<sub>4</sub>H<sub>9</sub>) for primary separation and uranium purification, and 20% TBP in kerosene for plutonium purification (Wick, 1980, 532–540). Butex has a greater stability than hexone toward nitric acid. It can be used with nitric acid only as the salting-out agent, thus allowing concentration of FP wastes by evaporation. In addition, it is less flammable than hexone. The Butex solvent, while having a high flash point, has three disadvantages (Morss et al., 2006, 2731): high viscosity, high density (close to that of water), and a tendency to form crystalline uranium complexes of the type UO<sub>2</sub>(NO<sub>3</sub>)<sub>2</sub>·3H<sub>2</sub>O·C<sub>12</sub>H<sub>26</sub>O<sub>3</sub>. As a result, this process was discontinued as more efficient processes such as the PUREX process emerged. The approximate partition coefficients of U and Pu in BUTEX can be found in Table 14.5.

#### 14.4.1.4 PUREX

The PUREX is the predominant method for reprocessing commercial reactor fuel throughout the world. For this reason, it will be described in more details in Section 14.5 than other fuel separation processes.

In the original PUREX process (Choppin, Khankhasayev, and Plendl, 2002; Morss et al., 2006; NEA, 1997; Richter et al., 2006; Selvaduray, 1978), hexavalent uranium (UO<sub>2</sub><sup>2+</sup> or U(VI)) and tetravalent plutonium (Pu<sup>4+</sup> or Pu(IV)) are extracted from a nitric acid solution of dissolved irradiated fuel by an organic phase composed of TBP diluted to nominal 20%–30% (by volume) with a normal paraffinic hydrocarbon (NPH) organic diluent such as odorless kerosene, or dodecane. Note that the diluent is required only to maintain the physical characteristics of the organic phase (primarily viscosity and density) in a workable range for use in the salient solvent extraction equipment (Herbst, Baron, and Nilsson, 2011, 141–175).



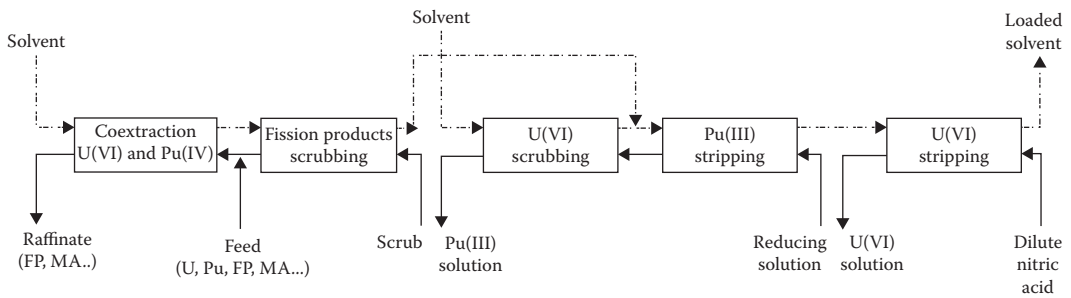
**FIGURE 14.4** Outline of selected Butex process. (From Flagg, J.F., ed. 1961. Nuclear science and technology, chemical processing of reactor fuels, *Other Wet Separation Methods*. New York: Academic press, pp. 305–347, Chap. VIII.)

**TABLE 14.5**  
Approximate Partition Coefficients of U and Pu in BUTEX

	3N HNO <sub>3</sub> in Aqueous Phase	8M Ammonium Nitrate and 0.2N HNO <sub>3</sub> in Aqueous Phase
	K	K
U(VI)	1.5	3
Pu(VI)	1.8	2.5
Pu(IV)	7	7
Pu(III)	<0.01	<0.002

Source: Flagg, J.F., ed. 1961. Nuclear science and technology, chemical processing of reactor fuels, *Other Wet Separation Methods*. New York, NY: Academic press, pp. 305–347, Chap. VIII.

The process aims at specifically separating U(VI) and Pu(IV) from UNF using liquid-liquid extraction (Figure 14.5). The first extraction step leaves most of the FP activities in the aqueous phase. Uranium and plutonium are decontaminated together, and the partitioning of tetravalent plutonium from hexavalent uranium is done by the reduction of Pu(IV) to P(III), which is then back-extracted into nitric acid, while U(VI) remains in the solvent phase. Decontaminated Pu(III) is removed from the organic stream, free of uranium, and the uranium fraction is recovered by back-extraction in very dilute nitric acid.



**FIGURE 14.5**

Simplified countercurrent PUREX flow-sheet. (Adapted from Simpson, M. and J. Law. 2012. Nuclear fuel: Reprocessing of nuclear energy. *Selected entries from the Encyclopedia of Sustainability Science and Technology*. Ed. Nicholas Tsoulfanidis. New York: Springer.)

Plutonium and uranium can be subsequently cleaned to the desired purity and uranium stored or disposed as low-level waste (LLW) if desired.

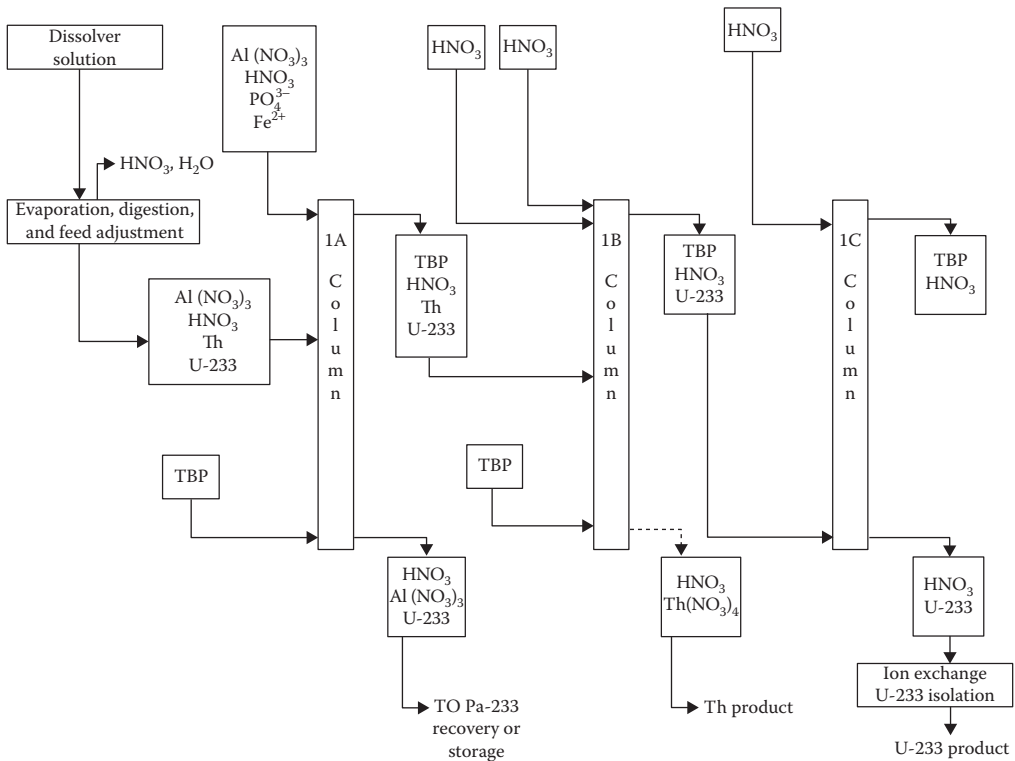
Currently, in various PUREX processes used and planned for in Europe and Japan, plutonium is stripped from the initial extractant containing U(VI) and Pu(IV) and used to fabricate a MOX fuel. Excess uranium is then stored or recycled and the higher actinides and FPs are vitrified, ultimately to be sent to a geological repository (Richter et al., 2006).

In the AREVA version of the PUREX process, plutonium is removed from the uranium after the original extraction, converted to oxide and used in MOX fuel fabrication. Neptunium, americium, curium, and other remaining FPs (including lanthanides) are vitrified and incorporated in the HLW form (Richter et al., 2006).

In Japan's application of the PUREX process, an equal amount of uranium solution is added to the aqueous plutonium stream from the PUREX process. Uranium and plutonium are coconverted to prepare MOX fuel directly without ever producing a pure plutonium oxide fraction at any time during the entire process (Richter et al., 2006).

#### 14.4.1.5 Thorex

Because of the relatively large amount of thorium resources in the world, a thorium-U-233 fuel cycle has been of historical interest. A flowsheet to process the fuels from such a nuclear cycle has seen considerable development over the years, but it remains somewhat a novelty. In general outline, the Thorex process is quite similar to the PUREX process. Two products, thorium-232 and uranium-233, are to be recovered from fuel cycles of high-temperature gas-cooled reactor (HTGR), light-water breeder reactor (LWBR), heavy-water reactor, and the German thorium high-temperature reactor (THTR), freeing them from FPs and separating them from each other (Benedict, Pigford, and Levi, 1981; Flagg, 1961; Selvaduray, 1978). This solvent extraction process employs nitric acid catalyzed with fluoride ion as the thorium dissolution agent. Like the PUREX process, TBP is the extractant diluted to 30–45 volume/% in an aliphatic diluent to extract in the 1A contactor U-233 and Th-232 (Figure 14.6). Protactinium-233, the precursor of U-233, remains in the aqueous phase. Nitric acid may be used as the primary salting agent throughout the Thorex process, but at least some aluminum nitrate is required in the 1A contactor to complex the fluoride ion that is used to catalyze the dissolution of thorium; otherwise, fluoride would complex thorium and interfere with thorium extraction. In a second column 1B (Figure 14.6), thorium is stripped from the organic phase by a dilute nitric acid solution. Finally, the U-233 product is further decontaminated and isolated by ion exchange, as illustrated in



**FIGURE 14.6**

Thorex Process Chemical Flowsheet. (Adapted from Selvaduray, G.S. 1978. Comparative evaluation of nuclear fuel reprocessing techniques for advanced fuel cycle concepts. PhD Dissertation, Stanford University.)

Figure 14.6. The troublesome FPs are ruthenium, zirconium, and niobium. The first cycle of solvent extraction consists of three contactors (1A, 1B, 1C). The usually auxiliary steps such as solvent washing and intercycle evaporation are also included.

One of the major issues of the Thorex process is the creation of a third phase between thorium and TBP if the thorium concentration in the solvent is too high. Furthermore, the partition of uranium and thorium is more difficult than the partition of uranium and plutonium. No change of the thorium oxidation state is required, but the separation of thorium from uranium in the 1B contactor must be obtained entirely by a rather delicate adjustment of salting strength inside the contactor. It appears that the Thorex process has been variable in performance. Decontamination from ruthenium has varied and has been particularly poor when short-cooled thorium-based fuel was processed.

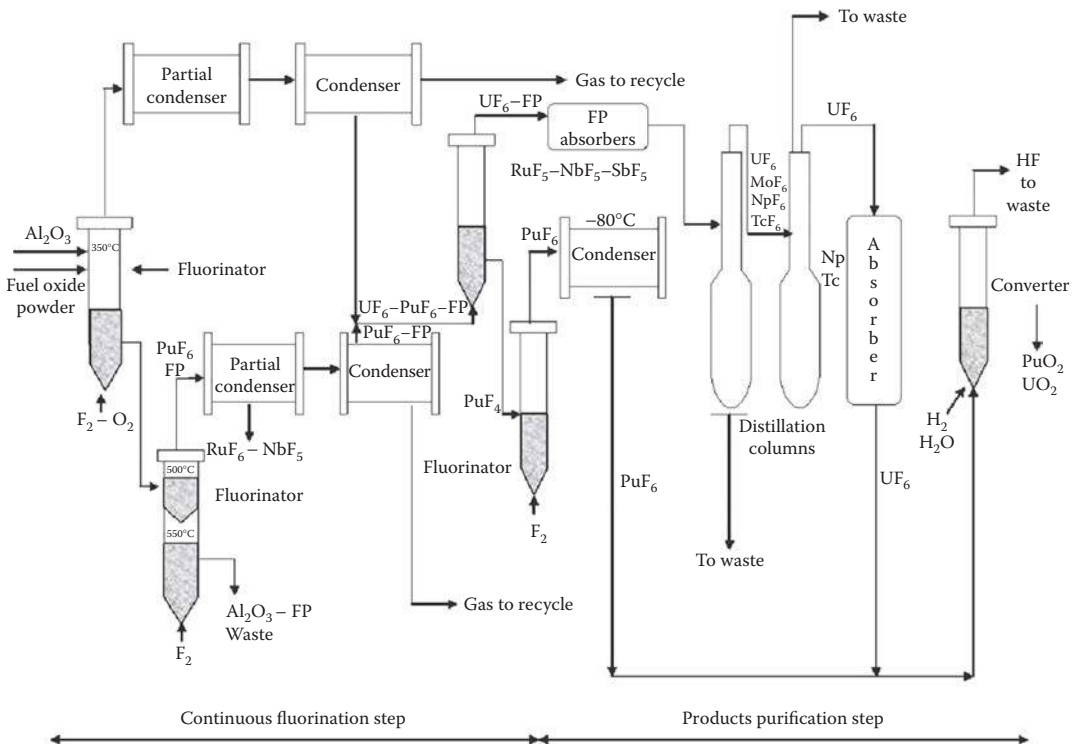
## 14.4.2 Nonaqueous Processes

### 14.4.2.1 Volatility Process: Fluoride Volatility

Fluoride volatility processes utilize the high volatilities of the hexafluoride of plutonium and uranium to separate them from FPs and were investigated primarily for the reprocessing of irradiated oxide fuels (Cleveland, 1979, 504–508; Selvaduray, 1978, 258–259;

Trevorrow, Fischer, and Riha, 1963, 457; Wick, 1980, 544, 545).  $\text{PuF}_6$  and  $\text{UF}_6$  are volatile, but the latter is much more stable and hence more easily formed. Separation of plutonium and uranium by fluoride volatility has never been attempted on a production scale, but the large body of experimental data permits the construction of a full-scale flowsheet with considerable confidence (Figure 14.7). After decladding, the irradiated oxide fuel is fed into the first-stage fluorinator, where it is fluorinated on an alumina bed at 350 °C with a 20% fluorine–80% oxygen mixture, which volatilizes the major part of the uranium as  $\text{UF}_6$ , but 5% of the plutonium. Second-stage fluorination with undiluted fluorine in a two-zone alumina bed at 500 °C and 550 °C volatilizes the remaining plutonium as  $\text{PuF}_6$ . FPs that do not form volatile fluorides remain on the alumina bed. The  $\text{PuF}_6$  and  $\text{UF}_6$  streams are recombined and passed through a fluidized-bed thermal decomposition unit, which converts  $\text{PuF}_6$  to nonvolatile  $\text{PuF}_4$ . The  $\text{UF}_6$  stream is further purified by passing it through alkali and alkaline earth sorption traps to remove residual  $\text{RuF}_5$ ,  $\text{NbF}_5$ , and  $\text{SbF}_5$ . The resulting  $\text{PuF}_6$  is blended with the  $\text{UF}_6$  in the desired proportions after which the mixture is converted to  $\text{PuO}_2\text{-UO}_2$  by reaction with steam and hydrogen in a fluidized bed. Fluoride volatility processes have three principal advantages that are simplicity, a high decontamination factors (of the order of 108), and a separation of uranium in the form necessary for gaseous diffusion plant processing. The main disadvantage is that there are more chemical conversion steps than for other nonaqueous processes.

The fluorination of oxide, carbide, or metallic fuels is usually done in a fused-salt medium or a fluidized bed for better control of temperature and the reaction rate. The use of fused



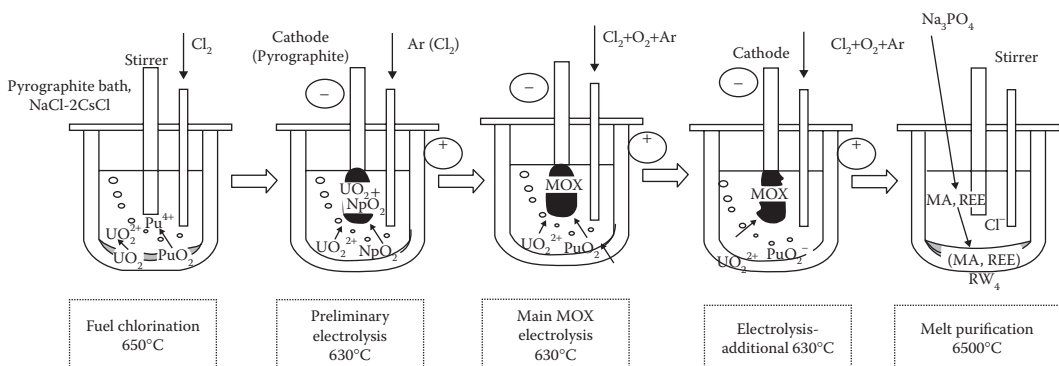
**FIGURE 14.7** Fluoride volatility process. (Adapted from Cleveland, J.M. 1979. *The Chemistry of Plutonium*, 653. La Grange Park, IL: American Nuclear Society.)

salts has been demonstrated for zirconium–uranium alloy fuel, which was immersed in  $\text{NaF-ZrF}_4$  at 600–700 °C and converted to  $\text{ZrF}_4$  and  $\text{UF}_4$  by bubbling anhydrous HF through the melt. Large-scale fluoride volatility plants for spent nuclear fuel reprocessing appear to be feasible, but none have been built. A pilot plant utilizing fluoride volatility as a component of the overall plant system was constructed by Nuclear Research Institute (NRI) Řež of the Czech Republic and operated by the Russia Institute of Atomic Reactors (RIAR) in Dimitrograd, Russia, for the processing of fast-reactor demonstration fuels. This was known as the FREGAT-2 line and was discontinued in 1998 after the re-evaluation of fast reactors by the nuclear power industry. Little information has been released on the utility of this prototype facility or on the economics of volatility processing under the conditions imposed by spent fuel. The pilot-scale FERDA line has more recently been installed at NRI Řež plc. It is capable of processing up to 3 kg of fuel per hour and has been tested with uranium oxides and simulated spent oxide fuel (Uhlřř, Mareček, and Škarohlřřd, 2012). Current testing is primarily the verification of equipment operation before installation in a hot cell. Particular focus has been placed on optimizing the flame fluorination process. Up to 92% conversion of  $\text{U}_3\text{O}_8$  has been achieved and up to 97% conversion of  $\text{UO}_2$  has been achieved.

#### 14.4.2.2 Oxide Electrowinning

RIAR has been developing a process for recovering MOX or  $\text{PuO}_2$  from spent MOX fuel utilizing molten salt solvents (Kofuji et al., 2007; Vavilov, Kobayashi, and Myochin, 2004). This is a critical component of their DOVITA/DOVITA-2 fuel cycle, for which development has been ongoing since 1992. The oxide electrowinning process is designed to be used with MOX-fueled fast reactors such as BN-600 and the soon to be completed BN-800. It consists of the following steps in a molten pool of  $\text{NaCl-KCl}$  at temperatures ranging from 680 °C to 700 °C. These are illustrated in Figure 14.8.

- Chlorination: This serves the purpose of dissolving the fuel into the salt.
- Preliminary electrolysis: This results in codeposition of uranium and neptunium oxides on a cathode.
- MOX codeposition (electrowinning): This results in codeposition of uranium and plutonium oxides on a cathode.



**FIGURE 14.8**  
Oxide Electrowinning Process.



- Additional electrolysis: This draws down the remaining uranium and plutonium from the salt with deposition on a cathode.
- Phosphate precipitation: This step purifies the salt by selectively precipitating rare earths as phosphates. The purified salt can then be used for additional fuel processing.

Chemically, the oxide electrowinning process involves chemical reaction, dissolution, electrolysis, and precipitation. But in theory, it can all be achieved in a single pot or batch of molten salt. This should provide the compactness and simplicity of equipment requirements, even though it must be operated completely within a shielded, inert-atmosphere hot cell. In testing of this process, MOX deposits up to 2.7 kg have been achieved with a composition of 30% Pu/70% U.

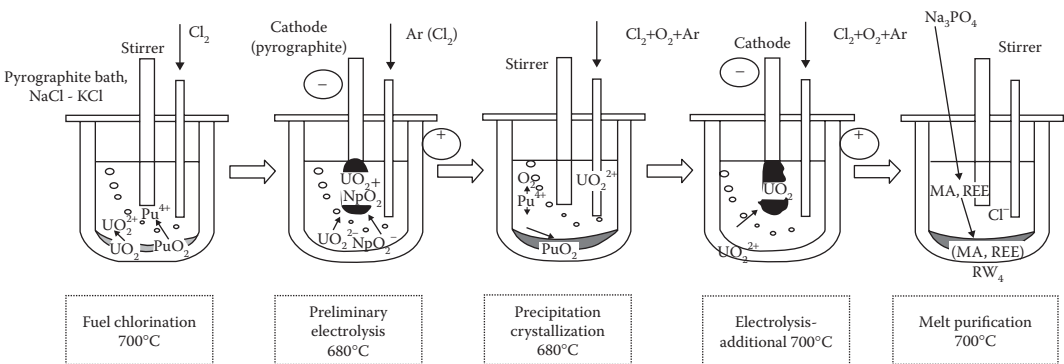
From a safeguard perspective, this process appears to provide the threat of plutonium proliferation. RIAR claims that a precipitation step in which oxygen gas is sparged into the salt after the preliminary electrolysis can separate  $\text{PuO}_2$  exclusively—see the third step of Figure 14.9. It is unclear to what extent the minor actinides such as Am and Np will be carried along with the Pu.

**14.4.2.3 Electrometallurgical Treatment of EBR-II Metal Fuel**

Although aqueous systems have played the dominant part in fuel reprocessing, nonaqueous electrochemical separations with molten salt electrolytes have proven effective in the United States for recycling actinides in defense-related (primarily plutonium-bearing) materials as well as metallic fuel from the Experimental Breeder Reactor II (EBR-II). Plant designs based on preliminary research and development results indicate that this specific version of pyroprocessing (as will be nominally referred) has the potential to reduce the size of plants and equipment needed (*Nuclear wastes technologies for separations and transmutations*, 1996, 43–44; Wick, 1980, 544–549), and, thus, the cost, as compared with a plant based on aqueous reprocessing.

The interest in such processes stems from their considerable advantages or potential advantages over the more conventional aqueous processes:

- Lower cost, the basis for which is the compactness and simplicity of the process.
- Proliferation resistance, which stems from inherent difficulties in extracting a highly purified plutonium product from electrorefining. Note, however, that the



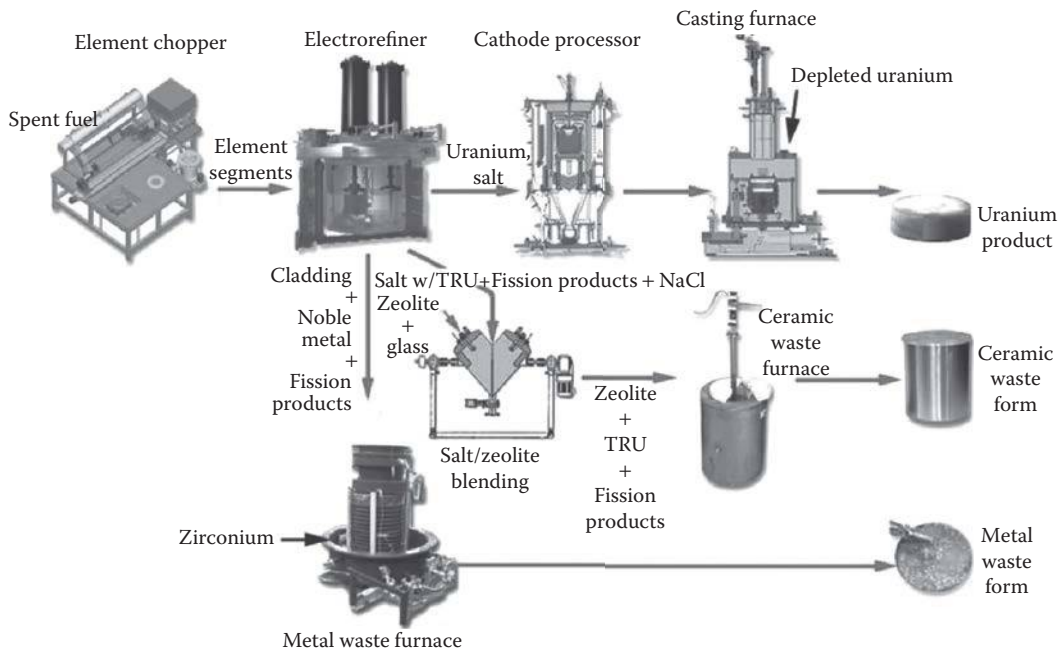
**FIGURE 14.9**  
The RIAR oxide electrowinning process.

US government no longer considers pyroprocessing to be proliferation-resistant. This stems from rethinking the requirements for proliferation and the potential misuse of the process technology. The view that pyroprocessing is not capable of separating pure plutonium from low-grade materials such as spent fuel has not changed.

- Ability to process feed after only a short cooling period because no materials subject to radiation decomposition are present.
- Production of FP waste in a dry, concentrated form more readily stored or processed to recover specific FPs.
- Alleviation of criticality problems in plant design due to the absence of neutron moderating materials with a consequent reduction of costs.

Criticisms of the technology are based on the low processing rate due to batch operations, low purity of recovered products, and potential high cost of constructing and remotely operating high temperature systems in a hot cell environment.

So far, only one spent fuel pyroprocessing technique has been operated on a significant scale in the United States. This is the integral fast reactor (IFR) process developed by Argonne National Laboratory, initially for the demonstration of an integrated fast reactor and fuel recycle system. Later after the 1994 shutdown of the IFR program, the same pyroprocessing technology was demonstrated for conditioning the EBR-II fuel for safe, permanent disposal. An official demonstration project was funded by DOE from 1996 to 1999 (Benedict and Henslee, 1997). A record of decision was later issued by DOE, selecting the electrometallurgical process as the preferred treatment method for the EBR-II fuel. Since that time, Argonne National Laboratory-West, which later changed to INL, has been continuing to process the EBR-II fuel. A diagram showing the electrometallurgical process for treating the EBR-II fuel is shown in Figure 14.10. This process is carried out in the Fuel Conditioning Facility (FCF) at INL. In contrast to the IFR process, this application of the technology is essentially a partitioning-conditioning process because neither plutonium nor other TRU elements are routinely recovered for recycling. Rather, they are allowed to go into the HLW streams. The process is used to facilitate the disposal of a fuel that could not otherwise be sent directly to a geologic repository. The uranium metal fuel is submerged in  $\text{LiCl} + \text{KCl} + \text{UCl}_3$  molten bath and subsequently electrorefined to a steel cathode where it forms a purified U metal in the dendritic form. The cladding hulls, noble metals, and some residual actinides are left in the anode basket. The contents of the anode basket are loaded into a metal waste furnace which serves to both distill surface salt from the cladding hulls and melt the cladding hulls with their contents into a metallic ingot suitable for permanent disposal in an HLW repository. In addition to undissolved actinides, this primarily stainless steel/zirconium waste form also contains the long-lived Tc-99 radionuclide. Meanwhile, the U product from the electrorefiner is transferred to a cathode processor where the surface salt is distilled, and the uranium metal is melted into a solid ingot. That uranium product would be highly enriched if it was produced from the driver fuel. In that case, INL down blends the U product with depleted uranium in a casting furnace to achieve a U-235 enrichment level of less than 20%. As fuel processing continues, the electrorefiner salt becomes more and more contaminated with active metal FPs (Group I, II, and rare earths) and TRUs. Most of these elements form chlorides in the electrorefiner salt. At some point, the salt's contamination



**FIGURE 14.10**  
Electrometallurgical process for treating EBR-II fuel.

level reaches a maximum, and the salt will need to be disposed and replaced with clean  $\text{LiCl-KCl-UCl}_3$ . The process that has been designed and demonstrated for making waste forms from this salt is called the ceramic waste process (see Figure 14.10). In this process, dry zeolite-4A particles are mixed with milled salt particles at high temperature ( $500\text{ }^\circ\text{C}$ ). This results in the absorption of salt into the zeolite-4A pores (Simpson et al., 2001). Then the powder mixture is cooled, and glass frit is blended in with the salt-loaded zeolite. After the glass and zeolite are well mixed, the powder is transferred to a stainless steel canister that is loaded into a ceramic waste furnace and heated to over  $915\text{ }^\circ\text{C}$ . This results in densification, consolidation, and conversion of the zeolite crystals to sodalite. The final waste form is a monolithic ingot of approximately 400 kg consisting of glass-bonded sodalite. The salt content in this waste form is only 7.5 wt%, which is why there is significant interest in developing selective salt separation methods to concentrate the FPs and recycle  $\text{LiCl-KCl}$  to the electrorefiner. But none of these methods have been implemented for the production process, since the total amount of salt to dispose is relatively low. The cost savings from waste volume reduction are seen as insufficient to justify the additional cost for technology development and equipment installation.

Processing of the EBR-II fuel is still ongoing, largely due to the nonoptimized operational aspects of FCF that was never designed as a production facility; rather, it has been an experimental facility since its inception. This combined with variable program funding from the DOE has drawn out the time frame for completing the fuel campaign. The entire inventory of the EBR-II fuel that needed to be processed was about 25 MTHM. To date, about 90% of this fuel remains untreated.

## 14.5 Modern Aqueous Reprocessing

### 14.5.1 Basic Commercial Process: PUREX Process

The PUREX process for reprocessing spent nuclear reactor fuels was developed in the United States and first used at the US Atomic Energy Commission (AEC) Savannah River Site in 1954. It was used next at the US AEC Hanford site in 1956, and soon thereafter achieved worldwide acceptance as the premier fuel-reprocessing scheme.

It still occupies that position today. Even though the PUREX process is universally accepted and used for reprocessing of irradiated nuclear reactor fuel, there is no single PUREX process flowsheet that is universally agreed on. Differences exist among PUREX processes and plants in one country, and from country to country.

Plutonium is handled and processed in spent reactor fuel-reprocessing plants, in MOX ( $\text{PuO}_2$ ,  $\text{UO}_2$ ) fuel fabrication plants, and in research centers where transuranium elements are studied. Large-scale reprocessing of irradiated uranium and uranium-plutonium fuels is performed to recover and purify uranium and plutonium either for further use in nuclear reactors or, in the case of plutonium, in nuclear weapons. The principal uranium and plutonium products are found in Table 14.6.

Most PUREX plants aim at a virtually complete separation of the main streams, with decontamination factors of  $10^6$ – $10^8$ . Very complete separations are necessary to meet many of the product specifications. Specifications for the uranium product are generally designed to enable the uranium to be handled in the same way as natural uranium, and, if it contains U-235, to make it suitable as a feed to a gaseous isotopic enrichment plant. For the plutonium (or uranium/plutonium) product, specifications are designed to permit handling in a glove box with limited gamma shielding, and to restrict the amounts of metal impurities that might affect the properties of the metal or oxide.

As an example, the AREVA La Hague site (Figures 14.11 and 14.12) is a nuclear fuel recycling plant on the French Cotentin Peninsula that provides the first stage of the recycling of UNF taken from nuclear reactors. The facility has the industrial capacity necessary for the annual processing of used fuel from 80 to 100 nuclear reactors, amounting to 1,700 t. This is the leading center of its kind in the world in the processing of UNF. The recovered plutonium is recycled into a mixed oxide (MOX) fuel fabricated at AREVA MELOX facility, according to the plutonium flux adequacy policy, which ensures that no separated plutonium in excess above industrial needs will accumulate (Carre and Delbecq, 2009). About 100 t of MOX fuel are fabricated annually in France and recycled once into twenty-one 900 MW-class PWRs licensed reactors (30% of core) which contributes to about 10% of nuclear

**TABLE 14.6**

Nuclear Fuel Reprocessing: Principal Uranium and Plutonium Products (Schulz, Burger, Navratil, 1990)

Product	Source	Principal Uses
Depleted U (0.2–0.4% U-235)	Spent natural U fuel	Pu production in fast neutron-flux reactors
Slightly enriched U (0.8%–1% U-235)	Spent low-enriched U fuel	Feed to U-235 enrichment plant
Pu	Most types of spent fuel	Reactor Fuel (mixed with U); nuclear weapons



**FIGURE 14.11**

The La Hague Plant Facility on the Normandy coast, France. (Courtesy of AREVA.)

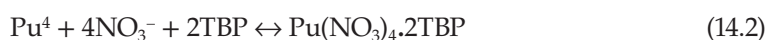
production. Forty-one reactors worldwide or 10% of the world of light water reactors have been loaded with MOX fuel assemblies since the beginning of the 1970s: 36 of these are in Europe (21 in France, 10 in Germany, 3 in Switzerland, and 2 in Belgium), 1 is in the United States, and 4 are in Japan.

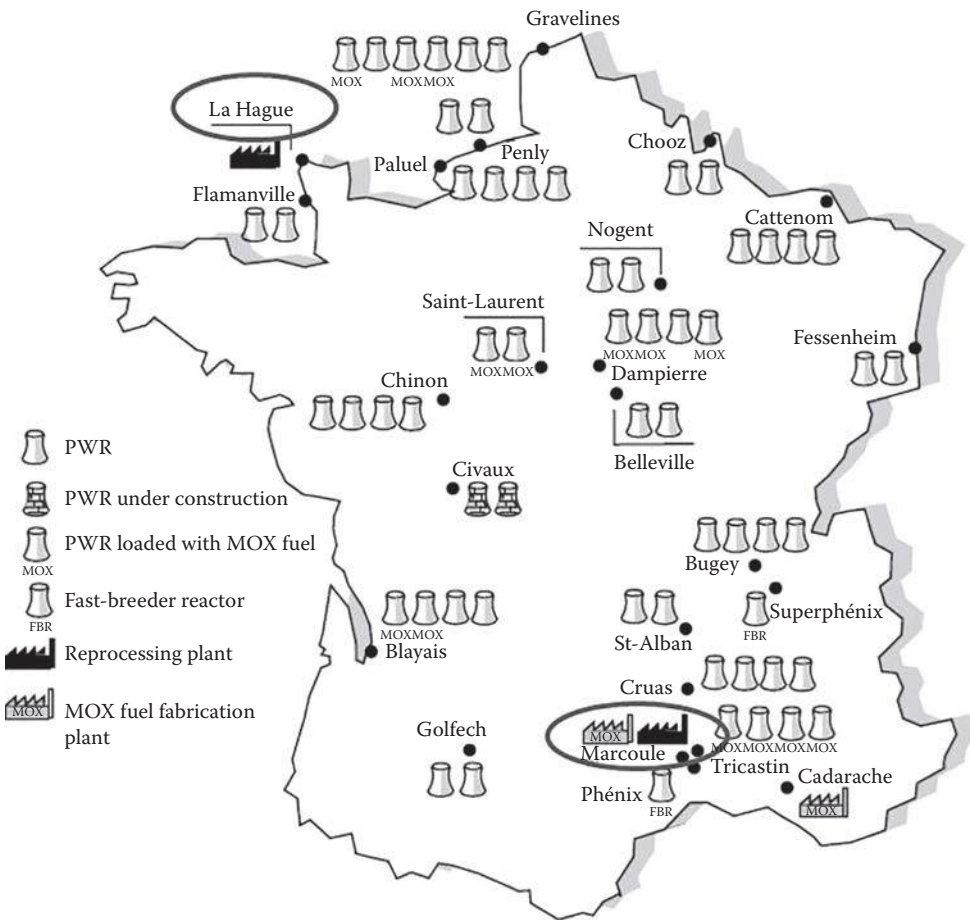
A PUREX plant divides naturally in four sections:

- The head-end which comprises all operations from the receipt of the UNF with its conversion to a feed suitable for solvent extraction, usually by dissolution in nitric acid
- The separation section, which must include means for solvent clean-up and recycling
- Post-separation operations on the product and waste streams
- Off-gas treatment for removal of particulate, NO<sub>x</sub>, and volatile FPs

There are four principal types of solvent extraction cycles in PUREX process operation, as detailed below:

*Codecontamination:* Uranium and plutonium are extracted and stripped together. TBP extraction of uranium(VI) and Pu(IV) and HNO<sub>3</sub> from acidic aqueous nitrate solutions generally proceeds according to the following equations:





**FIGURE 14.12** Map of France with 58 nuclear power plants, one UNF recycling facility located at la Hague site, and one MOX fabrication facility, MELOX plant, located at Marcoule site.

*Partition:* Uranium and plutonium are extracted together, and stripped separately in different contactors.

*Uranium purification:* Uranium from a partition cycle is purified by a further extraction-strip cycle.

*Plutonium purification:* Plutonium is purified by additional extraction cycles.

**14.5.1.1 Head-End Steps: Receipt and Storage, Shearing, Dissolution, Input Accountancy**

Head-end operations in a PUREX plant are determined largely by the characteristics of the used fuel to be recycled. Head-end operations comprise some or all of the following steps:

- Fuel receipt and storage
- Fuel element breakdown/cladding removal
- Fuel dissolution



**FIGURE 14.13**  
Storage of spent fuel in pool, La Hague France. (Courtesy of AREVA.)

- Treatment of off-gases to control volatiles, for example, NO<sub>x</sub>, I-129, I-131
- Treatment of the dissolved fuel solution, for example, clarification, Pu valence and acidity adjustment, prior to first-cycle, TBP extraction operations

#### 14.5.1.1.1 Receipt and Storage

On arrival, the fuel casks need to be unloaded. This can be done under water, or in a dry facility. By remote control, the fuel elements are then put into a storage pond for further cooling. The UNF currently being reprocessed may be stored for several years (Figure 14.13). In fact, because of the use of remote control techniques, the fuel can almost be reprocessed immediately (after one year or so). This logically implies that the “contamination” of the material being put through is higher, which might increase risks.

#### 14.5.1.1.2 Shearing and Dissolution

The fuel elements are chopped into pieces. Using a horizontal shear, the top and the bottom ends, containing no fuel, are chopped off. After that, the fuel rods are cut into slices of about 3 cm (1.3”). Currently, the shearing can be managed remotely and the fuel does not need to be under water. At the French recycling plant, La Hague, a dilute acid nitric solution (1 M HNO<sub>3</sub>) is used at controlled low temperature to remove fuel from cladding. In previous times, chemical procedures were also used to remove cladding material from thermal reactor fuels processed at the Eurochemic plant, Mol, Belgium.

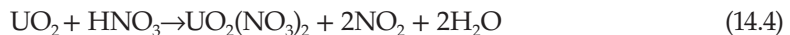
Today, chemical decladding and total dissolution processes are generally avoided because of the difficult waste-disposal problems they engender. The preferred head-end treatment involves the disassembly or dismantling of irradiated fuel assemblies, followed by shearing individual rods. Subsequently, the sheared fuel is leached with boiling nitric acid to produce a dissolver solution suitable for downstream solvent extraction operations.

No matter how prepared, the nitrate-based aqueous dissolver solution usually requires some additional treatment prior to TBP-solvent extraction operations. Such treatment may include any or all of the following:

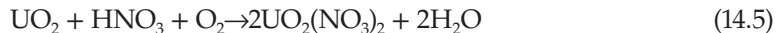
- Cooling
- Clarification
- Adjustment of acidity
- Valence adjustment for Pu

The objectives of fuel dissolution are to: (1) bring uranium and plutonium in the fuel completely into the aqueous solution, (2) complete the separation of fuel from cladding, (3) determine as accurately as possible the amounts of uranium and plutonium charged to reprocessing, and (4) convert uranium and plutonium and FPs into the chemical states most favorable for their subsequent separation (Benedict, Pigford, and Levi, 1981).

The principal reactions taking place in nitric acid are



Ordinarily, both reactions take place to some extent with the first dominant at acid concentrations below 10 M and the second at higher concentrations. In principle, the formation of gaseous reaction products could be avoided by the addition of oxygen directly to the dissolver:



This process is now a fumeless dissolving and is used in European plants. Small amounts of nitrogen oxides and gaseous FPs are also formed.

Plutonium in oxide fuel is dissolved as a mixture of tetravalent and hexavalent plutonyl nitrates, both of which are extractable by TBP. Neptunium dissolves as a mixture of non-extractable pentavalent and extractable hexavalent nitrates.

Most of the FPs go into the aqueous solution. However, at high burn-ups above 30,000 MWd/MT, some elements such as molybdenum, zirconium, ruthenium, rhodium, palladium, and niobium may exceed their solubility limits and be present as solids (Benedict, Pigford, and Levi, 1981).

In the solution, americium and curium, most of the FPs are in a single relatively nonextractable valence state. Iodine and ruthenium are important exceptions. Iodine may appear as nonextractable iodide or iodate or as elemental iodine, which would be extracted by the solvent and react with it. Ruthenium may appear in any valence state between 0 (insoluble metal) and 8 (volatile ruthenium tetroxide), and, at valence 4, it may form a number of nitrosyl ruthenium (Ru(IV)NO) complexes of varying extractability. An important objective of dissolution and the preconditioning of the feed solution prior to extraction is to convert these FP elements into states that will not contaminate uranium, plutonium, or solvent in subsequent solvent extraction (Benedict, Pigford, and Levi, 1981).

Uranium dioxide dissolves more rapidly than plutonium dioxide or thorium dioxide. Irradiated fuel dissolves faster, probably because of cracking during irradiation.



Because extensive foaming results when fuel is added directly to boiling nitric acid, the preferred procedure in the batch dissolver is to add  $\text{UO}_2$  to cold acid, then bring the solution to just below the boiling point, with adequate cooling available to deal with heat evolved from chemical reaction and radioactive decay.

#### 14.5.1.1.3 Separation from Cladding

After the reaction of fuel with acid has been completed, the resulting solution and suspended fine particles are drained from the coarser cladding fragments. The cladding is washed, first with dilute nitric acid and then with water. The cladding is checked by gamma spectroscopy to establish the removal of adherent fuel and then discharged for packaging as radioactive waste. The fuel solution, possibly containing suspended particles, is clarified by centrifugation. Centrifuged solids are accumulated and periodically leached as described beforehand for the recovery of plutonium and uranium (Benedict, Pigford, and Levi, 1981).

#### 14.5.1.1.4 Accountability Measurements

The dissolver solution and washings are collected in a calibrated accountability tank and mixed thoroughly. The volume and density of the solution are measured as accurately as possible, and samples are taken for the determination of uranium and plutonium concentrations. This is the first point of reprocessing at which a quantitative measure of input amounts can be made (Benedict, Pigford, and Levi, 1981).

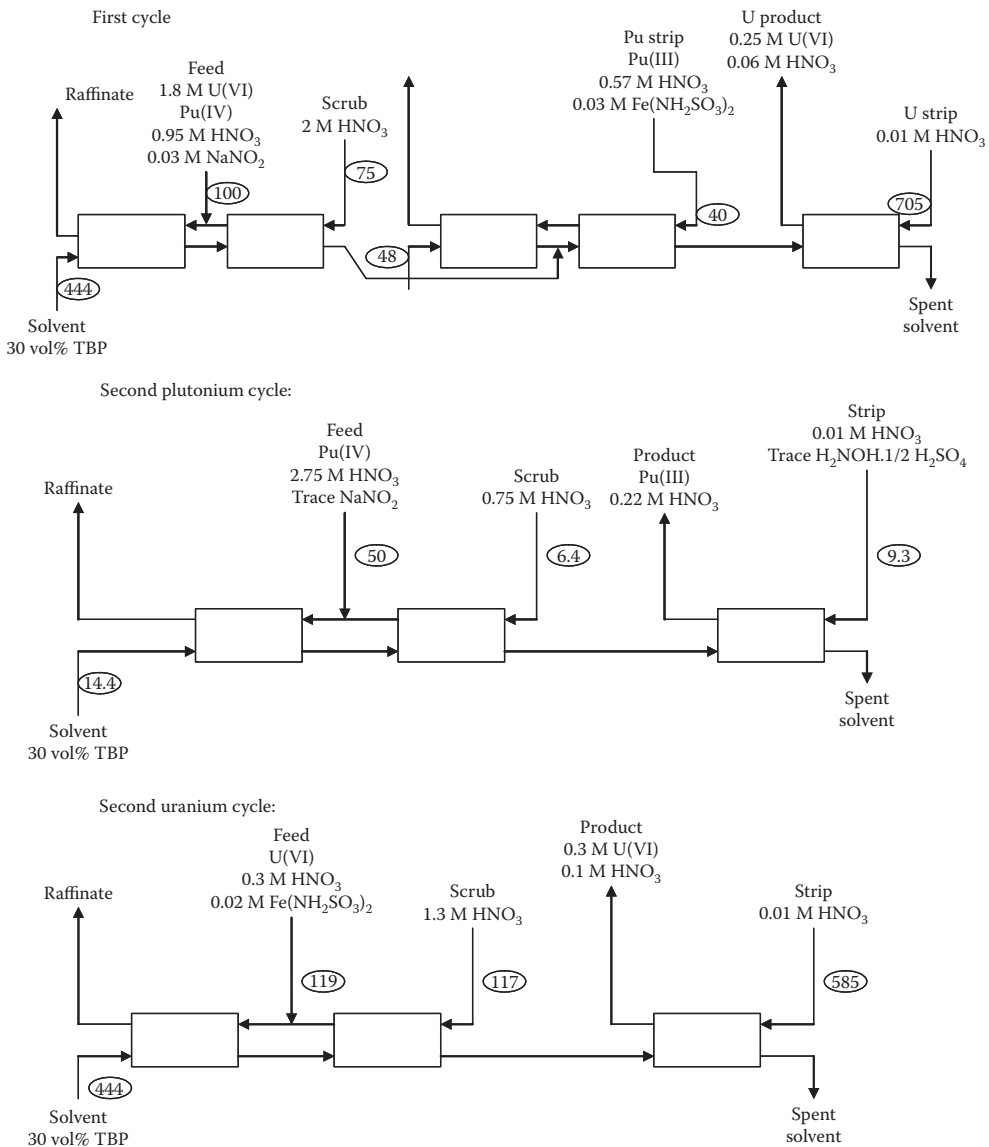
### 14.5.1.2 PUREX Process Description

The steps comprise (1) the extraction of U(VI) and Pu(IV) into the organic phase by a mixture of TBP/diluent, (2) scrubbing of the organic phase with a dilute nitric acid solution (~1 M) to facilitate the removal of impurities—Zr and Ru, for example (Herbst, Baron, and Nilsson, 2011), (3) separation of Pu(IV) from U(VI) by the use of reductant, (4) stripping of U(VI) from the organic phase by water of very dilute nitric acid (~0.01 M), and (5) recycling of the solvent.

#### 14.5.1.2.1 Extraction

Because of several disadvantages of the diethyl ether process including high costs, complexity, and explosion hazards, in 1949 and 1950, scientists and engineers at the US Oak Ridge National Laboratory conducted studies to evaluate various alternative reagents for their ability to extract uranium from aqueous nitrate solutions (Schulz, Navratil, and Bess, 1987, 3, 4). Reagents investigated included methyl isobutyl ether, diethyl ether, and TBP. Tests clearly demonstrated the superiority of TBP over other reagents. The fundamental consideration is, of course, the ability of TBP to extract actinides at valences IV and VI, while leaving most of the FPs, as well as actinides at valences III and V, in the aqueous phase. The typical PUREX plant solvent contains 30 vol/% TBP mixed to 70 vol/% diluent such as dodecane.

The most important TBP process is the PUREX process, in which 20–40 vol/% TBP in the hydrocarbon diluent is used. The flowsheet commonly used in the United States specifies 30 vol/% TBP (Long, 1978). A typical American flowsheet is shown in Figure 14.14. Uranium is extracted from the feed and decontaminated in the first contactor, separated from plutonium in the second contactor, and stripped back into an aqueous phase in the third contactor. A high aqueous phase to organic phase flow ratio



**FIGURE 14.14**

Example of a PUREX low-acid flow sheet. O indicates relative flow rate by volume. (Data from Long, J.T. 1978. *Engineering for Nuclear Fuel Reprocessing*, Nuclear Science and Technology Series. La Grange Park, IL: American Nuclear Society, pp. 236–242.)

is necessary in the stripping section to maintain an extraction factor much less than 1.0. The uranium product from the first cycle is accordingly quite dilute and must be concentrated (by evaporation) before it is fed to the second uranium cycle. The second cycle uranium feed is salted with additional nitric acid and then extracted, scrubbed, and stripped as before.

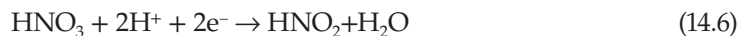
Plutonium is extracted and decontaminated along with uranium in the first extraction–scrub contactor.

#### 14.5.1.2.2 Partitioning of U(VI) from Pu(IV)

Once plutonium and uranium are coextracted and codecontaminated, plutonium is separated from uranium in the partitioning contactor by reduction to Pu(III) with a reductant. Over the years, a number of plutonium reductants have been proposed. The most widely used reductant to partition plutonium from uranium in the PUREX process was  $(\text{Fe}(\text{SO}_3\text{NH}_2)_2)$ ; other alternates were proposed such as hydrazine-stabilized ferrous nitrate or uranous nitrate, and hydroxylamine salts.

*14.5.1.2.2.1 Ferrous Sulfamate as Pu(IV) Reductant* In the United States, one of the first chemicals to be used as plutonium reductant for the partitioning of plutonium and uranium in PUREX-solvent extraction systems was ferrous sulfamate  $(\text{Fe}(\text{SO}_3\text{NH}_2)_2)$  which was effective because Fe(II) rapidly reduced Pu(IV) to Pu(III) and because the sulfamate ion stabilized Pu(III) (Walser, 1970). Unfortunately, several disadvantages (Mc Kibben and Bercaw, 1971) appeared with its use and the generation of sulfates by the sulfamate which (1) contributed to plutonium losses (via aqueous plutonium sulfate complexes), (2) increased the volume of radioactive wastes to be stored, (3) accelerated the stainless steel process equipment corrosion, and (4) enhanced the complexity of the development of waste forms for immobilizing these effluents because the large amount of sulfate in the waste can cause problems in the formulation of glass waste forms (Mc Kibben and Bercaw, 1971; Schlea et al., 1963).

*14.5.1.2.2.2 Ferrous Nitrate as Pu(IV) Reductant* Alternative reductants were considered: ferrous nitrate, stabilized by hydrazine or uranous nitrate also stabilized by hydrazine. The mixture of ferrous nitrate and hydrazine (0.03M  $\text{Fe}(\text{NO}_3)_2$  – 0.075M  $\text{N}_2\text{H}_4$ ) were tested in the late 1960s at the Hanford plant (Walser, 1970). Even though ferrous nitrate added iron to the waste, it avoided the issue caused by the presence of sulfate ions. On the other hand, using ferrous sulfamate did not require the use of a stabilizer such as hydrazine because sulfamate served this purpose. Using ferrous nitrate required the use of an auxiliary reductant like hydrazine. The principal function of the holding reductant hydrazine was to serve to supplement the sulfamate ions as a scavenger for nitrous acid (Kessinger et al., 2009). In the PUREX process, nitric acid is the aqueous medium where all the reprocessing operations are taking place. It is known that nitric acid is easily reduced to nitrous acid according to the following reaction (Equation 14.1):



In the PUREX process, the oxidizing property of nitric acid and the formation of nitrous acid are not favorable to maintain plutonium as a trivalent species. The sufficient amount of hydrazine is added to the system to ensure the stability of Pu(III) with the destruction of nitrous acid according to Equation 14.7. Despite the fact that hydrazine is particularly advantageous because the reaction products,  $\text{N}_2$ ,  $\text{N}_2\text{O}$ , and  $\text{H}_2\text{O}$ , do not contribute to the volume of stored wastes (Schlea et al., 1963), the interaction of hydrazine and nitrous acid can initiate, in a TBP-nitric acid system, and under specific operating conditions, the formation of hydrazoic acid ( $\text{HN}_3$ ) which is a hazardous and potentially explosive compound (Equation 14.7) (Dukes and Wallace, 1962). Further oxidation leads to the formation of nitrous oxide and nitrogen gases as depicted in Equation 14.8



followed by



Hydrazoic acid is an unstable compound: the pure chemical form can be exploded when heated or shocked. Its relatively low boiling point ( $T = 35.7\text{ }^{\circ}\text{C}$  under atmospheric pressure) makes it volatile at room temperature, which has considerable impact on safety. Scientists at the Savannah River Site investigated the hazards of hydrazoic acid in connection with the use of hydrazine as a holding reductant and uranous nitrate as the partitioning reagent (Horner, 1969). In their study, they showed that hydrazoic acid never reached the critical explosive concentration (4.7 M in the aqueous solution) in their process because dilute solutions ( $<0.2\text{ M}$ ) of hydrazine were used. By analogy, there was no hazard associated with the use of ferrous nitrate—hydrazine solutions because the maximum hydrazine concentration was 0.2 M. However, the extraction of hydrazoic acid into the organic phase may result in the formation of sodium azide in the alkaline-solvent wash solutions which use sodium bicarbonate to recycle TBP. Thus, it was recommended to destroy hydrazoic acid by treatment with sodium nitrite before being discharged into waste tanks.

*14.5.1.2.2.3 Uranous Nitrate as Pu(IV) Reductant* Uranous nitrate was proposed as an alternative reductant and had the advantage of adding no extraneous ions to the system but was more difficult to apply than ferrous nitrate (Schlea et al., 1963). Uranium nitrate can act as an effective reducing agent only if it is stabilized by a chemical, such as hydrazine, which as mentioned previously, will rapidly and completely destroy the nitrous acid. In acidic solutions, U(IV) is a strong enough reducing agent to reduce Pu(IV) to Pu(III) (Equation 14.9), and the reaction is very rapid:



Furthermore, uranous nitrate is being successfully used at the Rokkasho Mura reprocessing plant in Japan to separate uranium from plutonium. One of the advantages of U(IV) is that it does not introduce nonvolatile or corrosive constituents as compared with ferrous sulfamate or ferrous nitrate.

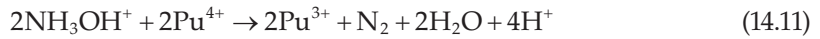
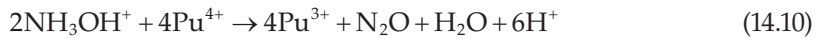
*14.5.1.2.2.4 Hydroxylamine Nitrate as Pu(IV) Reductant* The use of hydroxylamine nitrate (HAN)–hydrazine mixture as reductants to obtain plutonium and neptunium decontamination in the second uranium cycle of a PUREX-based process was successfully demonstrated by Walser in 1970. Walser showed that in contrast to the test of hydrazine stabilized U(IV) where about 60% of the added hydrazine was converted to hydrazoic acid, only about 5% of the hydrazine was converted to hydrazoic acid in this particular test (Table 14.7). Moreover, HAN has a number of advantages because it is nonmetallic, readily decomposed to innocuous products by heating. Following this observation and Barney's investigations (Barney, 1975;1976), the routine use of HNA–hydrazine in the second uranium cycle during uranium–plutonium processing was recommended. In addition, this change reduced the process additions of both iron and sulfate by about 30%.

**TABLE 14.7**

Comparison of Mixtures of Reductants for the Conversion of Hydrazine to Hydrazoic Acid

Mixture of Reductants	HN <sub>3</sub> Production	Reference
U(IV): Hydrazine	60% of N <sub>2</sub> H <sub>4</sub> converted to HN <sub>3</sub>	Walser 1970
Ferrous Sulfamate: Hydrazine	25% of N <sub>2</sub> H <sub>4</sub> converted to HN <sub>3</sub>	Dukes and Wallace 1962
Hydroxylamine: Hydrazine	5% of N <sub>2</sub> H <sub>4</sub> converted to NH <sub>3</sub>	Walser 1970

Hydrazine is also used in conjunction with HNA to impede hydroxylamine oxidation by nitrous acid, always present in nitric acid media, which increases the HNA availability for the plutonium reduction. Both hydrazine and HNA interact with nitrous acid ( $\text{HNO}_2$ ), but the hydrazine oxidation is much faster. On the other hand, the use of HAN in conjunction with nitric acid introduces the possibility of an autocatalytic reaction resulting in the overpressurization of the system or explosion in a reprocessing facility as pointed out in Barney's report (Barney, 1998). The main function of HNA is to reduce plutonium from the tetravalent state to the trivalent state and, thus, separate the plutonium from the uranium, which is retained in the hexavalent oxidation state and, hence, in the organic phase. The reduction reaction by HNA is the result of two irreversible reactions (Equations 14.10 and 14.11):



Reaction (14.11) occurs predominantly in the presence of an excess of HNA, while reaction (14.10) dominates for an excess of plutonium (Paviet-Hartmann and Senentz, 2007). Equation 14.12 is used to calculate the reduction rate of Pu(IV) by HNA:

$$-\frac{\partial[\text{Pu(IV)}]}{\partial t} = k_{\text{Barney}} \frac{[\text{NH}_3\text{OH}^+]^2}{[\text{H}^+]^4 (K_d + [\text{NO}_3^-])^2} \frac{[\text{Pu(IV)}]^2}{[\text{Pu(III)}]^2} \quad (14.12)$$

where  $k_{\text{Barney}} = k_4 K_1^2 K_h^2 K_d^2 K_a^2$  is the rate constant as established by Barney (1976).

HNA was incorporated into many nuclear fuel reprocessing plants in the early 1970s replacing the ferrous sulfamate and hydroxylamine sulfate for plutonium reduction because it possessed the proper Pu(IV) to Pu(III) reduction attributes and the gaseous reaction products:  $\text{N}_2$ ,  $\text{N}_2\text{O}$ , and water contributed to the minimization of the volume of solid wastes produced. The French PUREX process at the La Hague site safely uses a mixture of HNA and nitric acid for the reductive stripping phase of plutonium. The British also used HNA in the thermal oxide reprocessing plant (THORP) for over several years (Barney, 1998).

#### 14.5.1.2.3 Purification

The plutonium-stripping stream is salted with sufficient nitric acid to keep the uranium almost quantitatively in the organic phase.

Because PUREX does not need process salting to work efficiently and because solvent and aqueous reagent streams can be reclaimed for recycle, the volumes of waste fluids are greatly reduced. The FP fractions can be discharged in the concentrated form to holding tanks for the radioactive decay prior to final vitrification and disposal. The process system works very reliably and produces a plutonium product with a decontamination factor of FPs  $>10^7$ . The uranium fraction has a similar decontamination factor.

In the extraction step of each type of cycle, impurities remain behind in the aqueous phase. Typically, three extraction-strip cycles of decontamination are required to obtain sufficiently pure plutonium, while sufficiently pure uranium may be obtained in only two extraction-strip cycles.

#### 14.5.1.2.4 Stripping of U(VI)

Prior to the solvent recycling step, U(VI) is stripped from the organic phase by water or very diluted nitric acid solution (~0.01 M). The back-extraction of U(VI) is performed at elevated temperature (~50 °C), and the aqueous-to-organic phase ration (O/A) is slightly higher than unity to help maintain a large concentration profile between phases.

#### 14.5.1.2.5 Solvent Clean-Up

Over time, chemical and radiolytic processes cause TBP and diluent to degrade (Wright and Paviet-Hartmann, 2010), resulting in lower solvent performance. Some of the degradation products including the major ones like di-butylphosphoric acid (HDBP) and mono-butylphosphoric acid (H<sub>2</sub>MBP), form strong complexes with the metals present in reprocessing streams, which lead to the reduced decontamination of products, losses of uranium and plutonium products and contaminated solvents (Hardy and Scargill, 1959; Stieglitz and Becker, 1983). Studies have found that they each form large polymers as well as complexes with several metal ions including those of plutonium, uranium, and zirconium (Stieglitz and Becker, 1983). Also, in the past, decomposition products formed colloidal suspensions in the organic phase and sediments at the phase boundary which decreased the efficiency of the solvent extraction (IAEA, 1998); modern recycling plants do not observe these phenomena at all, thanks to the use of a combination of the alkaline washing and the organic waste treatment unit, which allows the elimination of most solvent degradation products prior to recycling TBP and diluent into the plant (Alcock et al., 1956; Manohar et al., 1999). A necessary adjunct to the separation section of a solvent cleanup operation enables the TBP solvent to be routinely recycled. The standard reagent is a dilute sodium carbonate (Na<sub>2</sub>CO<sub>3</sub>) solution, which removes acidic degradation products of TBP and diluent. Carbonate ion forms very strong, aqueous soluble complexes with plutonium. In some cases, the Na<sub>2</sub>CO<sub>3</sub> wash may be followed by aqueous sodium hydroxide and/or nitric acid washes. The spent solvent wash liquors contain substantial amounts of sodium and are appreciably radioactive.

#### 14.5.1.2.6 Post-Separation Operations

There are two types of HLW: FPs and TRU elements separated from UNF, and UNF elements themselves from the reactor core when they are not reprocessed. Both types of HLW must be treated prior to disposal. HLW from reprocessing is incorporated into solid blocks of borosilicate glass, in a process known as vitrification. For direct disposal, used fuel assemblies require encapsulation in containers made, for example, of stainless steel or copper.

In reprocessing, when the FPs are first extracted from UNF, they are in the liquid form, having been dissolved in acid (usually nitric acid). This liquid can be safely retained in stainless steel tanks that are equipped with cooling systems until it is converted into a solid, which is a more convenient material for management, storage, transport, and disposal. After drying, it is incorporated into molten borosilicate glass which is allowed to solidify inside a stainless steel canister.

Vitrification produces a stable solid that has the HLW incorporated its structure. A year's waste from a 1000 MWe reactor is contained in 5 t of such glass, or about 12 canisters 1.3 m high and 0.4 m in diameter. These can be readily transported and stored, with appropriate shielding.

In either case, however, there is a cooling period of about 50 years between removal from the reactor and disposal, with the conditioned UNF or conditioned HLW being retained

in interim storage. This is because the level of radioactivity and heat from the used fuel fall rapidly in the first few years, and are down to about 1000th of the level at discharge by 40 years. Such long-term storage facilities may be at one central place as in Sweden or at the reactor site, as in the United States. UNF may be stored under water in a pool or in dry cask storage, where circulating air removes the heat generated by UNF. The structure and design of both the building and containers protect the outside world from radiation exposure and the fuel from potential outside hazards.

#### 14.5.1.2.7 End Products

The contractual end product for uranium is the nitrate solution. If a utility wants to re-use it, it has to be converted to the hexavalent form or to the oxide form, which can be used in MOX fuels, but, in general, it is more attractive to use natural or depleted uranium. The plutonium cannot be economically stored in the nitrate form because critical masses or concentrations in aqueous solutions can become very small. For this reason, the plutonium solution is converted to oxide powder by treating it with a hot oxalic acid, causing the plutonium to precipitate as Pu(IV) oxalate. When dried, it is calcined while modestly annealed, the product being Pu(IV) oxide ( $\text{PuO}_2$ ). This is weighed (with a typical uncertainty of 0.1%) and put in storage containers.

### 14.5.2 Waste Generation and Processing

Because of the great variety of aqueous waste streams and differences in process flow arrangements in different plants, there is no standard flowsheet for processing aqueous wastes from the PUREX process. Figure 14.15 depicts the principal steps in one possible scheme for concentrating the wastes and recovering water and nitric acid from them. Low-level, low-acid, low-salt wastes are neutralized, if necessary, and concentrated in a simple flash or vapor-compression evaporator to produce LLW concentrates and waste sufficiently decontaminated for return to process.

The intermediate level-waste concentrator handles the LLW concentrate, contaminated aqueous solutions from solvent washing, and many other streams with appreciable solid content. With more exhaustive entrainment removal, as by the partial reflux of condensate through a bubble-plate or sieve-plate column, water sufficiently pure for return to process can be produced. If concentrator bottoms are concentrated to the point of incipient crystallization, they are routed to waste storage. If still unsaturated, they are routed to the HLW concentrator.

The principal feed to the HLW concentrator is the HLW stream from the contamination solvent extraction cycle. This typically contains about 2.5 M  $\text{HNO}_3$ , 3–9 g FPs  $\text{L}^{-1}$ , and 400–1200 Ci  $\text{L}^{-1}$ , and generates heat at the rate of 2.6 W  $\text{L}^{-1}$ . Additional feed may be intermediate-level waste concentrate and nitric acid evaporator bottoms. The HLW concentrator is usually operated at subatmospheric pressure and made of corrosion-resistant material such as titanium to extend life and minimize maintenance. Wastes are concentrated as far as possible without appreciable solid formation. If solids other than FPs are absent, a concentration of about 90 g FPs per liter can be obtained. Products are contaminated nitric acid overhead, slightly under 2.5 M, and evaporator bottoms, about 7 M in  $\text{HNO}_3$ . Because evaporator bottoms self-heat at a rate up to 1 °C  $\text{mn}^{-1}$ , the evaporator and the bottoms storage tanks must be provided with reliable cooling (Benedict, Pigford, and Levi, 1981, 489).





The degradation of TBP may primarily form butanol and butyl nitrate. The butanol formed via the hydrolysis of TBP, monobutyl phosphate ( $H_2MBP$ ) and dibutyl phosphate (HDBP), may subsequently undergo nitration to form butyl nitrate. Alternatively, butanol may undergo oxidation to yield carboxylic acids and gases (carbon dioxide and nitrogen monoxide).

The degradation of TBP and subsequent exothermic oxidation of the associated by-products by nitric acid introduces the risk of a runaway reaction and related over-pressurization and/or gas explosion event due to the property of the reactions and the off-gases produced (Mc Kibben, 1976; Agnew, 1994; Barney and Cooper, 1995; Campbell and Mailen, 1988; James and Sheppard, 1991; Rudisill, 2000). As early as 1934, the exothermic reaction of organic compounds and nitric acid and the formation of gaseous products from the nitration of various paraffins were studied. Laboratory studies of TBP–nitrate thermal reactions have been conducted at a number of DOE sites over the past 40 years, primarily at Hanford and Savannah River. These experiments have characterized the reactions of TBP and its adducts with metal nitrates and nitric acid. Furthermore, these experimental studies have shown that the behavior of such systems depends on the temperature, composition of the system, degree of confinement, and heat transfer afforded to the system (Colven, Nichols, and Siddall, 1953; Nichols, 1960).

Numerous attempts have been made to quantify the probability of having TBP runaway reactions. However, these assessments are usually inconsistent with the historical record of TBP accidents (Usachev and Markov, 2003; LANL DOE Report, 1993). In fact, during this time period, six events have been reported (Barney and Cooper, 1994).

In a broad sense, red oil is any unstable organic-based material that forms or accumulates during an operation at high temperature and progresses, as the temperature increases, to a decomposition reaction yielding heat and gaseous products (Gordon, O'Dell, and Watkin, 1994). The material may be degraded organic products, such as nitrated, oxidized diluent or compounds consisting of butyl groups from TBP or, an organic-nitrate complex, such as plutonium adduct. Examinations of the residue remaining after the incidents identified a red-colored organic phase containing uranium, TBP, nitrate, and organic material derived from the diluent. The red color did more likely result from the nitration of relatively unstable constituents in the diluent. These diluent degradation products may have also contributed to the event sequence. Once the decomposition started, the temperature had increased the major reactants were, however, probably TBP adducts and associated by-products from the hydrolysis reaction.

#### 14.5.2.1.2 Off-Gas Treatment

The main release of gaseous radionuclides occurs when UNF is breached and dissolved in boiling nitric acid. The following radionuclides iodine-129 (I-129), and krypton-85 (Kr-85) are likely to appear in the off-gas of an air sparge of the dissolver in a conventional PUREX fuel-reprocessing flowsheet. Iodine in the fuel element is believed to occur mainly as iodide ( $I^-$ ); however, when contacted with nitric acid, the iodide rapidly oxidizes to elemental  $I_2$ . Much of the elemental iodine is volatilized to the off-gas, but a portion remains dissolved in solution; this ratio (gas phase/liquid phase) is the distribution coefficient or sometimes expressed as a Henry's Law constant. Further oxidation of the liquid phase iodine results in the formation of the nonvolatile iodate ion,  $IO_3^-$ . However, the formation of the iodate ion occurs slowly and reduction back to elemental iodine is promoted by nitrous acid, which is present in the dissolver system (McKay, Miquel, and White, 1982).

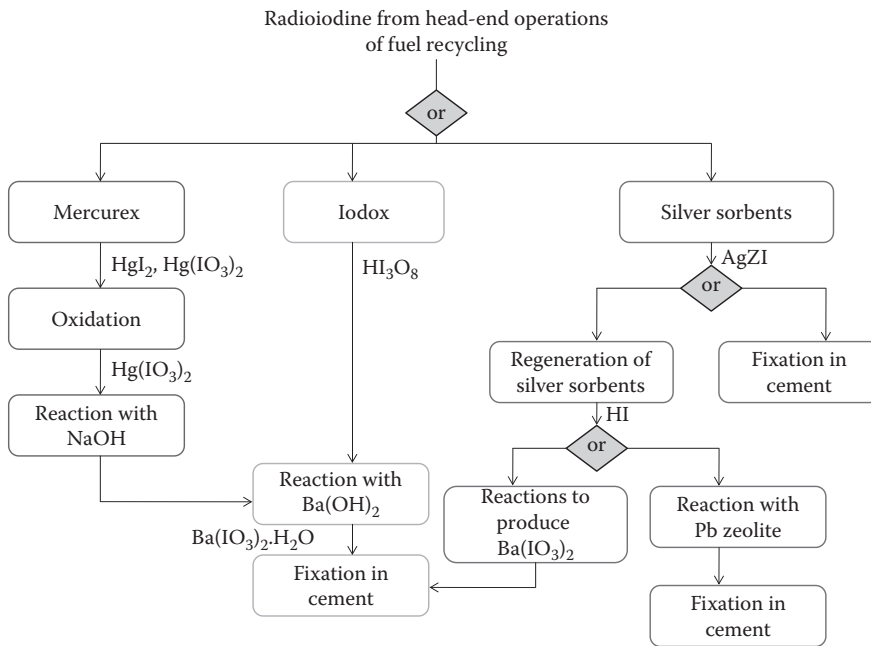
Tritium (H-3) and carbon-14 (C-14) are also volatile radionuclides but H-3 has a short half-life (12.3 yr) and its low beta decay energy (18.5 keV) makes it a relatively mild

radiological hazard. No measures are currently in place to immobilize the relatively small quantities of H-3 arising from fuel reprocessing plants, it is commonly assumed that the discharge of H-3 to the ocean, with subsequent isotopic dilution, is an adequate management mode at coastal sites (Taylor, 1991). C-14 is also only weakly radioactive (half-life, 5,730 yr, beta decay energy, 156 keV) and is not abundant in fuel reprocessing plants. Like H-3, it is not currently subject to control measures at fuel reprocessing plants and the requirement for separation and immobilization is marginal. The off-gases usually contain nitrogen (~80%), oxygen (0–20%), xenon (~500 ppm), krypton (~70 ppm) and various impurities such as NO<sub>x</sub> (NO, NO<sub>2</sub>, N<sub>2</sub>O) gas, steam and traces of hydrogen, carbon dioxide, nitric acid, iodine and hydrocarbons (Lennemann, 1976). Gaseous radionuclides which may become a problem in the future are tritium, krypton-85 and possibly iodine-129. While the capacity of the environment with respect to tritium, krypton-85, and iodine-129 is not well established, it has been estimated that the global concentrations of tritium and krypton-85 may approach acceptable limits during the first half of this century (Paviet-Hartmann, Kerlin, and Bakhtiar, 2010).

*14.5.2.1.2.1 Concentration and Recovery of Iodine* I-129 is produced by both natural and man-made sources. The natural sources include the spontaneous fission of uranium and the interaction of cosmic rays with xenon in the atmosphere. The annual production through these natural paths is approximately 10 mg yr<sup>-1</sup>. Up to the year 2000, the European reprocessing plants emitted about 3,500 kg (23 TBq) I-129, of this 70% had been released by La Hague reprocessing plant (IPSN, 1999, Michel et al., 2002). A single light water reactor (LWR) will produce 234 g of I-129 based on uranium burn-up of 30,000 MWd/t (Jubin, 1981). In handling UNE, a single 5 t day<sup>-1</sup> nuclear fuel recycling facility would have to process  $3.2 \times 10^5$  g of I-129 per year (Jubin, 1981). The main factors steering the development of processes for the collection and isolation of I-129 are its long half-life (15.5 Ma), its low concentration standard prescribed by 10 CFR 20, and its tendency to concentrate in the thyroid gland. An additional impetus for the development of radioiodine capture is the release limit for I-129, prescribed by EPA criteria in 40 CFR 190, which requires a high efficiency of collection and retention (Trevorrow et al., 1983). Extensive work in the area of iodine removal from reprocessing plant off-gas streams using various types of solid sorbent materials has been conducted worldwide over the past three decades (Jubin, 1988). More recently, the use of silver-containing sorbents has been the subject of considerable research (Haefner and Tranter, 2007). The most recent work in the United States has addressed the use of silver-exchanged mordenite-type zeolites (Chapman, Chupas, and Nenoff, 2010). In the industrial PUREX process, a small fraction of the total amount of I-129 are released into the shear off-gas as elemental iodine (I<sub>2</sub>); however, organic iodides are produced by reactions with organic contaminants in the dissolver off-gas (DOG) and vessel off-gas (VOG) systems (Hermann et al., 1996). The predominant compound formed by these restrictions is methyl iodide (CH<sub>3</sub>I), which may amount to several percentages of the total iodine in the DOG and VOG systems. In the current PUREX process, the radioiodines I-129 and I-131 can be eliminated from gaseous waste streams by countercurrent scrubbing of the gas with aqueous solutions of caustic, mercuric nitrate, nitric acid, or by chemical adsorption on zeolites treated with silver or on other metals with an affinity for iodine (Lennemann, 1976). If voloxidation or thermal outgassing is employed, a major fraction of the iodine might appear in the off-gas of these operations. Voloxidation is a process where the fuel is size-reduced by chopping or cutting, and then exposed to a high-temperature atmosphere containing air, oxygen, ozone, or some combination of these gases. Voloxidation promotes the conversion of UO<sub>2</sub> to U<sub>3</sub>O<sub>8</sub> and results in the breakdown of the lattice structure of the

fuel and release of gaseous FPs. Tritium is effectively removed in this manner, but other gases, including I-129, have varying levels of removal based on the particular process conditions (McKay, Miquel, and White, 1982).

If neither voloxidation nor thermal outgassing operations are used, most of the iodine present in UNF will proceed into the dissolver. The complex chemistry of iodine poses difficulties in its collection because iodine tends to distribute into several fuel-reprocessing streams. Without voloxidation or thermal outgassing, most of the iodine is expected to appear in the DOG. During the past three decades, various systems have been studied to reduce the iodine release to the environment. Three liquid scrubber systems have been investigated: the Iodox system, the mercuric nitrate–nitric acid (Mercurex) system, and caustic scrubbers. Solid sorbents have also been studied, either as secondary systems to provide final filtering following the use of the liquid-scrubbing techniques or as primary systems to replace the liquid methods. These solid sorbents include silver faujasite-type zeolite (but it has been demonstrated that faujasite is adversely affected by water and nitric vapors), silver mordenite, alumina silicates (but there is a lack of acid resistance), and macroporous resins. Figure 14.16 summarizes the different separation technologies of I-129 from reprocessing/recycling plant. Table 14.8 summarizes the wet-scrubbing techniques available for iodine capture. Mercurex, Iodox, and fluorocarbon absorption have the potential for the highest DFs, and remove elemental as well as organic iodine. All wet-scrubbing techniques suffer from collecting the iodine into a liquid solvent, which must undergo additional processing prior to permanent disposal of the iodine. This could entail one or combinations of several processes including: direct stabilization (grout, low-melting glass, ceramic), chemical conversion of the iodine, or separation and placement on a preferred matrix material.



**FIGURE 14.16** Radioiodine separation technologies from reprocessing/recycling plant. (Trevorrow L.E. et al., *Compatibility of Technologies with Regulations in the Waste Management of H-3, I-129, C-14, and Kr-85*, Parts I and II, ANL-83-57)

**TABLE 14.8**

Summary of Wet Scrubbing Methods for I-129 Capture

Process	Scrub Solution	Decontamination Factor for I <sub>2</sub>	Decontamination Factor for Organic Iodine	State of Development	Comments
Caustic scrub	1–2 M NaOH	10–100	~1	Deployed at full scale	Essentially no organic iodine is retained, KOH may be used if carbonate precipitation is problematic
Mercurex	0.2–0.4 M Hg(NO <sub>3</sub> ) <sub>2</sub> 14 M HNO <sub>3</sub>	10 <sup>2</sup> –10 <sup>4</sup>	10 <sup>2</sup> –10 <sup>4</sup>	Pilot scale	Mercurex process was designed for NFSWestValley but was never implemented
Iodox	20–23 M HNO <sub>3</sub>	>10 <sup>4</sup>	>10 <sup>4</sup>	Pilot scale	Application limited to laboratory and engineering scale
Electrolytic scrubbing	8–12 M HNO <sub>3</sub> 0.1 M Co <sup>3+</sup>	600	100	Lab scale	Presence of 1% NO <sub>2</sub> will reduce DF by a factor of 10
Fluorocarbon	CCl <sub>2</sub> F <sub>2</sub> (Freon R-12)	10 <sup>4</sup>	10 <sup>4</sup>	Pilot scale	Absorption operated at 300 psig, –25 to +10°F
Silicon-organic liquid	Polymethylsiloxane	150	~1	Lab scale	Iodine retained in elemental form, organic iodine not absorbed

Source: Mailen and Hoener, 1976; Stephenson and Eby, 1976; Toth, Bell, and Fuller, 1978; Holladay, 1979; Nakhutin et al., 1980; Mc Kay, 1982; IAEA, 1987.

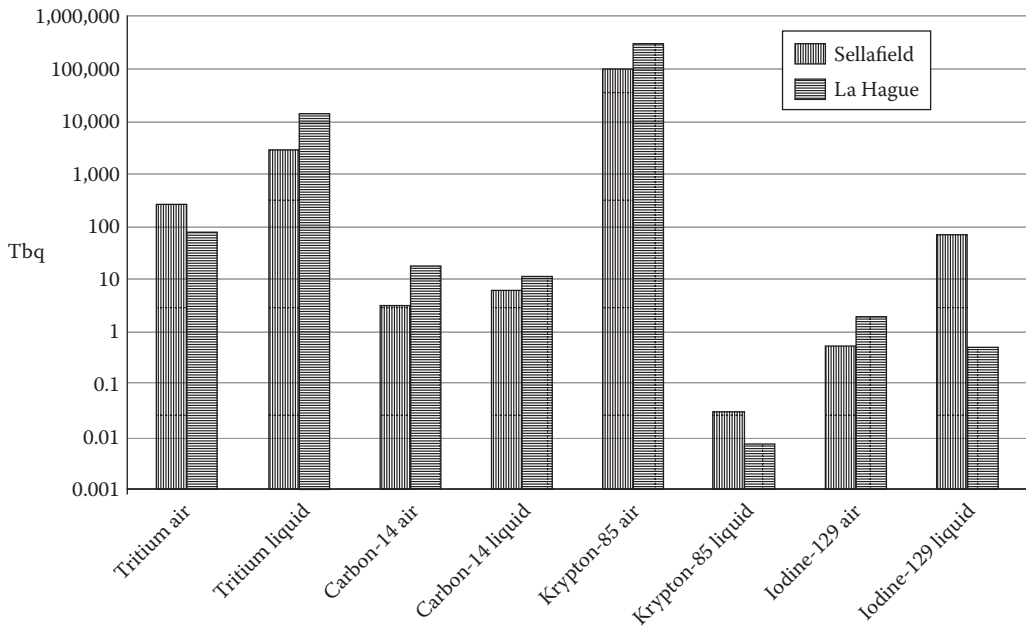
Activated carbons have also been examined as a reference material. However, activated carbon cannot be considered as a primary sorbent for treating reprocessing plant off-gas because of its low ignition temperature and its adverse reactions with nitrogen oxide which could lead to the formation of explosive compounds and to poisoning by organic contaminants in the off-gas (Hermann et al., 1996; Jubin, 1981, 1988).

*14.5.2.1.2.2 Iodex Process and Products* The Iodex process (Jubin, 1988) involves passing the DOG or VOG streams through 20 to 22 M  $\text{HNO}_3$  in a bubble cap column. The iodine product is obtained by evaporating the nitric acid from the liquid effluent from the bottom of the bubble cap column. This results in a solid iodine product. This solid product is, however, water soluble and must undergo further treatment prior to final disposal. The recommended conversion step converts the soluble iodic acid into insoluble  $\text{Ba}(\text{IO}_3)_2$  by simple reaction with  $\text{Ba}(\text{OH})_2$ . Excess  $\text{Ba}(\text{OH})_2$  should be used to ensure complete reaction and does not affect the cement stability. Up to 9 wt% of iodine can be incorporated in a cement matrix as barium iodate. About 3 m<sup>3</sup> of cement would be required to immobilize 340 kg of halogen FPs.

*14.5.2.1.2.3 Mercurex Process and Products* In the Mercurex process, the iodine is absorbed in a  $\text{Hg}(\text{NO}_3)_2$ - $\text{HNO}_3$  solution and converted to iodates and mercury complexes. The scrub solution is 8–12 M  $\text{HNO}_3$  and 0.2–0.4 M  $\text{Hg}(\text{NO}_3)_2$ . The used Mercurex solution loaded with iodine is treated in a subsequent process in which iodine and mercury are separated. After separation, mercury can be recycled to the Mercurex process while iodine is precipitated and conditioned for storage (Paviet-Hartmann, Kerlin, and Bakhtiar, 2010). An alternate process involves contacting process solutions with caustic soda to produce  $\text{NaIO}_3$  followed by conversion to  $\text{Ba}(\text{IO}_3)_2$  as described for the Iodex process (Jubin, 1988). The Mercurex process followed by adsorption on silvered products was retained by SCK/CEN (Belgium) for investigation on both organic and inorganic retention of iodine compounds in recycling plant off-gases in the late 1980s.

*14.5.2.1.2.4 Caustic Scrubbing* Caustic scrubbing involves the conversion of elemental iodine to iodide and iodate. Organic iodine compounds pass through this type of scrubber virtually unreacted.  $\text{CO}_2$  and  $\text{NO}$  react in the scrubber to form carbonates and nitrates. The caustic scrubbing solutions are generally recirculated through the scrubbing tower with a small makeup stream of fresh caustic to maintain acceptable caustic and carbonate concentrations. A side stream is also generally withdrawn which contains the iodides and iodates. The iodine containing species are, for the most part, only minor constituents, and the process is generally limited by the solubility of  $\text{Na}_2\text{CO}_3$  if  $\text{NaOH}$  is used. Variations to the process have been proposed to overcome the problems of sodium by the replacement of potassium hydroxide because the carbonate form is more soluble (Jubin, 1988). The I-129 waste generated by caustic scrubbing in the United Kingdom Windscale-FRP, as well as in France La Hague (Figure 14.17) has been routinely discharged to the sea.

*14.5.2.1.2.5 Iodine Capture on Sorbents* The leading approach to capture radioactive gaseous iodine involves sorption onto silver-containing zeolites. For several decades, the silver-containing zeolite mordenite (MOR) has been a benchmark for radiological iodine capture; however, the molecular basis for its performance remained largely unexplored. To rationally optimize the capture process, Nenoff's group (Chapman, Chupas, and Nenoff, 2010) are presently examining the structure and distribution of the  $\text{AgI}$  when iodine is captured by silver containing MOR. Using a differential approach they have recently shown that the structure and distribution of the  $\text{AgI}$  when iodine is captured by silver containing



**FIGURE 14.17**

Discharges in 1999 from Sellafield reprocessing plant, UK and La Hague reprocessing plant, France (Tbq, logarithmic scale). (From Paviet-Hartmann, P, W. Kerlin, and S. Bakhtiar, Treatment of gaseous effluents from recycling: A review of the current practices and prospective improvements, *Proceedings of the NEA/OECD Workshop on Actinide and Fission product partitioning and transmutation Exchange Meeting*, Nov. 1–5, San Francisco, CA, 2010.)

MOR depends on whether the silver is reduced. Table 14.9 summarizes the solid adsorbents available for iodine capture. The silver-based alumina, silica, and mordenite adsorbents have comparable characteristics and are the currently preferred sorbents. An advantage of the mordenite is that the regeneration of the sorbent is possible, thereby utilizing the silver over the course of several cycles. The stripped iodine is then available for sorption using a cheaper metal or conversion to a waste form preferred for long-term storage.

**14.5.2.1.2.6 Iodine Immobilization** Three different waste forms were proposed for I-129 in the 1980s (Trevorrow et al., 1983). Two of these utilized zeolites for capture. Two of the final waste forms incorporated cement into the system for immobilization and the third consisted of pellets. As Trevorrow et al. (1983) stated in their report: “The very small release limit for I-129 specified by 40 CFR 190 (Table 14.10) makes extreme demands on technologies of collection and retention for this nuclide. Furthermore, the assessment of whether the technologies will comply with this limit is affected by uncertainties such as the distribution of iodine in the plant or the efficiency of the technology for collecting I-129 from gas streams” (Mc Kay, 1980).

**14.5.2.1.2.7 Concentration and Recovery of Krypton** Kr-85 has a half-life of the radioactive decay of 10.7 yr and is produced by both natural and man-made sources. The natural sources include the interaction of cosmic rays with stable isotopes of Kr in the atmosphere. The man-made Kr-85 is principally produced during the fission reactions in LWR or during nuclear atmospheric tests; it is also released from nuclear-fuel reprocessing activities. Due to its atmophile nature, most Kr (>98%) resides in the atmosphere and becomes isotopically well mixed within a few years (Yokochi, Heraty, and Sturchio, 2008). There are no

**TABLE 14.9**

Summary of Solid Adsorbents for I-129 Capture

Adsorbent	Temperature (°C)	Adsorbent Capacity (mg I <sub>2</sub> /g)	Decontamination Factor	NO <sub>x</sub> Resistance	Comments
Activated carbon	<120		10–10 <sup>3</sup>	Unsuitable	
Macroreticular resins	<50	200–1000	10 <sup>3</sup> –10 <sup>4</sup>	High	Organic iodine capacities are lower by a factor of 10, high humidity may adversely affect performance
AgA: silver impregnated alumina	~150	100–235	10 <sup>2</sup> –10 <sup>3</sup>	High	
AC-6120: silver impregnated silica gel	~130	~135	10 <sup>2</sup> –10 <sup>5</sup>	High	Used at the WAK plant Germany, sorbent is not regenerated
AgX: silver exchanged faujasite	~150	80–200	10 <sup>2</sup> –10 <sup>5</sup>	Low	Presence of hydrocarbons and NO <sub>x</sub> adversely affects performance
AgZ: silver exchanged mordenite	~150	~170	10 <sup>2</sup> –10 <sup>5</sup>	High	Thermally stable to 500 °C, can be regenerated
AgZ: silver exchanged mordenite	~150	~170	10 <sup>2</sup> –10 <sup>5</sup>	High	Silver reduced to metallic state by conditioning with hydrogen

Source: Moore, 1970; Pence, Duce, and Maeck, 1972, 1972a; Wilhelm and Schuttelkopf, 1972, Wilhelm and Furrer, 1977; Ackley, 1973; IAEA, 1987.

**TABLE 14.10**Fraction of I-129 that must be Isolated from the Environment as Indicated by 40 CFR 190<sup>1</sup>

Release Limit 40 CFR 190, Ci/GWe yr <sup>-1</sup>	Quantity Processed in UNF <sup>1</sup> Ci/GWe yr <sup>-1</sup>	Fraction <sup>2</sup> that must be Isolated from Environment
$5 \times 10^{-3}$	1.1	0.996

<sup>1</sup> Based on expected nuclide inventories in fuel from the 2:1 ratio of PWR/BWR power plants (DOE/NE-0017) with a 1.5 year cooling period.

<sup>2</sup> Calculated by Trevorrow et al. (1983).

significant natural mechanisms (biological, physical or chemical) to concentrate krypton in the environment (Taylor, 1990). The annual production through these natural paths is approximately  $4 \times 10^{11}$  Bq (Berard et al., 2001). Table 14.11 compares Kr-85 activities in the atmosphere for several accidents and annual release from reprocessing plants.

For a few decades, Kr-85 artificial production shows a continuous increase of its average volumic activity into the atmosphere. In the North hemisphere, the atmospheric concentration was of 0.1 Bq m<sup>-3</sup> in 1959, 0.8 Bq m<sup>-3</sup> in 1980, and 1.2 Bq m<sup>-3</sup> in 2001 (Berard et al., 2001). One cannot say that the removal of tritium and krypton from gaseous waste is an immediate waste management requirement, but it may become one in the future (Lennemann et al., 1975). Opinions differ as to the need to isolate Kr-85. Thus, Geary (1988) states that dispersal is almost certainly preferable to disposal, based on the relatively low inventories and hazards involved. Mellinger (1985) suggested that the population risks arising from the routine release of Kr-85 would not exceed the occupational risk associated

**TABLE 14.11**

Release of Kr-85 in the Atmosphere

<b>Cumulated Release</b>	<b>Activity in Bq (Ci)</b>
Winscale Accident (1957)	$5.9 \times 10^{13}$ ( $1.59 \times 10^3$ )
Chernobyl Accident (1986)	$3.3 \times 10^{18}$ ( $8.92 \times 10^7$ )
Nuclear Tests	$2.0 \times 10^{17}$ ( $5.41 \times 10^6$ )
<b>Annual Release</b>	<b>Activity in Bq (Ci)</b>
Release from la Hague reprocessing plant, France (1999)	$2.9 \times 10^{17}$ ( $7.84 \times 10^6$ )
Release from Sellafield reprocessing plant, UK (1997)	$9.5 \times 10^{18}$ ( $2.57 \times 10^8$ )
Release from Tokai Mura reprocessing plant, Japan (1994)	$1.8 \times 10^{18}$ ( $4.86 \times 10^7$ )

*Source:* Berard P, B. Le Guen, M.L. Perrin, E. Gaillard-Lecanu, V. Chambrette, J. Brenot, and B. Crabol. 2001. IPSN, Institut de Protection et de la Surete Nucleaire, Fiche radionuclide Krypton 85.

with recovery, immobilization, and storage, and that both risks are very low. After January 1, 1983, the U.S. EPA regulations (40 CFR 190) have limited the release of Krypton-85 to the environment from commercial facilities, but criteria for final emplacement of nuclear wastes do not give specific guidance on the disposal of this nuclide. To comply with these regulations, Krypton-85, which would be released during reprocessing of commercial nuclear fuel, would have to be collected and stored if a processing/recycling facility would be built in the United States. Technology currently exists for the separation of krypton from other inert gases and for its storage as a compressed gas in steel cylinders. The requirements, which would be imposed for the 100 year storage of Krypton-85, have led to the development of processes for the encapsulation of krypton within a stable solid matrix (Trevorrow et al., 1983). The satisfaction of not only final emplacement criteria but also transportation regulations will be difficult due to the large amounts of Kr-85 liberated in fuel reprocessing and the incomplete development of the packaging technology. A large fraction of the Kr-85 in UNF may be released into the shear off-gas, which will be combined with the off-gas from subsequent operations. Any Kr-85 remaining in sheared fuel will be released during dissolution to the DOG stream. Various methods of trapping the Krypton contained in off-gases have already been considered (Figure 14.18).

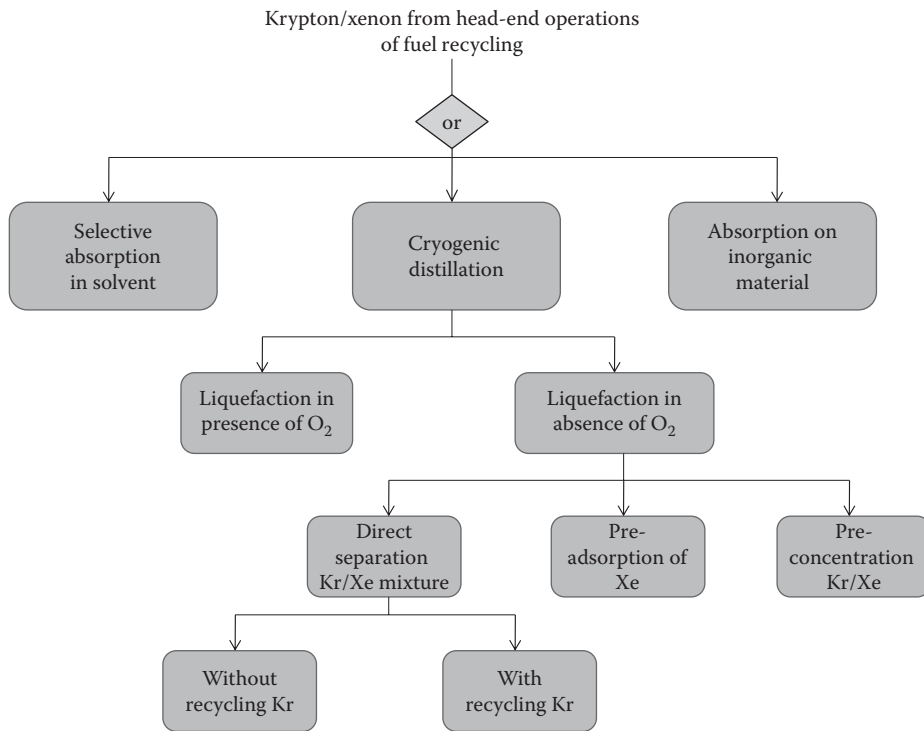
Cryogenic distillation and selective absorption in solvents have been the subject of numerous studies (IAEA, 1980). Adsorption on activated charcoal has the advantage of simplicity and is widely used in reactors in the delay line for the decay of short-lived isotopes, while for its part Kr-85 is discharged in the atmosphere. By using two parallel beds operating cyclically in the adsorption/desorption mode it is, however, possible to trap Kr and Xe. At the same time, this method requires large volumes and fire hazards exist owing to the presence of oxidizing agents.

**14.5.2.1.2.8 Cryogenic Methods** In the air-products industry, cryogenic distillation is a well-known technology for the commercial extraction of krypton, xenon, and other gases from the atmosphere. Cryogenic methods have therefore received much attention for the recovery of Kr-85 from reprocessing plants. A variety of conceptual plant designs are described in (IAEA, 1980).

The cryogenic distillation process has been summarized by Bohnenstingl et al. (1976):

- Removal of I-129 and I-131 on silver-coated silica gel
- Deposition of particulate materials by HEPA-filters



**FIGURE 14.18**

Krypton/ Xenon separation technologies from reprocessing/recycling plant.

- Elimination of  $O_2$  and  $NO_x$  by catalytic conversion with  $H_2$  to  $N_2$  and  $H_2O$
- Drying of the gas stream with molecular sieve
- Deposition of xenon in the solid form at about 80 K, while the remaining gas components are liquefied
- Enrichment of Kr by low temperature distillation of the liquid–gas mixture
- Withdrawal of the highly enriched Kr-fraction in pressurized steel cylinders for final disposal
- Purification of Kr-85 contaminated Xe for further industrial reuse by batch distillation

Difficulties in this operation have been outlined by several authors (Hutter et al., 1986; IAEA, 1980; Taylor, 1990). The other volatile radionuclides should be removed from the off-gas stream before the krypton recovery stage. Thus, processes for the recovery of H-3, C-14, I-129 should precede the krypton isolation process, which would be the final stage of off-gas treatment. Another issue observed is the formation of ozone due to the radiolytic decomposition of the liquid oxygen by the beta radiation emitted by Kr-85. The accumulation of ozone either alone or in combination with traces of hydrocarbons represents an explosion hazard, so most of Kr-85 recovery plants include oxygen removal in the pretreatment stage. The presence of nitrogen oxides as radiolysis by-products or as components of DOG may also react violently. Processes to remove these by-products include an alkaline wash (for  $NO_2$  only) and catalytic decomposition or reduction. Finally, the crystallization of xenon at cryogenic temperatures appears surmountable by appropriate plant design.

Cryogenic distillation was used at the Idaho Chemical Processing Plant (ICCP) in the mid-1960s. This unit was the first large-scale plant based on the cryogenic distillation process and was designed to recover and purify several thousand curies of Kr-85 daily. This process has seen extensive development work in Europe too, especially in Germany and Belgium (Trevorrow et al., 1983). Hutter et al. (1986) summarized in a 10 year development program their cryogenic distillation investigation at the Forschungszentrum Karlsruhe, Germany. This three-stage process REDUKTION/ADAMO/KRETA culminated in a successful three month continuous campaign and the krypton produced was stored in steel cylinders.

**14.5.2.1.2.9 Other Methods** Table 14.12 groups together various separation techniques and shows their advantages and disadvantages. Fluorocarbon absorption is a potential alternative to cryogenic distillation. It is based on the relatively high solubility of krypton and xenon, compared with oxygen, nitrogen, and argon, in liquid freon 12 (difluorodichloromethane,  $\text{CF}_2\text{Cl}_2$ ). Krypton recovery efficiencies in excess of 99.9% using this method with a product purity of 97% were reached at the Oak Ridge Gaseous Diffusion Plant (Taylor, 1990). Following absorption, co-absorbed carrier gas is removed in a fractionator after which a rare-gas concentrate is recovered in a stripper column and the Freon is then reprocessed. One disadvantage is the  $\text{CF}_2\text{Cl}_2$  release and its effect on the ozone layer which may constrain its use in a large-scale extraction process.

Over a dozen prospective adsorbents for krypton were studied and evaluated with respect to adsorption capacity and cost for DOG streams from nuclear reprocessing plants (Monson, 1981). Results showed that, at subambient temperature ( $-40$  to  $-80$  °C), the commercially available hydrogen mordenite had sufficient adsorptive capacity to be the most cost-effective material studied. Silver mordenite has a higher capacity for krypton retention, but is 50 times more expensive than hydrogen mordenite. The results indicated that a solid adsorbent system was feasible and competitive with other developing systems which utilize fluorocarbon absorption and cryogenic distillation. Further work by Ruthven et al. (1981) confirmed that a dealuminated (i.e., aluminum depleted) hydrogen mordenite had

**TABLE 14.12**

Techniques for the Separation of Krypton-85

Process	Advantages	Disadvantages
Cryogenic Distillation	Reasonable cost Separation Kr/Xe Basic technology known and tested Potentially very reliable	Formation of ozone with risk of explosion Requires pretreatment of gases
Absorption by solvents	Low cooling and solvents costs Little risk of explosion No major gas pretreatment	Difficulties in Kr/Xe separation Corrosion risks Fluorocarbons as secondary wastes
Adsorption on activated charcoal	Simple operation Good reliability Low bed volume in the case of low-temperature adsorption	Fire and explosion risk No Kr/Xe separation High cooling cost in the case of low-temperature adsorption
Clathrates, Kryptonates, chemical compounds of fluorine	Kr-85 recovered as a solid or in a chemical compound	Poor stability

Source: Adapted from Hebel W and Cottone G., *Methods of Krypton -85 Management*, Radioactive Waste Management Series, Vol. 10, pp 28–65, Harwood Academic Publishers, UK, 1983.

the best selectivity for krypton of several molecular sieve materials they investigated. Excellent separation of krypton, initially present at 0.2% in  $N_2$ , was reported in a pilot plant operating at an average flow rate of  $2.5 \text{ L min}^{-1}$ .

*14.5.2.1.2.10 Krypton Immobilization* Depending on regulatory criteria, Kr-85, once separated should be immobilized for at least 100 years. The solid state chemistry is limited to a few highly reactive unstable compounds; therefore the immobilization of Kr-85 will rely on physical methods. A variety of methods for Kr-85 encapsulation have been investigated such as (1) ion implantation/sputtering by which gases can be implanted into solids by bombarding the surface of the solid with energetic ions, (2) zeolite encapsulation, in which gases can be entrapped by sintering in solids such as zeolite or porous VYCOR®-type glass which have molecular defects in their structure; gas can be diffused into the solid, and encapsulated in a structural vacancy when the molecular lattice of the solid substrate is sintered at high temperatures (Parsons and Knecht, 1994).

*14.5.2.1.2.11 Remarks* Fundamental information was developed in the 1970s and 1980s. Currently, there are no plans to include krypton or iodine retention by the nuclear fuel reprocessing plants. It appears that there has been an apparent discontinuation of gaseous waste recovery process development implying that no new technologies have been developed recently. With the new fuel cycle R&D, research has restarted on this subject. It may become necessary in the future to contain I-129 or Kr-85 released from nuclear fuel reprocessing activities to reduce the exposure to the local population and the radioactive background throughout the world. Of course, whatever the process will be, either silver-containing zeolite mordenite for I-129 for example or cryogenic distillation for Kr-85, the choice of the process will always depend on the plant operator and the regulatory requirements (Paviet-Hartmann, Kerlin, and Bakhtiar, 2010).

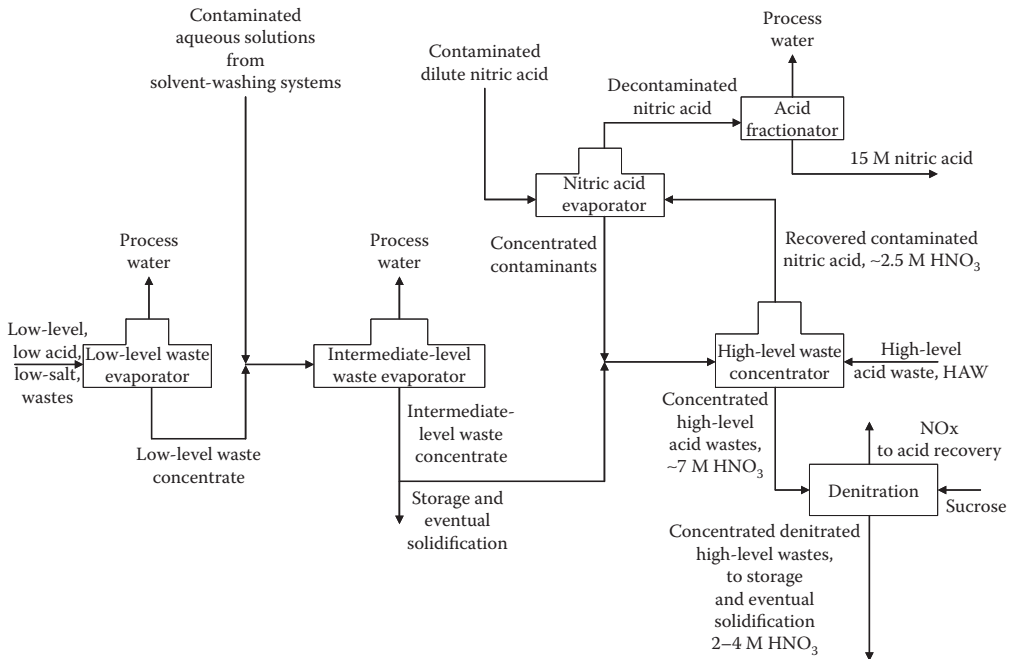
#### *14.5.2.1.3 Nitric Acid Recovery*

The nitric acid evaporated from the HLW concentrator is too dilute and contains too much entrained radioactivity to be recycled without additional treatment. This acid, together with dilute acid waste streams from the uranium and plutonium purification solvent extraction systems, is decontaminated in the nitric acid evaporator. Entrainment can be suppressed by providing partial reflux through a few bubble-plate or perforated-plate trays, backed up by wire-mesh mist eliminators.

Decontaminated acid is separated in the acid fractionator into 15-M acid, water and acid of intermediate concentration as needed in the reprocessing plant (Figure 14.19). The 15-M upper limit is the concentration of nitric acid–water azeotrope. The acid evaporator and fractionator are made of stainless steel and usually run at an absolute pressure of 70–200 Torr to reduce corrosion and the reaction of nitric acid with traces of TBP dissolved or entrained in the acid feed. At 200 Torr, the azeotrope boils at  $86.5 \text{ }^\circ\text{C}$  (Benedict, Pigford, and Levi, 1981).

### **14.5.2.2 Theoretical Models: Distribution Equilibria in Aqueous Systems**

The knowledge of distribution equilibria in PUREX-based processes is useful in designing the solvent extraction systems for a PUREX reprocessing plant and in predicting the change in the performance of an existing plant when operating conditions are changed. The first experimental measurement of the partition of uranyl nitrate, between an aqueous phase water, nitric acid and an organic phase 30% volume TBP in a commercial solvent



**FIGURE 14.19**

Steps in PUREX waste processing and acid recovery. (From Benedict M., *Nuclear Chemical Engineering*, Pigford T.H., H.W. Levi, Eds, Second Eds., Chap 10—Fuel Reprocessing. McGraw-Hill, 1981. With permission.)

(Gulf BT or Amsco 123–15) for a temperature of 25 °C were performed by Coddling, Haas, and Heumann (1958).

Following a brief early period when various extractants for uranium and plutonium were used, TBP became universally used and is the basis for the PUREX process.

Distribution data (partition coefficients) have been measured for particular concentration ranges of solutes and TBP compositions of interests. Many of these data have been published and collections of such data are available (Kolarik and Petrich, 1981). Once the distribution data are available, they have then to be compiled into a form suitable in the calculation procedure to be applied to the liquid–liquid extraction problem. If the procedure is graphical, then the standard McCabe and Tiele approach may be used (Schulz, Burger, and Navratil, 1990). Originally, Fenske or McCabe diagrams were used to calculate the number of stages required for a given operation (extraction, scrubbing, or stripping). With the need for a better understanding of the physical procedures involved in liquid–liquid extraction, demands to optimize plant and process design and to demonstrate to the regulators effective process control and stability, the necessity for improved computational techniques became apparent (Schulz, Burger, and Navratil, 1990). The application of graphical calculation methods was followed in the late 1950s by computer-based techniques which are now exclusively used in PUREX process design.

The first computer program to be published in the open literature (in 1970) was the solvent extraction processes having interacting solutes (SEPHIS) code developed at the US Oak Ridge National Laboratory. The SEPHIS code predicts the equilibrium distribution of uranium, Pu(IV), nitric acid and water between an aqueous phase containing these components and an organic phase containing TBP at any concentration between 2.5 and 100 v/o

at temperature 0–70 °C (Benedict, Pigford, and Levi, 1981; Schulz, Navratil, and Bess, 1987). The U.K. code SIMTX and in its later form—Quantitative TBP Extraction processes (QUANTEX)—is similar to the US code SEPHIS, although there are some basic differences. In the SEPHIS code, the distribution coefficient algorithm is essentially theoretically based, and some experimental data were available for its validation. In the SIMTEX/QUANTEX code, however, the distribution coefficient algorithm is based on many experimentally obtained distribution data. The SEPHIS and SIMTEX/QUANTEX codes are widely used, especially the SEPHIS code which, because of its availability, is internationally used. The output from both the SEPHIS and SIMTEX/QUANTEX codes is a tabulated concentration profile for individual species as a function of time steps which relate to the integration procedure in the computer programs.

Other codes have been developed, such as SEPHIS/MOD4 which simulates the solvent extraction portions of the PUREX or the Thorex process. The program iteratively applies an unsteady state mass balance to a series of ideal mixer settler stages to follow the changes in solute concentrations. Reasonably good agreement has been found between the calculated and experimental concentrations. The components considered by SEPHIS/MOD4 are nitric acid, thorium, uranium, Pu(IV), Pu(III), a plutonium reductant, and inextractable nitrate salts. A maximum of 100 stages is allowed. SEPHIS/MOD4 does not deal with FP or solvent degradation product behavior. Perfect mixing is assumed to occur in the mixers and in each zone of the settlers (Mitchell, 1984).

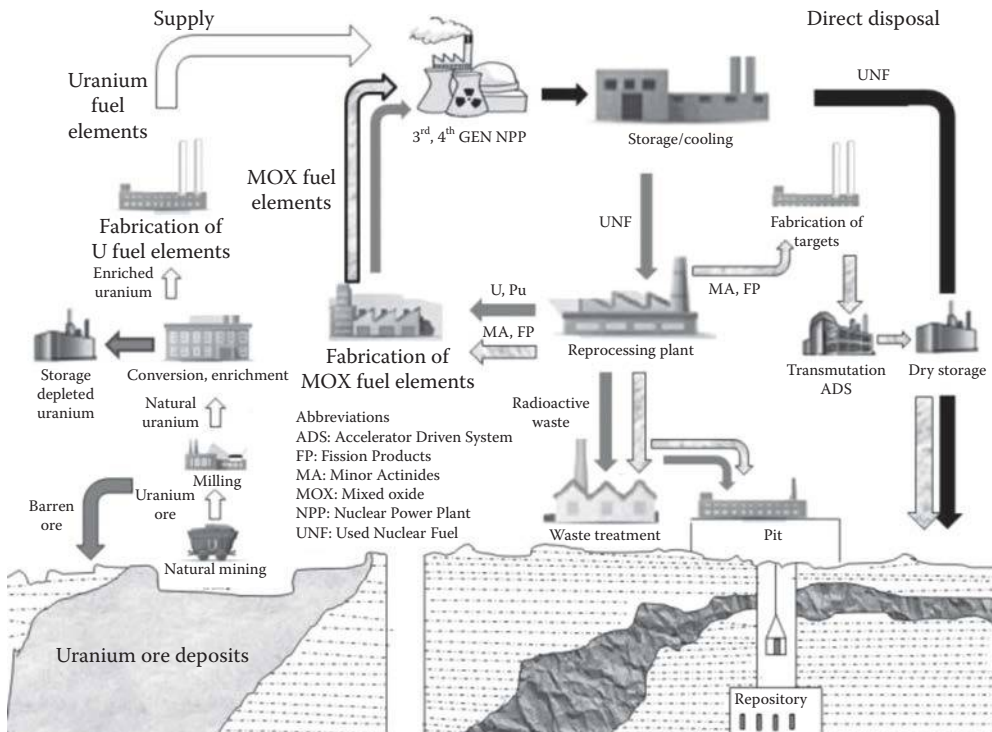
PUBG is another code of the PUREX-solvent extraction system by which plutonium and uranium are recovered from spent nuclear fuel rods. The system comprises several mixer-settler banks. This discrete stage structure is the basis of the algorithms used in PUBG. The stages are connected to provide for the countercurrent flow of the aqueous and organic phases. PUBG uses the common convention that has the aqueous phase enter at the lowest numbered stage and exit at the highest one; the organic phase flows oppositely. The volumes of the mixers are smaller than those of the settlers. The mixers generate a fine dispersion of one phase in the other. The high interfacial area is intended to provide for rapid mass transfer of the plutonium and uranium from one phase to the other. The separation of this dispersion back into the two phases occurs in the settlers. The species considered by PUBG are hydrogen(1+), plutonium(4+), uranyl oxide(2+), plutonium(3+), nitrate anion, and reductant in the aqueous phase and hydrogen(1+), uranyl oxide(2+), plutonium(4+), and TBP (tri-*n*-butylphosphate) in the organic phase. The reductant used in the PUREX process is uranium(4+) or HAN. A maximum of 100 mixer-settler stages all feed streams enter the contactor through the mixers, with only one feed stream of each phase able to enter a mixer. All product streams exit the contactor from the settlers, with only one product stream of each phase able to exit a settler. A product stream may re-enter the system through only one mixer, in addition to any feed streams entering that mixer (Geldard and Beyerlein, 1985).

### 14.5.3 Advanced Recycling Technologies

Even though nuclear power has demonstrated over the past 30 years its capacity to produce base-load electricity at a low, predictable, and stable cost due to the very low economic dependence of the price of uranium, the management of UNF remains the “Achilles” heel of this energy source because the storage of UNF is increasing as evidenced by the following number with 2,000–23,000 of UNF produced each year by the 104 US nuclear reactor units which equates to a total of 62,000 UNF assemblies stored in dry cask storage and 88,000 stored in pools (Paviet-Hartmann, 2012). The sustainability of the current nuclear fuel cycle is not completely achieved because the current industrial PUREX process does

not ensure a complete and efficient recycling of actinides. Shifting from an actual PUREX process to a more advanced recycling technology in which natural resources are saved, nuclear waste is minimized, efficiently confined, and disposed of safely requires (1) the optimization of uranium and plutonium recycling, (2) the implementation of MAs recycling as they are the main contributors to the long-term heat power and radiotoxicity of nuclear waste, and (3) a suite of thermal and fast neutron spectra (Poinssot et al., 2012).

As of today, two strategies are offered to countries using nuclear energy. Some countries such as the United States, Sweden, Finland, or Canada have opted for a once-through fuel cycle (Figure 14.20), or geological disposal of UNF. Other countries such as France, Japan, Russia, United Kingdom, and India have decided to recycle UNF (Figure 14.18) using the available industrial PUREX process, followed by the conditioning of HLW in borosilicate glass for final geologic disposal. A third option called closed fuel cycle, which may integrate a modified PUREX process, envisions the recycling of minor actinides (MA) and FPs to be burned into a fast reactor or used as a target for transmutation in an accelerator-driven system (ADS) (Figure 14.20). Reusing nuclear materials from light water reactors (LWR) used fuel not only supplies fissile and fertile fuels needed for fast reactors but also reduces the potential radiotoxic inventory of HLW to be disposed of while reusing used MOX fuels. The R&D efforts on the separation of MA have been under investigation for several years worldwide. Recycling MA is not a necessity but it is an option that may contribute to optimizing closed fuel cycle strategies while contributing 1- to reducing the



**FIGURE.14.20**

Nuclear fuel cycle depicting the three available strategies: once-through fuel cycle, modified open cycle and closed fuel cycle. (From Paviet-Hartmann, P. 2012. Overview of the international R&D recycling activities of the nuclear fuel cycle. *International Journal on Nuclear Safety and Simulation*, 3:267–275.)

long-term decay heat and radiotoxic inventory of waste disposed in a geologic formation, 2- to slightly increasing the energy generated from natural uranium (above 90% vs. about 80% with recycle of U/Pu only), and 3- to possibly enhancing proliferation resistance (while increasing the decay heat and neutron source in re-fabricated nuclear fuel).

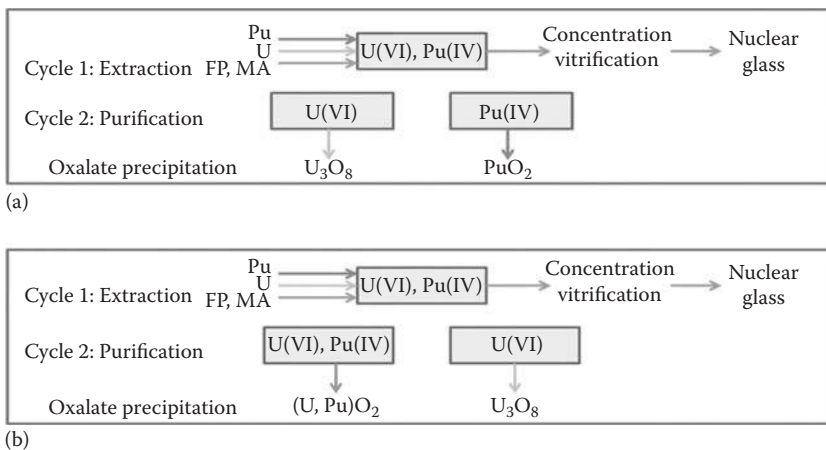
Advanced separations and transmutation technologies such as the ones developed currently in the United States and in Europe, (UREX, TALSPEAK, DIAMEX, SANEX, GANEX processes ...) will be an excellent first step in demonstrating all the necessary technologies for a sustainable future for nuclear energy (Finck, 2005).

**14.5.3.1 COEX™ Process**

To reduce the risk of plutonium diversion from the PUREX process, as shown in Figure 14.21(a), the COEX™ process (COEXtraction process) (Figure 14.21(b)) is being developed by CEA and AREVA. This is an evolutionary process, based on the PUREX process, designed to eliminate the production of pure plutonium product stream. By not separating plutonium, and by being compatible with future advanced MA recycling processes, the COEX™ process aims at further enhancing proliferation resistance (Senentz, Drain, and Baganz, 2009). The main characteristics of COEX™ are (1) no plutonium separation at any point of the process and (2) coprecipitation by oxalic acid to produce a mixed solid solution of uranium and plutonium dioxide (U, Pu)O<sub>2</sub>. This process aimed at further enhancing proliferation resistance, and keeping open possible evolutions to take into account new types of reactors or future changes in management strategies of the TRU elements (Herbst, Baron, and Nilsson, 2011).

**14.5.3.2 UREX**

From 2003 to 2007, a number of potential UREX + flowsheets for the processing of commercial UNF were developed and tested with actual UNF at laboratory-scale at Argonne National Laboratory (Regalbuto, 2011; Vandegrift, 2007) (Table 14.13). The UREX + 1 series was intended for the extraction of the TRU elements as a group; there is no separation among the TRUs which are to be burned as fuel in fast spectrum reactors (FR).



**FIGURE 14.21** Global recycling scheme of PUREX (a) and COEX™ (b) processes.

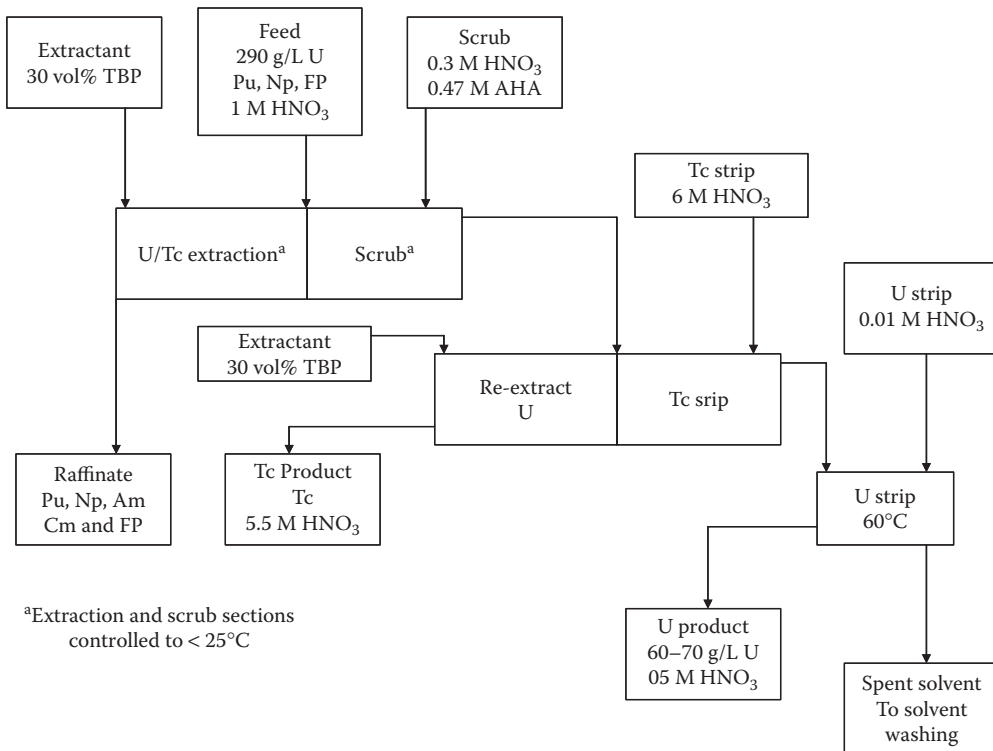
**TABLE 14.13**

UREX + Process: Possible Variations

Process	1 <sup>st</sup> Product	2 <sup>nd</sup> Products	3 <sup>rd</sup> Product	4 <sup>th</sup> Product	5 <sup>th</sup> Product	6 <sup>th</sup> Product	7 <sup>th</sup> Product
UREX + 1	U (highly purified)	Tc, I (LLFPs, dose issue)	Cs, Sr (short-term heat mgmt.)	Other FPs	TRU + Ln (Temporary storage)		
UREX + 1a	U (highly purified)	Tc, I (LLFPs, dose issue)	Cs, Sr (short-term heat mgmt.)	FPs (including lanthanides)	TRU (group extraction)		
UREX + 2	U (highly purified)	Tc, I (LLFPs, dose issue)	Cs, Sr (short-term heat mgmt.)	Other FPs	Pu + Np (for Fast reactor recycle fuel)	Am + Cm + Ln (Temp. storage)	
UREX + 3	U (highly purified)	Tc, I (LLFPs, dose issue)	Cs, Sr (short-term heat mgmt.)	FPs (including lanthanides)	Pu + Np (for Fast reactor recycle fuel)	Am + Cm (heterogeneous targets)	
UREX + 4	U (highly purified)	Tc, I (LLFPs, dose issue)	Cs, Sr (short-term heat mgmt.)	FPs (including lanthanides)	Pu + Np (for Fast reactor recycle fuel)	Am (heterogeneous targets)	Cm (storage)

Source: Vandegrift, G.F. 2007. Demonstration by countercurrent solvent extraction of the TALSPEAK process for separation of Np, Pu, Am, and Cm from rare earth elements. *31st Annual Separations Conference*, June 11–15, Las Vegas, NV.



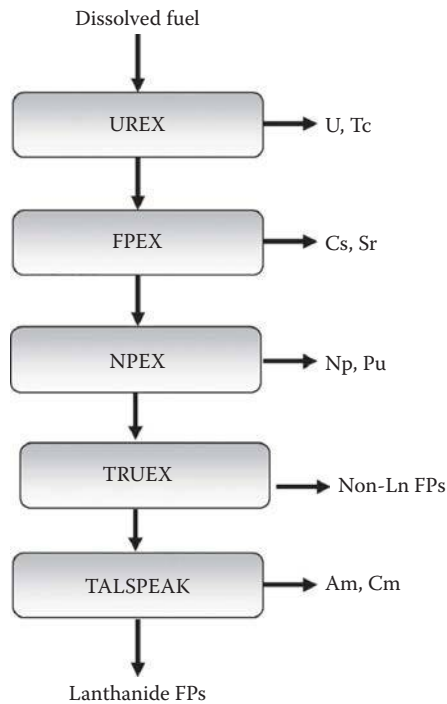


**FIGURE 14.22**

UREX process Flowsheet. (From Thompson, M.C. et al., Demonstration of the UREX solvent extraction process with Dresden reactor fuel solutions, *WSRC-TR-2002-00444*, Savannah River Site, Aiken, SC, 2002.)

The UREX +2 and UREX +3 process series were intended for LWR or FR reactor recycle of plutonium and neptunium. The UREX +4 process was selected as it provides the added option of burning americium in specially designed target assemblies. UREX +2, +3, and +4 were designed for heterogeneous recycling, possible as an evolutionary step to preclude the need for the remote fabrication of fuel. The flowsheet (Figure 14.22) has been successfully tested with simulated solutions in batch and centrifugal contactor tests at Argonne National Laboratory (ANL) (Thompson et al., 2002).

The UREX process is the first solvent-extraction operation in a group of separation processes generically called UREX + that were developed as a baseline set of separation operations to partition the components of the used light-water reactor fuel to enable the recycling of the actinides and more efficient disposal of the FPs (Figure 14.23). Simple hydrophilic hydroxamic ligand such as acetohydroxamic acid (AHA) was used for the stripping of tetravalent actinides in UREX process flowsheet. This approach is based on the high coordinating ability of hydroxamic acids with tetravalent actinides,  $\text{Np}^{4+}$  and  $\text{Pu}^{4+}$ , compared with hexavalent uranium,  $\text{UO}_2^{2+}$ . Thus, the use of AHA offers a route for controlling neptunium and plutonium in the UREX process by complexant based stripping of  $\text{Np(IV)}$  and  $\text{Pu(IV)}$  from the TBP-solvent phase, while  $\text{U(VI)}$  ions are not affected by AHA and remain solvated in the TBP phase. Therefore, the UREX process selectively extracts uranyl nitrate ( $\text{UO}_2(\text{NO}_3)_2$ ) and pertechnetate from dissolving used fuel in nitric acid into a mixture of tributylphosphate and hydrocarbon diluent as  $\text{UO}_2(\text{NO}_3)(\text{TcO}_4)\cdot 2\text{TBP}$  species, and back extracts these species into diluted nitric acid. After the lead UREX separation, there is



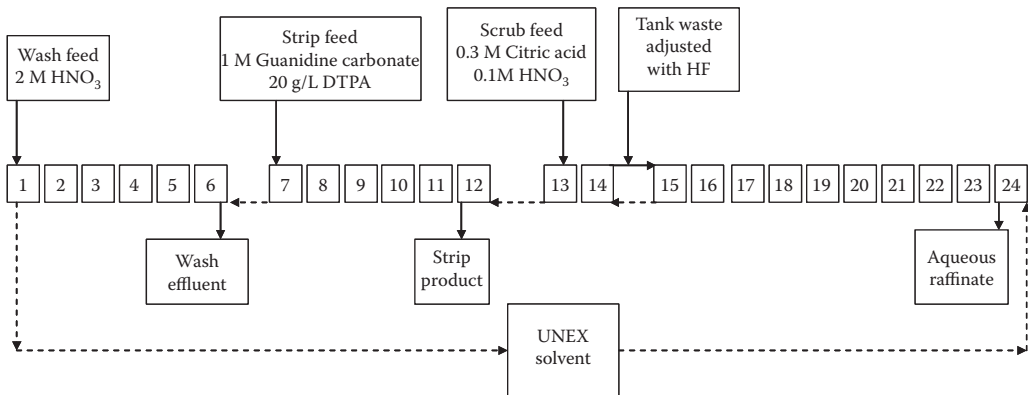
**FIGURE 14.23**  
UREX+ processes outline.

flexibility in the follow-on separation modules (Figure 14.23). Figure 14.23 depicts a cascade of five different extraction processes where the UREX process is followed by the FPEX process for the specific extraction of cesium and strontium by the fission product extraction (FPEX) process is based on two highly specific extractants: 4,4'(5')-Di(*t*-butyldicyclohexano)-18-crown-6 (DtBuCH18C6) for the extraction of Sr and Calix[4]arene-bis-(*tert*-octylbenzo-crown-6) (BOBCalixC6) for the extraction of Cs (Riddle et al., 2005). Laboratory test results of the FPEX process, using the simulated feed solution spiked with radiotracers, indicated good Cs and Sr extraction and stripping performance (Law et al., 2006).

The NPEX process separated plutonium and neptunium from the other components in the FPEX raffinate. Like UREX, NPEX solvent is TBP. Although the process was designed to extract neptunium and plutonium, uranium is added to the extracted neptunium and plutonium in the strip section to yield a U–Np–Pu product with a controllable elemental ratio (Regalbuto, 2011). Following the NPEX process, the TRUEX process is used to recover TRUs and lanthanide fission products using octyl(phenyl)-*N,N*-di-isobutylcarbamoylmethylphosphine oxide (CMPO). The final process is the TALSPEAK process with the separation of minor actinides americium and curium from the lanthanide FPs using di-2-ethyl-hexylphosphoric acid.

### 14.5.3.3 UNEX Process

Collaborative research by the Idaho National Laboratory and the Khlopin Radium Institute (St. Petersburg, Russia) has developed and validated the concept of a Universal Extraction (UNEX) process, for the simultaneous separation of cesium, strontium, the actinides, and the lanthanides from acidic waste solutions. The UNEX process solvent consists of



**FIGURE 14.24**

Flowsheet for demonstration of the UNEX process. (Data from Law, J., S. Herbst, T. Todd, V. Romanovskiy, V. Barbain, V. Esimantovskiy, I. Smirnov, and B.N. Zaitsev, 2001. The universal solvent extraction (UNEX) process: II. Flowsheet development and demonstration of the UNEX process for the separation of cesium, strontium, and actinides from actual acidic radioactive waste. *Solvent Extraction and Ion Exchange* 19(1):23–36.)

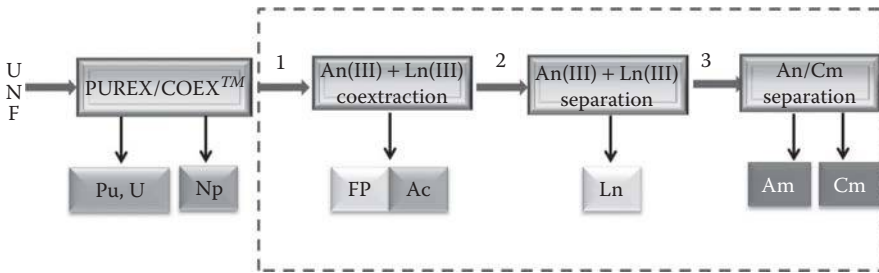
chlorinated cobalt dicarbollide for the extraction of <sup>137</sup>Cs, polyethylene glycol for the extraction of <sup>90</sup>Sr, and diphenyl-*N,N*-dibutylcarbamoyl phosphine oxide for the extraction of the actinides and lanthanides. A non-nitroaromatic polar diluent consisting of phenyltrifluoromethyl sulfone has been developed for this process. A UNEX flowsheet (Figure 14.24) consisting of a single-solvent extraction cycle has been developed and demonstrated with actual acidic radioactive tank waste at the INL using 24 stages of 2 cm diameter centrifugal contactors installed in a shielded cell facility (Law et al., 2001; Romanovskiy et al., 2001).

For future advanced nuclear systems, minor actinides are considered more as a resource to be recycled and transmuted than to be disposed of directly into a nuclear repository. A key feature of advanced fuel cycles technologies would be to separate MA and ultimately americium from curium. Several countries are investigating the separation of MA from a PUREX/COEX™ based process raffinate or a modified PUREX process raffinate using new extractant molecules with two potential options for actinide separations:

1. A selective separation of MA for interim storage, pending a decision regarding their transmutation in heterogeneous recycling mode either in fast reactor (blankets) or in ADS. As an example, the DIAMide EXtraction (DIAMEX)/Selective ActiNide EXtraction (SANEX) processes developed in France (Serrano-Purroy et al., 2005), TRUEx/TALSPEAK processes developed in the United States (Nilsson and Nash, 2007, 2009; Vandegrift and Regalbuto, 1995), ARTIST process developed in Japan) (Figure 14.25) are aiming to separate selectively MA.
2. Other simplified processes are aiming to separate a group of actinides using an integrated fuel cycle (with online fuel recycling and re-fabrication) with the prospect of their homogeneous recycling in a fast reactor (i-SANEX and GANEX processes developed in Europe, TRUSPEAK process developed in the USA).

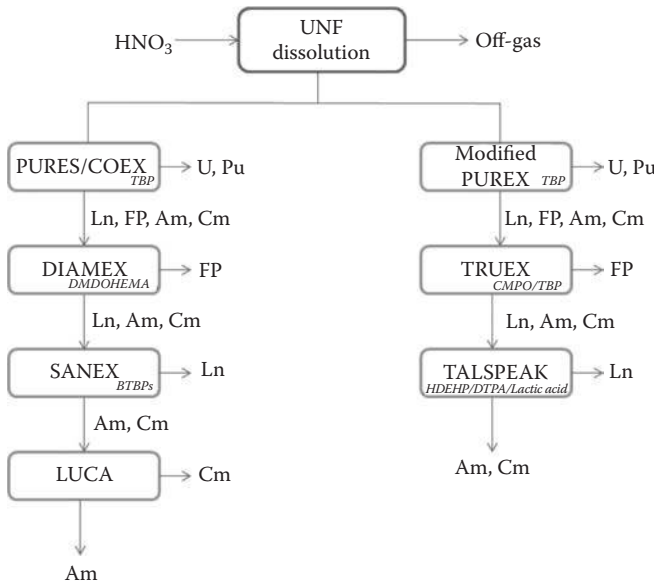
#### 14.5.3.4 Coextraction Followed by Separation of Trivalent Actinides and Trivalent Lanthanides

In France, the DIAMEX/SANEX processes (Figure 14.26), studied at the Commissariat à l'Énergie Atomique (CEA) Marcoule, aims at recovering MA selectively from PUREX raffinate,



After PUREX/COEX™ process  
 Step 1—Coextraction of trivalent actinides (An(III)), and trivalent lanthanides (Ln(III))  
 Step 2—Separation of trivalent actinides (An(III)) from trivalent lanthanides (Ln(III))  
 Step 3—Separation of trivalent americium (Am(III)) from trivalent curium (Cm(III))

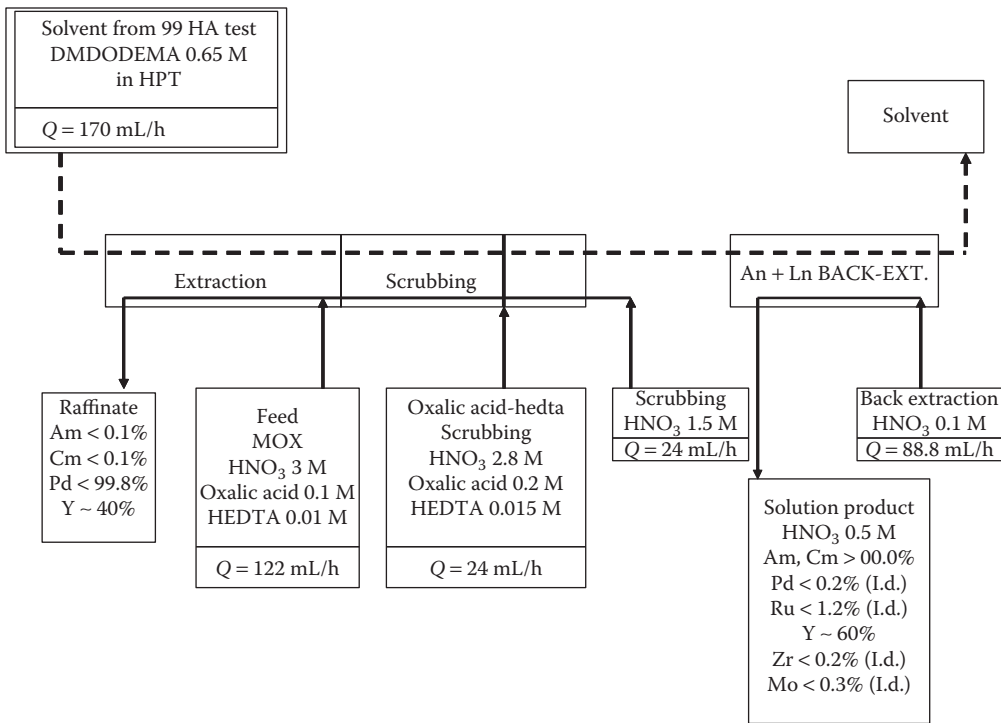
**FIGURE 14.25**  
 Minor actinides separation strategy.



**FIGURE 14.26**  
 Processes involving a coextraction followed by separation of minor actinides.

by solvent extraction. One of the reference molecules of the DIAMEX process is *N,N'*-dimethyl-*N,N'*-dioctyl-hexyl-ethoxy-malonamide (DMDOHEMA) (Charbonnel et al., 2002; Geist et al., 2006). This malonamide is used alone as the extractant, whereas it is mixed with di(2-ethylhexyl) phosphoric acid (HDEHP), an acidic extractant, in the other parts of the process.

The DIAMEX (DIAMide Extraction) process was first developed by Musikas (1987) at the Fontenay-aux-Roses Research Center (France) and by Madic and Hudson (1998) to extract minor actinides (americium and curium) and lanthanides to reduce the radiotoxicity and volume of waste. This process is based on the use of malonamide extractants (Figure 14.27). For the first version of DIAMEX, di-methyl-di-butyltetradecylmalonamide (DMDBTDMA) was developed (Madic et al., 2002; Musikas, 2006; OECD, 1999, 124–125), including countercurrent

**FIGURE 14.27**

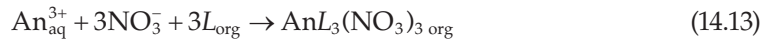
DIAMEX 2000 ATALANTE Hot Text: Flowsheet and main results. (Data from Musikas, C. 2006. Minor actinides partitioning. European research efforts in separation chemistry. *Presentation Performed at Oak Ridge National Laboratory*, March 2006, Oak Ridge, TN.)

hot tests. However, some problems were reported while using the process on tests with high-activity liquid wastes (Baudin et al., 1993; Morss et al., 2006, 2756). A continuous countercurrent experiment using a 16-stage centrifugal extractor battery was tested using 1 M DMDOHEMA in HPT. Coextraction of Zr, Mb, and Pd was prevented by using oxalic acid and HEDTA (Modolo et al., 2007). High decontamination factors have been obtained while four stages for back-extraction were sufficient to recover more than 99.9% of lanthanides and actinides. Synergetic effects have been observed when mixing DMDOHEMA with HDEHP at 0.5 M–1 M  $\text{HNO}_3$  while antagonism was noticed at lower acidity with the formation of an adduct “HDEHP-DMDOHEMA” which decreases the effective HDEHP concentration (Berthon et al., 2012). The diamide derivatives possess a good extractability and respond to the CHON concept (completely incinerable and produces no radioactive solid waste by combustion). Americium and curium are then separated from the lanthanides in a following SANEX process.

The Selective ActiNide Extraction (SANEX) process has been developed by Madic and Hill (Hill et al., 2002, 17–21; Madic et al., 2002). As part of the management of minor actinides, it has been proposed that the lanthanides and trivalent minor actinides (Am and Cm) should be removed from the PUREX raffinate by a process such as DIAMEX in France or TRUEX in the USA. To allow the actinides such as americium to be reused in industrial sources or used as fuel, the lanthanides must be removed. The lanthanides have large neutron cross sections and hence they would poison a neutron-driven nuclear reaction.

Several nitrogen terdentate ligands have been studied but the most exciting results were obtained with the pyridine-bis-1,2,4-triazines derivatives (BTPs), first synthesized

by Kolarik(Madic et al., 2002), which showed interesting properties such as the separation of trivalent actinides from trivalent lanthanides with a separation factor as high as  $SF_{Am/Eu} = 140$  and this for high aqueous nitric acid concentration. An(III) extraction proceeds through a solvation mechanism involving three molecules of BTP, as shown in the following reaction (14.13):

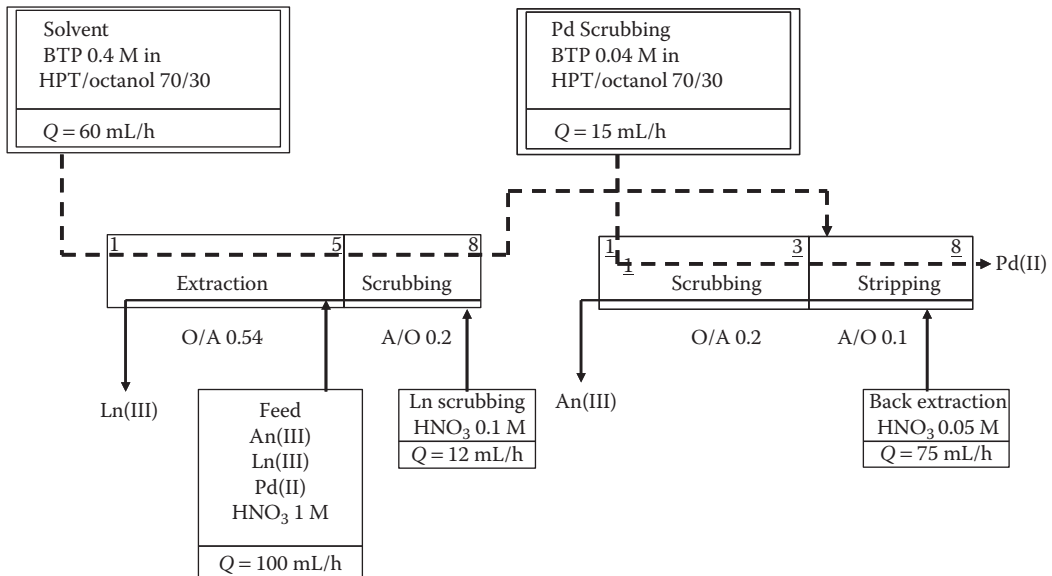


with  $L = \text{BTP}$ .

Among the different bis-triazinyl-pyridine molecules studied at CEA-Marcoule, the 2,6-bis(5,6-*n*-propyl-1, 2, 4-triazin-3-yl)-pyridin (*n*Pr-BTP) was chosen to carry out a counter-current SANEX test (Hill et al., 2002), the feed solution of which was a genuine back-extraction solution issued from a DIAMEX process. The observed performances were very satisfactory for the separation of trivalent actinides from trivalent lanthanides. Studies allowed designing a first flowsheet with the following sections: an extraction section, a scrubbing section to back-extract the part of lanthanides (III) that follows the actinides (III) in the solvent, a back-extraction section to recover actinides (III) with an organic scrubbing to remove Pd(II) well extracted by BTP (Rat and Heres, 2002) (Figure 14.28).

So far, around 140 new compounds have been synthesized in Europe and screened following standardized protocols. Seventy-five percent of these compounds are lipophilic extracting agents, whereas the remaining 25% are hydrophilic complexing agents (Bourg et al., 2012). The reference molecule for the SANEX process is as of today (November, 2013) 2,6-Bis(5,5,8,8-tetrahydro-1,2,4-benzotriazin-3-yl)pyridine (CyMe4-BTBP).

The Lanthaniden Und Curium Americium (LUCA) process is developed in Germany for the selective separation of Am(III) from the solution issued either from the DIAMEX-SANEX processes or the GANEX process. It uses a mixture of bis(chlorophenyl)dithiophosphinic acid ((ClPh)<sub>2</sub>PSSH) and tris(2-ethylhexyl)phosphate (TEHP) as the extractant.



**FIGURE 14.28**

Flowsheet of the BTP Test. (Data from Rat, B. and X. Heres. 2002. Modelling and achievement of a SANEX process flowsheet for trivalent actinides/lanthanides separation using BTP extractant (bis-1, 2, 4-triazinyl-pyridine). 7th Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation, October 14–16, Jeju, Korea.)

More than 99.5% of Cm(III) was directed to the raffinate (Modolo, Kluxen, and Geist, 2009). However, considerable degradation (by oxidation) of the  $(\text{CIPh})_2\text{PSSH}$  has been observed at higher acidity ( $>0.5 \text{ M HNO}_3$ ). Since 2008, the development and validation of the EXAm process (extraction of americium) is being pursued in France. Tridentate hydrosoluble tetraethylidiglycolamide (TEDGA) has been introduced in the EXAm process to complex Cm(III) and heavier lanthanides and allows the recovery of Am(III) alone (Charbonnel et al., 2012). In the United States, the TRUEX solvent extraction process (Figure 14.26) is capable of separating, with very high efficiency, small quantities of TRU elements (e.g., Np, Am, Pu, Cm) from aqueous nitrate or chloride solutions that are typically generated in reprocessing plant operations or in plutonium production and purification operations (Law et al., 1999).

The TRUEX (TRAnsUranic EXtraction) process was invented and developed at ANL (Horwitz et al., 1985; Leonard et al., 1994; Mathur and Nash, 1998; Schulz and Horwitz, 1988; Suresh, Murali, and Mathur, 2001; Vandegrift et al., 1984) to decontaminate the huge amounts of TRU waste accumulated in the United States during the Cold War in defense nuclear material production sites (Hanford, Idaho). The TRUEX process is characterized by the use of the bifunctional extractant CMPO (octyl(phelnyl)-*N,N*-di-isobutylcarbamoylmethylphosphine oxide) which is very efficient at extracting all 3+, 4+ and 6+ actinides from acidic solutions. This extractant is combined with TBP and a diluent to formulate the TRUEX solvent. The diluent is typically a normal paraffinic hydrocarbon (NPH) (Morss et al., 2006, vol. 2: 1281–1284 and vol. 4; OECD, 1999). The TRUEX process (Figure 14.29) comprises four operations: extraction, scrubbing of the organic phase, one or more stripping steps and solvent cleanup. One or two contactor stages for washing the stripped TRUEX process solvent with a dilute sodium carbonate solution are typically included removing solvent degradation products and any traces of metal ions.

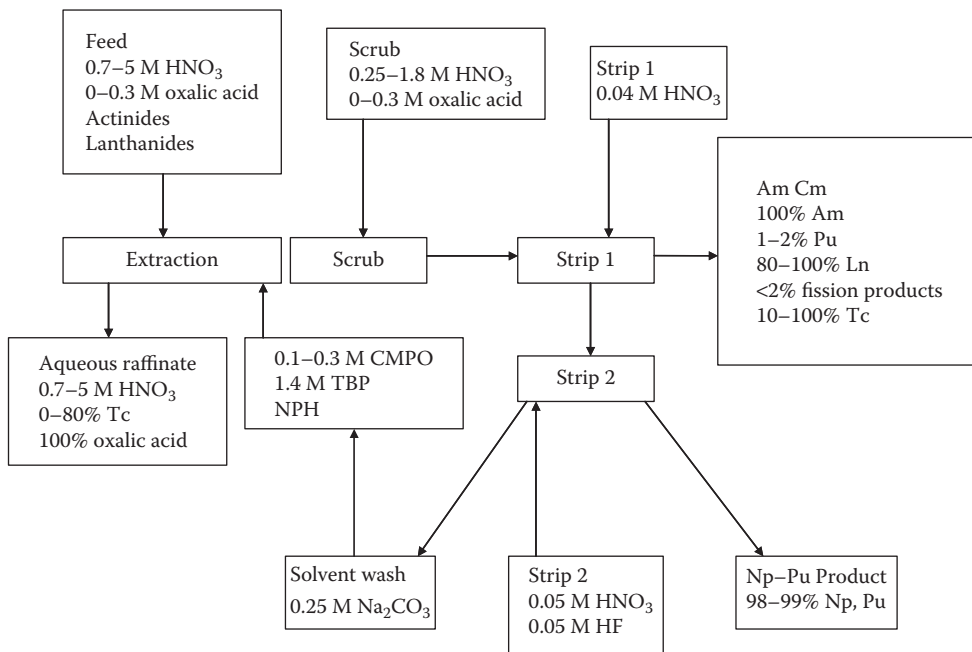


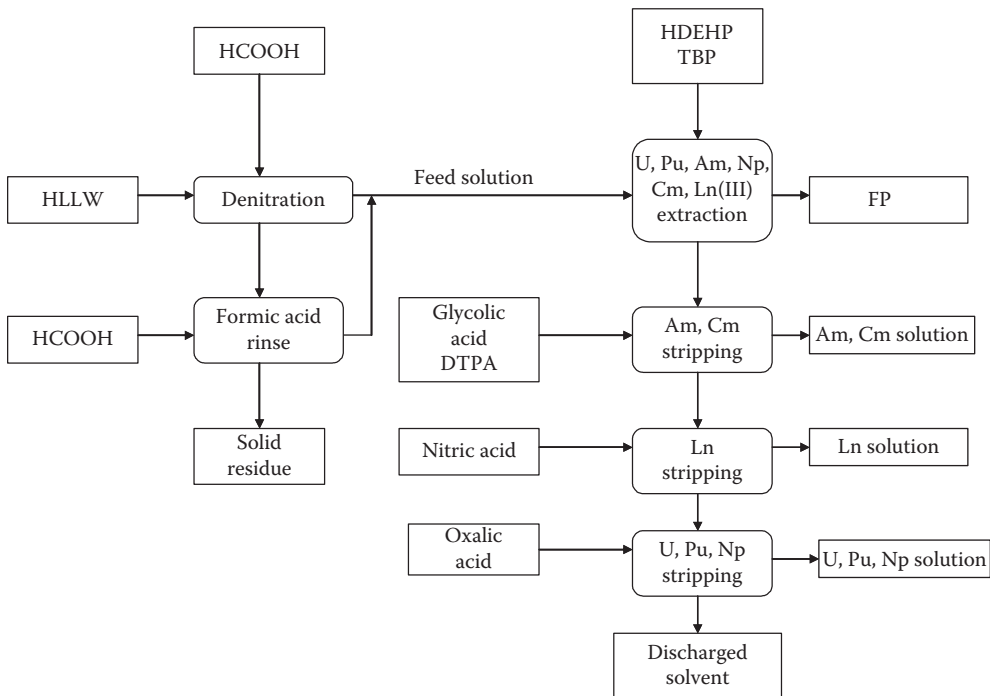
FIGURE 14.29

General TRUEX flowsheet from Choppin 2002: 9.

The most extensive pilot-scale testing of the TRUEX process has been done at the former Idaho National Engineering and Environmental Laboratory. Several TRUEX demonstration runs have been made on sodium-bearing wastes (Law, 1998), a secondary acidic HLW. Variations of the TRUEX process have been conducted in different countries and an extended list of the investigations can be found in Morss et al., 2006, vol. 2, 1283–1284 and vol. 4: 2743. Recent investigations evaluated the effects of radiolytic degradation on the efficacy of the TRUEX flowsheet and showed that gamma irradiation adversely impacted flowsheet performance as measured by the decreasing americium and europium distribution ratios in the extraction section as accumulated dose increased.

The generic applicability of the TALSPEAK process concept (Trivalent Actinide Lanthanide Separation by Phosphorus reagent Extraction from Aqueous Complexes) is demonstrated by the work of Baybarz (1965). This process (Figure 14.30) relies on aqueous complexing agents to selectively retain actinides in the aqueous phase to achieve the separation of trivalent lanthanides from trivalent actinides. The separation of trivalent actinides, Am and Cm, from the lanthanides is a particularly challenging task due to the similarities of the chemistries of these two groups of metal ions (Nilsson and Nash, 2007). Lanthanides (Ln) are separated from trivalent actinides (An) using di-(2-ethylhexyl) phosphoric acid (HDEHP) and then the partition of the different extracted elements is realized by selective stripping of An (III) from the loaded solvent with the help of aqueous stripping solutions containing the complexing agents glycolic acids (Morss et al., 2006, 2671, 2673; NEA, 1997, 50; OECD, 1999, 119–120).

In a modification of the TALSPEAK process known as the Reverse TALSPEAK process, trivalent actinides, and lanthanides together are extracted by 1 M HDEHP from the 0.1 M



**FIGURE 14.30**

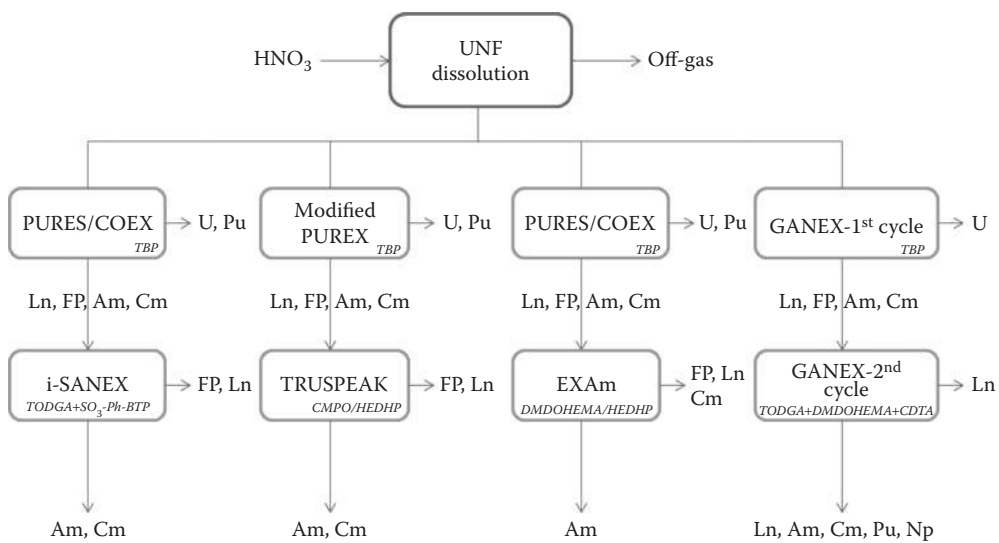
TALSPEAK process. (Data from OECD, 1999, *Actinide and Fission Product Partitioning and Transmutation, Status and Assessment Report*. Paris, France.)



HNO<sub>3</sub> feed solution. Am and Cm are then selectively stripped from the organic phase with a mixture of 0.05 M DTPA (diethylenetriamine pentaacetate and 1.5 M lactic acid at a suitable pH adjusted with ammonia (Morss et al., 2006, 2760). Though not deployed as such for accomplishing lanthanide-trivalent separations at process scale, the critical reagent in TALSPEAK aminopolycarboxylic acids, have repeatedly been employed in the conceptual development of actinide–lanthanide hydrometallurgical separation processes. This process has been proven in the laboratory to successfully separate lanthanides from trivalent actinides with a single-stage separation factor between the least extracted lanthanide (typically neodymium) and americium of 10–100, depending on conditions (Nilsson and Nash, 2009). In the context of modern process design, the aminocarboxylates are acceptable reagents, as they are composed of only C, H, O, and N, and, hence, can be fully incinerated. This class of compounds is known to cause difficulties in storage, as hydrogen generation in waste tanks at Hanford has shown (Morss et al., 2006, 2760).

**14.5.3.5 Simplified Recovery of Minor Actinides from PUREX or Modified PUREX Process Raffinate**

In Japan, the ARTIST process uses advanced amide-based extractants such as the tridentate ligand *N,N,N',N'*-Tetraoctyldiglycolamide (TODGA) (Sasaki et al., 2001) to separate actinides from FPs. TODGA has good radiolytic and hydrolytic stability properties which makes it a strong candidate for trivalent actinides separation. The separation produces two streams, U- and TRU-bearing phases that are then solidified and stored for future treatment. The radiolytic degradation products of TODGA were determined to be *N,N*-dioctyldiglycolamic acid, *N,N*-dioctylamine, and various *N,N*-dioctylmonoamides, which were formed by ruptures of amide- and ether-bonds in TODGA. The radiolysis of TODGA was enhanced in the presence of *n*-dodecane (Sugo et al., 2007). In Europe, alternatives concept to PUREX-DIAMEX-SANEX processes are being developed with either a one-cycle SANEX process or a new GANEX process (Figure 14.31). The one-cycle SANEX



**FIGURE 14.31** Simplified processes for minor actinides recovery.

(innovative SANEX or *i*-SANEX) separates MAs directly from a PUREX raffinate by combining TODGA and SO<sub>3</sub>-Ph-BTP molecules (Magnusson et al., 2007). The benefit with this process is the reduction of process steps from 2 to 1 compared with the DIAMEX+SANEX concept. Here, the trivalent actinides and trivalent lanthanides are first coextracted, leaving the remaining fission and corrosion products in the raffinate, followed by a selective stripping of the actinides by SO<sub>3</sub>-Ph-BTP (Geist et al., 2012). The new GANEX (Bell et al., 2012) aims at the recovery of all transuranium elements from the high activity UNF dissolution solution. It proceeds in two cycles. The first cycle extracts selectively uranium, while the second cycle achieves the coextraction of all transuranium elements (Figure 14.31). The solvent consists of a mixture of TODGA and DMDOHEMA in Exxsol D80. The undesired coextraction of some FPs is realized by adding 1,2-cyclohexanediaminetetraacetic acid (CDTA). In the United States, Lumetta et al. (2010) are developing a single process called TRUSPEAK process (Figure 14.31) which combines the attributes of TRUEx and TALSPEAK processes. The extractants from both processes can be combined into a single process solvent to separate (1) the lanthanides and actinides from acidic HLW and (2) the actinides from the lanthanides in a single solvent extraction process. CMPO is combined with the TALSPEAK solvent HDEHP in dodecane. In doing this, it is envisioned that the CMPO chemistry would dominate under conditions of high acidity ( $\geq 1$  M HNO<sub>3</sub>), resulting in the coextraction of the transuranium elements and lanthanide elements into the organic phase. After suitable scrubbing steps, contacting the loaded solvent with a buffered DTPA solution at pH ~3 to 4 will result in condition in which the HDEHP chemistry dominates, and the system will behave in a manner analogous to a reverse TALSPEAK process. The greater affinity of DTPA for the transuranium ions (TRUs) versus the lanthanides causes the TRUs to be selectively stripped into the aqueous phase, thereby separating them from the lanthanides.

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## 14.6 Modern Pyrochemical Reprocessing

In the early era of nuclear power, many processes were investigated for the recovery of fissionable material from nuclear fuel. From 1956 to 1962, pyrochemical processes such as oxidative slagging, halide slagging, pyroredox, and electrorefining were investigated of recovery material from fast-breeder reactors (Moser and Navratil, 1983). Uranium was recovered from Experimental Breeder Reactor (EBR-II) fuel by melt refining (a type of pyroprocessing) from September 1954 to 1960. These operations processed approximately 5 MT (35,000 metal fuel pins) of spent fuel and included remote fuel fabrication with the recycled fuel being used in EBR-II reactor (Stevenson, 1987). The cessation of fuel processing and fabrication was not due to separations technology or equipment problems, but was simply a change in EBR-II mission to an irradiation facility for the national fast reactor development program.

After the US Government decided to stop pursuing separations of pure plutonium for civilian nuclear power, the ANL began developing the IFR that included a new improved pyrochemical process that included fuel recycling (Burris, Steunenberg, and Miller, 1987). Separations are performed by an electrorefining process, as described in numerous articles (Ackerman et al., 1997; Laidler et al., 1997). Actinides are recovered as a group so that pure plutonium never occurs in the operations. This new pyrochemical process integrated the waste processes closely with the separations processes (Ackerman et al., 1997). While the process was being developed at bench scale in laboratories in Illinois, the Fuel

Conditioning Facility (FCF) at the EBR-II site at INL was being modified to demonstrate the technology with irradiated metal fuel. This demonstration was initially designed to include spent fuel separations, uranium–plutonium–zirconium metal fuel fabrication, and HLW processes (Mc Farlane and Lineberry, 1997). After EBR-II was shutdown in 1994, fuel fabrication was no longer a program objective but the technical feasibility of pyrochemical separations and waste processes remained. The entire process became operational in 1996 and its success was reviewed and verified by a special subcommittee of the National Research Council (Benedict et al., 2002; National research Council, 2000). Since the completion of this demonstration, INL pyroprocessing has been focused on understanding and developing engineering scale separation equipment performance and completing process scale-up of HLW processes. Since 2002, various programs led by the US DOE's Office of Nuclear Energy have funded additional pyroprocessing research and development including two new front-end processes: voloxidation and spent oxide fuel reduction, which allows oxide fuel in addition to metal to be pyroprocessed. This has primarily been motivated by an interest in adapting the process technology to be compatible with oxide fuel from commercial power reactors—though it is widely viewed that aqueous processing is likely to be the most cost-effective approach to reprocessing that fuel.

Several other countries are pursuing pyroprocessing, most notably South Korea led by the Korea Atomic Energy Research Institute. Japan's Central Research Institute of Electric Power Industry (CRIEPI) (Inoue and Tanaka, 1997) has historically been very interested in this technology, but that has taken a hit recently from the country's re-evaluation of nuclear energy in the wake of the Fukushima disaster. As previously mentioned in this chapter, the Research Institute of Atomic Reactors (RIAR, Russia) has been operating and developing a process that uses direct electrorefining of oxide fuels (Vavilov, Kobayashi, and Myochin, 2004)—which is quite similar to the US version of pyroprocessing.

#### 14.6.1 Key Distinctions vis-à-vis Aqueous Processing of Fast Reactor Fuels

The current commercial reprocessing systems are currently based on nitric acid and organic solvents to separate and purify actinides from FPs. The primary difference between pyrochemical process and the aqueous process is the use of high temperatures and the absence of aqueous process solutions. The pyrochemical processes offer several advantages:

1. The pyrochemical equipment is typically more compact and requires less facility space.
2. The use of high-temperature salts and metals eliminates degradation of process materials from radiation damage.
3. The absence of water and organics simplifies nuclear criticality concerns.
4. For metallic fuels, the reprocessing and fuel fabrication processes can be integrated.
5. Because actinides are recovered together, the separation of pure plutonium is not the part of the process.

#### 14.6.2 Head-End Steps

For metal fuels, the preparation of the fuel involves two mechanical separation steps. First, the individual fuel elements are separated from the fuel assembly hardware. These fuel elements are chopped into small segments and loaded directly into baskets that are compatible with electrorefining equipment. The challenge for the head-end process is the

measurement for material control and accountancy. Presently, detailed models of the reactor operating conditions have been shown to provide the adequate prediction of fuel composition which can be verified by taking samples of the segments at specific locations. However, this approach would not be acceptable for IAEA's implementation of international safeguards. Model results cannot be used for input verification, and the timeliness requirements cannot be met for sampling and destructive analysis. For metal fuel, this problem is not currently considered a high priority because no countries are known to be planning to scale-up and commercialize pyroprocessing for metal fuel treatment. However, the situation is different for oxide fuels based on the current plans of South Korea (Republic of Korea). South Korea's spent fuel is all in the oxide form, and they have made tentative plans to use pyroprocessing to treat their used fuel and provide enhanced waste management options. For oxide fuels to be treated via pyroprocessing, another head-end step is needed to convert the fuel from an oxide to a metal. This conversion can be accomplished by either a electrochemical technique (Herrmann, Li, and Simpson, 2007), or by chemical reduction process (McPheeters, Pierce, and Mulcahey, 1997). In either case, the same problem occurs with respect to input accountability. Oxide reduction works much the same as electrorefining with respect to the fuel remaining in a basket. It does not actually dissolve in the salt. Thus, the oxide reduction salt cannot be sampled or monitored to estimate the quantities of fissile nuclear material input into the process. This problem is being addressed currently under the US-ROK Joint Fuel Cycle Study (JFCS). One proposed solution which is being tested is to subject the oxide fuel first to voloxidation to form a powder and then to mix that powder in a ball mill. Voloxidation is a high temperature process in which oxide fuel is further oxidized with air or oxygen to form  $U_3O_8$ , break into small particles, and separate from cladding hulls. It can actually be applied to either aqueous or pyrochemical processing methods. After sufficient particle size homogenization in the ball mill, it has been shown that samples can be highly representative of the total fuel inventory in the mill. This has been tested with simulated spent fuel to date but has not yet been tested with real spent fuel.

### 14.6.3 Electrorefining

Actinides are separated from FPs by electrorefining. This electrochemical process uses a fused salt solution of LiCl–KCl eutectic and actinide chlorides. The spent fuel is electrotransported out of the anode baskets, and an equivalent amount of uranium are deposited on a steel cathode. The uranium is separated from the bulk of the FPs and TRU (Li and Simpson, 2005a; Li et al., 2005b). The TRU with some uranium can be recovered by using a liquid cadmium cathode (Koyama et al., 1997). The alkali, alkaline earth, rare earth, and halide FPs accumulate in the salt phase that is eventually disposed in a ceramic waste. The noble metals and fuel cladding remain in the anode baskets and are converted into a metallic waste form. Much recent work has been done on developing models for electrorefining to aid in scale-up as well as identify conditions for the potential codeposition of uranium and plutonium onto a solid cathode (Choi et al., 2010; Hoover et al., 2010; Zhang, 2013).

### 14.6.4 Cathode Processing

When the actinide products are removed from electrorefiners, the electrorefiner salts cover the metal. The cathodes are processed to distill adhering salt and to consolidate the actinide metals. These salts are recycled to the electrorefiner for further use (Westphal and Mariani, 2000; Westphal et al., 2002). In the case of the liquid cadmium cathode, the cadmium is

distilled and recycled. Recent research in this area has been focused on selected crucible materials that are robust, prevent ingot sticking, and minimize side-reactions with the uranium. Hafnium nitride coated niobium crucibles appeared promising, but problems scaling up their batch size caused INL to switch to zirconia castable liners in graphite crucibles (Westphal et al. 2008).

#### 14.6.5 Refabrication of Metal Fuels

For metal fuel fabrication, the actinide metals are alloyed in an injection casting furnace that melts, mixes the alloy and injects the molten metal into quartz molds. After quick cooling, the quartz mold is removed from the metal pin, which is cut to length and undergoes quality assurance measurements. These pins are placed into new fuel cladding that contains a small amount of metallic sodium, which provides a thermal bond in early irradiation in the nuclear reactor. These fuel elements are welded closed and are ready for the reactor. Recent research in this area has focused on modifying the process to minimize the volatilization of americium, which is a key component in U/TRU recovered for fast reactors and has a high vapor pressure.

#### 14.6.6 HLWs

The treatment of spent nuclear fuel by the IFR process results in two HLW forms: ceramic and metal. The ceramic waste form stabilizes the active FPs (alkali, alkaline earth, and rare earths) and the metal waste consists of stainless steel cladding and, fuel matrix material (zirconium) and noble metal FPs.

The ceramic waste form is a glass-bonded sodalite produced from the thermal conversion of zeolite. The salt is removed from the electrorefiner and initially ground to a powder. This powder is mixed with zeolite at 500 °C and is occluded into the zeolite structure. This salt-loaded zeolite is then mixed with glass frit and is transferred to a furnace where it is heated to 850 °C. During this process, the zeolite is converted to sodalite and a ceramic is formed (Goff et al., 2005; Marsden et al., 2005).

The metal waste is formed by distilling-off residual salts and alloying the cladding hulls with some additional zirconium to form an alloy with nominally 85 wt% steel and 15% zirconium. Both of these HLW forms have been extensively characterized for the proposed geologic repository.

Advanced salt separation methods have been investigated in an attempt to minimize the volume of ceramic waste produced per MT of spent fuel. There is an opportunity to dramatically decrease the amount of waste by selectively returning LiCl–KCl to the electrorefiners. A number of methods for separating FP chlorides from LiCl–KCl have been investigated, including:

- Zone freezing (Williams, Phongikaroon, and Simpson, 2011)
- Zeolite ion exchange (Yoo et al., 2010)
- Oxygen sparging (Cho et al., 2009)
- Lithium drawdown (Simpson et al., 2012)

The feasibility of directly disposing electrorefiner salt (without a waste form) in a salt repository has also been analyzed by researchers at Sandia National Laboratory and INL (Simpson et al., 2013).

## 14.7 Global Status of Used Nuclear Fuel Recycling

By reprocessing, the European and Asian nuclear industries have reduced high-level radioactive waste volumes and quantities and recycled the uranium and plutonium to produce more energy. After recycling about 3% of the original fuel quantity remains as HLW. UNF contains recoverable materials (uranium and plutonium) used to manufacture new fuel rods for nuclear reactors. UNF recycling conserves natural resources by extending the life of available uranium resources, as well as saving substantial amounts of oil that would otherwise be used for energy production. France, Japan, and the United Kingdom currently recycle their UNF and return the usable uranium and plutonium to the fuel cycle. Other developing countries such as China are planning to recycle their UNF. China has planned to reprocess its used reactor fuel since the start of its nuclear power program in the late 1980s. By 2020, it is estimated that about 1,000 t of UNF will have amassed and over 40 reactors could be in operation and adding to the total in China. Consequently, in April 2013, China has signed an agreement with AREVA to recycle UNF in a plant supplied by AREVA. The recycling plant will have a capacity to handle 800 t of UNF per year (World Nuclear News, October, 2013).

### 14.7.1 United States

In the United States, no civil reprocessing plants are now operating, though three have been built. The first, a 300 t yr<sup>-1</sup> plant at West Valley, New York, was operated successfully from 1966 to 1972. However, escalating regulation required plant modifications which were deemed uneconomic, and the plant was shut down. The second was a 300 t yr<sup>-1</sup> plant built at Morris, Illinois, incorporating new technology which, although proven on a pilot-scale, failed to work successfully in startup testing of the production plant. The third was a 1500 t yr<sup>-1</sup> plant at Barnwell, South Carolina, which was aborted due to a change in government policy which ruled out all US civilian reprocessing as one facet of the US non-proliferation policy. In all, the United States has more than 250 plant-years of reprocessing operational experience, the vast majority being at government-operated defense plants since the 1940s (McFarlane, 2004).

The United States may have considerable reprocessing experience, but the application of this to civilian-used fuel has been frustrated by political sensitivities motivated by proliferation concerns based on an understanding that reactor-grade plutonium is usable for weapons. Civil reprocessing was stopped in 1977.

From 2010 to 2012, the *Blue Ribbon Commission on America's Nuclear Future* ("BRC" or "Commission") conducted a comprehensive review and recommended a plan of action for the management and disposal of the nation's UNF and high-level radioactive waste, also referred to as the back-end of the nuclear fuel cycle. Representative Lee Hamilton and General Brent Scowcroft, two distinguished individuals with decades of public service and governing experience, co-chaired the Commission and led a panel of leading scientists, nuclear energy experts, industry leaders, and former elected officials. The BRC's report and recommendations provided a starting point for a strategy for the United States for the management and disposal of the nation's UNF and high-level radioactive waste (DOE, 2013).

With the appropriate authorizations from Congress, the US Administration currently plans to implement a program over the next 10 years that:

- Sites, designs and licenses, constructs and begins operations of a pilot interim storage facility by 2021 with an initial focus on accepting UNF from shut-down reactor sites
- Advances toward the siting and licensing of a larger interim storage facility to be available by 2025 that will have sufficient capacity to provide flexibility in the waste management system and allows for the acceptance of enough UNF to reduce expected government liabilities
- Makes demonstrable progress on the siting and characterization of repository sites to facilitate the availability of a geologic repository by 2048 (DOE, 2013)

Currently, the US DOE's Office of Nuclear Energy continues to fund research at national laboratories and universities on advanced nuclear energy and fuel cycles. This continues even in the absence of specific plans to demonstrate new technologies at commercial scale.

#### 14.7.2 France

France has 58 nuclear reactors operated by Electricité de France (EdF) with a total capacity of 63 GWe, supplying 417 TWh in 2008 with high levels of reliability and safety, 78% of the total generated there. The 58 PWR are designed on standardized PWR reactor series (34 × 900 MW-class units, 20 × 1300 MW-class units, and 4 × 1450 MW-class units) (Carre and Delbecq, 2009).

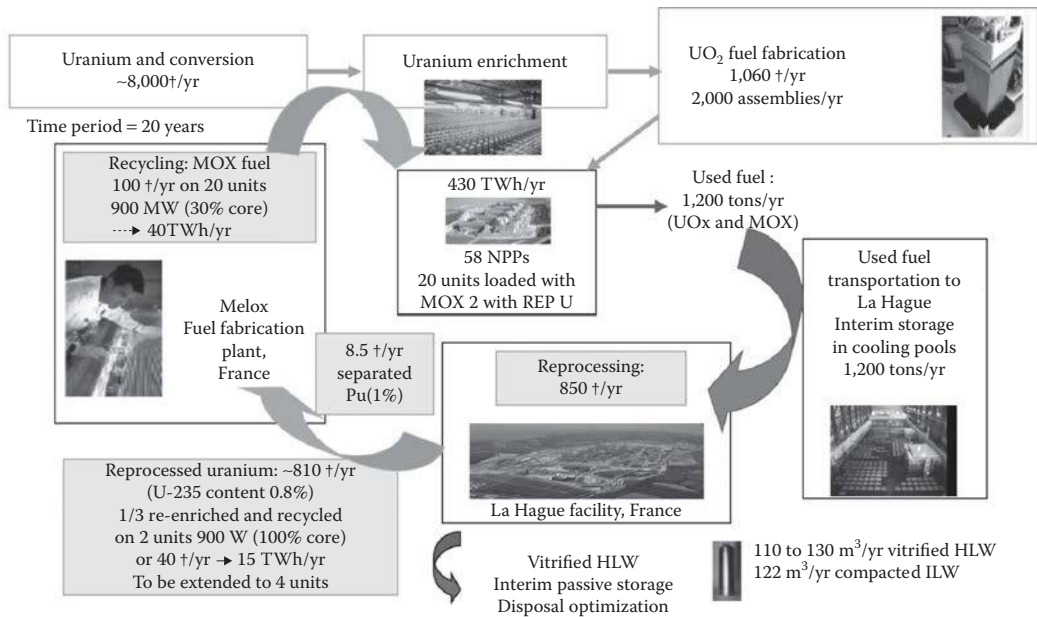
The present situation is due to the French government deciding in 1974, just after the first oil shock, to rapidly expand its nuclear power capacity. This decision was taken in the context of France having substantial heavy engineering expertise but few indigenous energy resources. Nuclear energy, with the fuel cost being a relatively small part of the overall cost, made good sense in minimizing imports and achieving greater energy security. As a result of the 1974 decision, France now claims a substantial level of energy independence and almost the lowest cost electricity in Europe.

Since opening in 1966, the La Hague plant near Cherbourg (Figure 14.12) has reprocessed fuel from three reactor types: gas-cooled, fast breeder, and light water. Since 1987, it has been dedicated to reprocessing French light reactor fuel, as well as fuel from Japan, Switzerland, Germany, Belgium, and the Netherlands. Recycling of oxide fuels is performed at the La Hague site since 1976; two 800 t yr<sup>-1</sup> plants are in operation. EdF has made provision to store reprocessed uranium (RepU) for up to 250 years as a strategic reserve.

A quantity of about 850 t of UO<sub>2</sub> spent fuel is reprocessed annually in France according to Carre and Delbecq (2009), as a result of trade-off between optimizing plutonium recycling and power generation. Thus, about 100 t of MOX fuel are fabricated annually and recycled once into twenty 900 MW licensed reactors (30% of core), which contributes to about 10% of nuclear production. In the same way, reprocessed uranium (REPU) recovered from spent fuel is used today to fuel two 900 MW reactors, which may be expanded depending on the associated economic benefit (Figure 14.32).

Even though the use of RepU in a 900 MWe power plants has been demonstrated by EdF, it is currently uneconomic due to conversion costing three-times as much as that for fresh uranium, and enrichment needing to be separate because of U-232 and U-236 impurities (the former gives rise to gamma radiation, the latter means higher enrichment is required).

In May 2006, the EdF board approved construction of a new 1650 MWe Evolutionary Pressurized reactor (EPR) unit at Flamanville, Normandy, adjacent to two existing PWRs.



**FIGURE 14.32**

The French industrial nuclear fuel cycle. (Adapted from Carre, F., and J.M. Delbecq. 2009. Overview on the french nuclear fuel cycle strategy and transition scenario studies. *Proceedings of Global 2009*, Paris France, Paper 9439, September 6–11, 2009.)

Construction work began in December 2007. The reactor was originally expected to start commercial operation in 2013, but due to delays is now expected to start up in 2016. The reactor vessel has been delivered in October 2013 to the construction site. An advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) is being designed by French Alternative Energies and Atomic Energy Commission (CEA) and its industrial partners (Bess et al., 2013) and it represents a technology platform that France would like to have available for use in around 2040. The goal is to build a reactor for demonstrating innovative design choices so that the fast neutron reactors (FNR) can meet the criteria for the fourth generation with:

- Enhanced recycling of nuclear material
- Robust safety demonstration that gives ASTRID the same safety level as the EPR
- High level of availability and reliability for operators
- Compliance with the requirements for anti-proliferation measures

Operating with a sodium-cooled fast neutron core, ASTRID is expected to produce approximately 600 MW of electricity. Before the construction of a first-off commercial unit, a demonstration facility is needed to test innovations with respect to previous FNRs. The FNR greatly improves the amount of energy derived from depleted or reprocessed natural uranium, enables plutonium to be used and recycled several times and can recycle minor actinides if needed. Such reactors are currently being built or are on the drawing board in India, Russia, China, and Japan.



### 14.7.3 Japan

Japan started its nuclear research program in 1954, with ¥230 million being budgeted for nuclear energy. The Atomic Energy Basic Law, which strictly limits the use of nuclear technology to peaceful purposes, was introduced in 1955. The law aims to ensure that three principles, democratic methods, independent management, and transparency, are the basis of nuclear research activities, as well as promoting international cooperation. Inauguration of the AEC in 1956 promoted nuclear power development and utilization. Several other nuclear energy-related organizations were also established in 1956 under this law: the Science & Technology Agency; Japan Atomic Energy Research Institute (JAERI) and the Atomic Fuel Corporation (renamed PNC in 1967, see in the following).

Japan imported its first commercial nuclear power reactor from the UK. Tokai-1, a 160 MWe gas-cooled (Magnox) reactor built by GEC. It began operating in July 1966 and continued until March 1998. Prior to the earthquake and tsunami of March 2011, and the nuclear disasters that resulted from it, Japan generated 30% of the country's electricity from its 50 nuclear reactors. The 2011 earthquake and tsunami caused the failure of cooling systems at the Fukushima I Nuclear Power Plant on March 11 and resulted in the closure of many of Japan's nuclear plants for safety inspections. The last of Japan's 50 reactors (Tomari-3) went offline for maintenance on May 5, 2012, leaving Japan completely without nuclear-produced electrical power for the first time since 1970.

Japan has built a major (Circoncione and Wolfsthal, 2004) (capacity of 800 t yr<sup>-1</sup>) plant at Rokkasho-Mura while having had most of its used fuel reprocessed in Europe; meanwhile it had a small (capacity of 90 t yr<sup>-1</sup>) plant operating at Tokai Mura (McFarlane, 2004). Building on demonstrated reprocessing designs from France and fitted with state-of-the-art inspection systems, the huge new facility will be able to process 800 t of spent fuel annually, recovering up to 8 t of plutonium. The facility is designed, however, not to release separated plutonium, but to internally combine the plutonium with recovered uranium (MOX). The mixture will be stored and eventually turned into fuel for use in Japanese LWRs. Even though none of the Japan's 50 commercial power reactors have used MOX fuel, 20 MOX fuel assemblies produced by AREVA at its Melox plant in France have been unloaded in June 2013 at Kansai Electric Power Company's (Kepeco's) Takahama nuclear power plant in Fukui Prefecture from two ships—the *Pacific Heron* and the *Pacific Egret*—marking the completion of the fifth such MOX shipment to Japan. Japanese fuel is currently reprocessed in Europe and some 38 t of separated plutonium are already in storage in Europe and Japan. The reprocessing facility is colocated with a uranium-enrichment plant and facilities for the burial of 3 million of barrels of low-level radioactive waste from power reactors, the temporary (30–50 years) storage of thousands of canisters of high-level vitrified waste from past reprocessing operations and the storage of thousands of used fuel rods awaiting reprocessing. The vitrified waste storage center at Rokkasho-Mura can hold 1440 canisters and is already more than half full, with 1100 canisters still to be returned from France and the United Kingdom (Cirincione, 2004).

### 14.7.4 Russia

In the 1950s and 1960s, Russia seemed to be taking impressive steps to contest world leadership in the civil development of nuclear energy. It had developed two major reactor designs: one from military plutonium production technology (the light water-cooled graphite moderated reactor—RBMK) and one from naval propulsion units, very much

as in the United States (the VVER series; pressurized, water cooled, and moderated). Russia's first nuclear power plant, and the first in the world to produce electricity, was the 5 MWe Obninsk reactor in 1954. Russia's first two commercial-scale nuclear power plants started up in 1963–64, then in 1971–73 the first of today's production models were commissioned. An ambitious plant, Atommash, to mass produce the latter design was taking shape near Volgodonsk, the construction of numerous nuclear plants was in hand, and the country had many skilled nuclear engineers. Atommash produced only three reactor pressure vessels, instead of the eight per year intended. By the mid-1980s, Russia had 25 power reactors in operation. Then Chernobyl put the whole nuclear industry into a long standby.

Between the 1986 Chernobyl accident and mid-1990s, only one nuclear power station was commissioned in Russia, the 4-unit Balakovo, with unit 3 being added to Smolensk. Economic reforms following the collapse of the Soviet Union meant an acute shortage of funds for nuclear developments. By the late 1990s, exports of reactors to Iran, China, and India were negotiated and Russia's stalled domestic construction program was revived as far as funds allowed.

Around year 2000, nuclear construction revived and Rostov-1 (now known as Volgodonsk-1), the first of the delayed units, started up in 2001, joining 21 GWe already on the grid. This greatly boosted morale in the Russian nuclear industry. It was followed by Kalinin-3 in 2004.

By 2006, the government's resolve to develop nuclear power had firmed, and there were projections of adding 2–3 GWe per year to 2030 in Russia as well as exporting plants to meet world demand for some 300 GWe of new nuclear capacity in that time frame.

RT-1 was commissioned in 1977 to reprocess spent fuel from VVER-440, BN-350, BN-600, research, and naval propulsion reactors. Most of the feed is from VVER-440 reactors. This is the only Russian facility that reprocesses spent power reactor fuel. Starting in 1978, fuel containing uranium reprocessed at Mayak has been used in BN-350, BN-600, and RBMK reactors, and more recently in VVER-1000s. The plant's nominal reprocessing capacity (based on spent fuel from the VVER-440 reactors) is 400 t of spent fuel per year. In 1991 and 1992, respectively, 170 MT and 120 MT of spent fuel were reprocessed at RT-1. According to Western estimates, RT-1 reprocessed 124 MTHM in 1993 and 160 MTHM in 1994. As of November 1995, RT-1 reprocessed over 3000 MT of spent fuel, corresponding to about 30 MT of reactor-grade plutonium. This plutonium is stored at Mayak in the form of powdered plutonium dioxide. Almost all uranium extracted from the spent fuel is sent to the Ust-Kamenogorsk fuel fabrication plant in Kazakhstan; very little remains at the plant. However, some VVER-440 uranium solution (containing typically 1.3% U-235) is blended with HEU to produce uranium with an enrichment level of about 2.0% for RBMK reactor fuel. The RT-1 facility is made up of a spent fuel storage pool, three chopping-dissolution process lines, and a modified PUREX process. High-level liquid radioactive waste from the reprocessing is converted into a glass-like material at the Vitrification Plant and then stored in special containers (Bukharin, 1996; Hibbs, 1996). A tender for the construction of a fuel reprocessing demonstration plant at Zheleznogorsk for the reprocessing of UNF has been announced in February 2013 by Russia's state nuclear corporation Rosatom (World Nuclear News, February 2013). A pilot demonstration center (PDC) will be built on the site of the Mining and Chemical Combine (MCC) at Zheleznogorsk near Krasnoyarsk, Siberia. Under the terms of the tender, the facility must be in operation by December 10, 2015. This center will initially be used to develop and test new technologies for reprocessing used fuel from VVER-1000 reactors. The PDC is expected to later develop

technology for the reprocessing of used fuel from fast reactors. The PDC is expected to provide the necessary data for the design of a large-scale reprocessing plant based on innovative technology. This will also be located at MCC's Zheleznogorsk site and could be operational by around 2025–30.

#### 14.7.5 United Kingdom

The United Kingdom has currently nine operating reactors. In March 2013, the British government published a Nuclear Industrial Strategy which in part stated that the nuclear industry has plans for about 16 GWe of new nuclear power stations by 2030, which is at least 12 new nuclear reactors at five sites. The Sellafield site, close to the village of Seascale on the coast of the Irish Sea in Cumbria, England, is home to a wide range of interdependent nuclear facilities and operations. These range from hazard and risk reduction, decommissioning, reprocessing, and nuclear waste management. Facilities at the site include the THORP nuclear fuel reprocessing plant, which deals with fuel from British Advanced Gas-cooled Reactors (AGR) and Light Water Reactors (LWR) from around the world, and the Magnox nuclear fuel reprocessing plant which reprocesses spent Magnox fuel in its B205 plant. It is also the site of the remains of Calder Hall, the world's first nuclear power reactor, commissioned in 195, now being decommissioned, as well as some other older nuclear facilities at Windscale.

In 2008, it was expected that Magnox reprocessing would be completed in 2012, which was then revised to 2016. The current target for shutting down reprocessing of Magnox fuel is set optimistically at 2018. Meeting this target will be possible in the event that Magnox reprocessing throughput can be maintained at 664 t per year until 2018. Should throughput drop to 450 t yr<sup>-1</sup>, then the closure of reprocessing would be further delayed to 2020.

The thermal oxide reprocessing plant (THORP) was commissioned in 1994 and the corresponding MOX fuel plant in 2001 (McFarlane, 2004). All the fuel from Britain's first- and second-generation nuclear stations has been chemically separated into uranium, plutonium and radioactive waste at Sellafield. Only spent fuel from one station, the pressurized water reactor at Sizewell in Suffolk, has not been reprocessed. The leak of 20 t of plutonium and uranium liquid mixed with nitric acid discovered in April 2005 at Sellafield's controversial Thorp reprocessing facility was undetected for nine months. The incident led to a three-year outage of the Thorp plant and hence affected its reprocessing throughput and output of highly radioactive liquor. Though the plant still has a number of foreign reprocessing orders to fill, it is expected to shut down by 2018.

#### 14.7.6 India

Nuclear power for civil use is well established in India. Its civil nuclear strategy has been directed toward complete independence in the nuclear fuel cycle, necessary because it is excluded from the 1970 Nuclear Nonproliferation Treaty (NPT) due to it acquiring nuclear weapons capability after 1970.

This self-sufficiency extends from uranium exploration and mining through fuel fabrication, heavy water production, reactor design and construction, to reprocessing and waste management. It has a small fast breeder reactor and is building a much larger one. It is also developing technology to utilize its abundant resources of thorium as a nuclear fuel.

The Atomic Energy Establishment was set up at Trombay, near Mumbai, in 1957 and renamed as Bhabha Atomic Research Centre (BARC) ten years later. Plans for building the first Pressurized Heavy Water Reactor (PHWR) were finalized in 1964, and this prototype, Rawatbhata-1, which had Canada's Douglas Point reactor as a reference unit, was built as a collaborative venture between Atomic Energy of Canada Ltd and NPCIL. It started up in 1972 and was duplicated. Subsequent indigenous PHWR development has been based on these units.

Electricity demand in India has been increasing rapidly, and the 534 billion kWh produced in 2002 was almost double the 1990 output, though still representing only 505 kWh per capita for the year. This per capita figure is expected to almost triple by 2020, with 6.3% annual growth. Coal provides over half of the electricity at present, but reserves are limited.

As of 2012, India has 20 nuclear reactors in operation in six nuclear power plants, generating 4,780 MW, while seven other reactors are under construction and are expected to generate an additional 5,300 MW. India has a flourishing and largely indigenous nuclear power program and expects to have 14,600 MWe nuclear capacity online by 2020. It aims to supply 25% of electricity from nuclear power by 2050 (World Nuclear Association, November 2013).

Used fuel from the civil Pressurized Heavy Water Reactors (PHWRs) is reprocessed by Bhabha Atomic Research Centre (BARC) at Trombay, Tarapur, and Kalpakkam to extract reactor-grade plutonium for use in the fast breeder reactors. Small plants at each site were supplemented by a new Kalpakkam plant of some 100 t yr<sup>-1</sup> commissioned in 1998, and this is being extended to reprocess (Fast Breeder Test Reactor) FBTR carbide fuel. Apart from this all reprocessing uses the PUREX process. A new 100 t yr<sup>-1</sup> plant at Tarapur was opened in January 2011, and further capacity is being built at Kalpakkam. As of early 2011, capacity was understood to be 200 t yr<sup>-1</sup> at Tarapur, 100 t yr<sup>-1</sup> at Kalpakkam, and 30 t yr<sup>-1</sup> at Trombay, total 330 t yr<sup>-1</sup>, all related to the indigenous PHWR program and not under international safeguards (World Nuclear Association, November 2013). India will reprocess the used PWR fuel from the Kudankulam and other imported reactors and will keep the plutonium. This will be under IAEA safeguards, in new plants. In April 2010, it was announced that 18 months of negotiations with the United States had resulted in agreement to build two new recycling plants to be under IAEA safeguards, likely located near Kalpakkam and near Mumbai—possibly Trombay. In July 2010, an agreement was signed with the United States to allow the reprocessing of US-origin fuel at one of these facilities. Later in 2010, the Atomic Energy Commission (AEC) announced the commencement of engineering activities for setting up of an integrated nuclear recycle plant with facilities for both reprocessing of used fuel and waste management.

To close the fast breeder reactor (FBR) fuel cycle, a fast reactor fuel cycle facility (FRFCF) has long been planned, with construction originally to begin in 2008 and operation to coincide with the need to reprocess the first prototype fast breeder reactor (PFBR) fuel. The PFBR and the next four FBRs to be commissioned by 2020 will use oxide fuel. After that, it is expected that metal fuel with higher breeding capability will be introduced and burn-up is intended to increase from 100 to 200 GWd t<sup>-1</sup>.

In 2003, a facility was commissioned at Kalpakkam to reprocess mixed carbide fuel using an advanced PUREX process. In 2010, the AEC said that used mixed carbide fuel from the fast breeder test reactor (FBTR) with a burn-up of 155 GWd t<sup>-1</sup>

was reprocessed in the compact reprocessing facility for advanced fuels in lead cells (CORAL). Thereafter, the fissile material was re-fabricated as fuel and loaded back into the reactor, thus “closing” the fast reactor fuel cycle. In July 2013, the government approved construction of the FRFCF at Kalpakkam (\$1.61 billion). Work is expected to start in 2013, initially under the auspices of the Indira Gandhi Centre for Atomic Research (IGCAR). It will serve the PFBR nearby, and will have capacity to cater for three such reactors.

#### 14.7.7 China

As of September 2013, Mainland China has 17 nuclear power reactors in operation, 30 under construction, and more about to start construction. Additional reactors are planned, including some of the world’s most advanced, to give a fourfold increase in nuclear capacity to at least 58 GWe by 2020, then possibly 200 GWe by 2030, and 400 GWe by 2050 (World Nuclear Association, September 2013). China has become largely self-sufficient in reactor design and construction, as well as other aspects of the fuel cycle, but is making full use of western technology while adapting and improving it. China’s policy is for closed fuel cycle and consequently has chosen to recycle UNF. For the moment, China relies increasingly on imported uranium as well as conversion, enrichment, and fabrication services from other countries.

China National Nuclear Corporation (CNNC) Ruining Technology Co. Ltd was set up by CNNC in November 2011 to industrialize used fuel reprocessing technology and mixed-oxide (MOX) fuel production to close the fuel cycle. A pilot reprocessing plant using the PUREX process was constructed from 2006 at Lanzhou Nuclear Fuel Complex, and completed hot commissioning in 2010. It will reprocess about 50 t of used fuel over 2013–15. An initial commercial reprocessing plant based on this, of 400 t yr<sup>-1</sup>, is planned to operate about 2017. A large (800–1000 t yr<sup>-1</sup>) commercial reprocessing plant based on indigenous advanced technology was planned to follow and begin operation about 2020, but plans for this are not firm and the 800 t yr<sup>-1</sup> AREVA project could displace it, despite being older technology. A second 800 t yr<sup>-1</sup> plant will follow.

In November 2007, AREVA and CNNC signed an agreement to assess the feasibility of setting up a reprocessing plant for used fuel and a mixed-oxide (MOX) fuel fabrication plant in China, representing an investment of €15 billion. The 800 t yr<sup>-1</sup> reprocessing plant would apparently be in Jinta county, north of Jiayuguan in Gansu province, employing proven French technology and operated by AREVA. Design, construction, and commissioning were expected to take 10 years from 2010. In November 2010, an industrial agreement on this was signed, which AREVA said was “the final step toward a commercial contract” for the project. In April 2013, a further agreement was signed with AREVA, setting out the technical specifications for the 800 t yr<sup>-1</sup> plant. The target date for operation is 2025. The China Institute of Atomic Energy (CIAE) envisages an industrial reprocessing plant of about 1000 t yr<sup>-1</sup> in operation from about 2021.

Technology for recycling uranium recovered from UNF from Chinese PWRs for use in the Qinshan Phase III Candu units is being developed.

In October 2010, GDF Suez Belgian subsidiary Tractabel, with Belgonucleaire, and the nuclear research center SCK-CEN signed an agreement with CNNC to build a pilot MOX fuel fabrication plant in China. Belgium has experience in MOX fuel development and production dating back to 1960.

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## 14.8 Economics

The relative economics of reprocessing waste disposal and interim storage direct disposal has been the focus of much debate over the past decades. Studies have modeled the total fuel cycle costs of a reprocessing recycling system based on one time recycling of plutonium in existing thermal reactors (as opposed to the proposed fast breeder reactor cycle) and compare this with the total costs of an open fuel cycle with direct disposal of UNF. The range of results produced by these studies is very wide, but all are agreeing that under current technological recycling conditions, the UNF recycle is more costly than direct disposal (Bunn et al., 2003; Orszag, 2007). Future energy systems will only develop if they meet the criteria of sustainability and cost-effectiveness. On the other hand, if we approach the economics at another angle, and we examine the need of uranium resources and also plutonium demand for fast nuclear reactors, the economics change drastically. The current world use of uranium for nuclear power is of the order of 70,000 MT per year. This is used to generate 365 GW of electrical power (Kok, 2013). At current nuclear level production, there is enough proven uranium resources for centuries to come (Carre and Delbecq, 2009). But if we consider the amount of uranium that would be needed to fuel nuclear power plants in construction worldwide (Table 14.14), and the fact that uranium conventional resources are limited (around 15 MT, according to the well-known IAEA and OECD Red Book), it appears that these resources could be at stake around 2050 (Carre and Delbecq, 2009). Uranium prices will increase drastically and recycling will make a strong economic difference. Not only recycling will allow preserving 15%–20% of the uranium natural resource, it will also produce the necessary plutonium to be used in fast reactor systems. As evidenced by Poinssot et al. (2012), fast reactor systems combined with plutonium recycling are the key to gain two orders of magnitude in the natural resource consumption efficiency.

One of the benefits of UNF recycling as depicted in Figure 14.33 is that the radiotoxicity level of recycled UNF implying the combination of partitioning of minor actinides and fast reactor technology will reach the uranium ore relative radiotoxicity level in 300 years as opposed to the 300,000 years required by a direct disposal of UNFL. Thus, the difference between the costs of recycling and direct disposal will be reduced, because it will take only 300 years to reach an acceptable level of environmental radiotoxicity.

In mid-2006, a report by the Boston Consulting Group (Peters et al., 2006) for AREVA and based on proprietary AREVA information showed that recycling used fuel in the United States using the COEX aqueous process would be economically competitive with direct disposal of used fuel. A \$12 billion, 2500 t yr<sup>-1</sup> plant was considered, with total capital expenditure of \$16 billion for all related aspects. This would have the benefit of greatly reducing demand on space of a repository with a capacity of 70,000 MTHM.

Boston Consulting gave four reasons for reconsidering the United States used fuel strategy which has applied since 1977:

- Cost estimates for direct disposal at a nuclear repository rose sharply and capacity was limited (even if doubled).
- Increased US nuclear generation, potentially from 103 to 160 GWe.
- The economics of recycling and associated waste disposal have improved.
- There is now a lot of experience with civil recycling as demonstrated by several countries.

**TABLE 14.14**

Worldwide Nuclear Power Reactors Under Construction and Proposed

Country	Plant, Type	Nuclear Capacity
Argentina	Atucah III-ACR 1000	1500 MWe
	Ambalse - PHWR	600 MWe
Bangladesh	Ruppur Nuclear Power Plant – VVER 1000	2000 MWe
Belarus	Ostrovets site AES-2006 model VVER pressurized water reactor	1200 MWe
Brazil	Angra 3 PWR	1405 MWe
	Northeast, Pernambuco, 4 PWR	6000–6600 MWe
	Southeast, Minas Gerais, 4 PWRs	4000–6000 MWe
Bulgaria	Kozloduy 7, VVER-1000 (AES-92) or western PWR	1000–1200 MWe
Czech republic	Temelin 3	1200 MWe
	Temelin 4	1200 MWe
China	Hongyanhe units 2-4, CPR-1000	3 × 1119 MWe
	Nindge units 2-4, CPR 1000	3 × 1089 MWe
	Fuqing units 1&2, CPR 1000	2 × 1087 MWe
	Yangjiang units 1-4, CPR 1000, CPR 1000 +	4 × 1086 MWe
	Fangjiashan/Wanjianshan units 1&2, CPR 1000	2 × 1087 MWe
	Sanmen units 1&2, AP 1000	2 × 1250 MWe
	Haiyang unist 1&2, Ap 1000	2 × 1250 MWe
	Taishan units 1&2, EPR	2 × 1750 MWe
	FGangchenggang units 1&2, CPR 1000	2 × 1080 MWe
	Changjiang units 1&2, CNP 600	2 × 650 MWe
	Fuqing units 3&4, ACP 1000	2 × 1100 MWe
	Hongyanhe units 5&6, ACPR 1000 or AP 1000	2 × 1080 MWe
	Yangjiang units 5&6, ACPR 1000	2 × 1080 MWe
	Fanchenggang units 3&4, ACPR 1000 or AP 1000	2 × 1250 MWe
	Ningde units 5&6, ACPR 1000	2 × 1080 MWe
	Xudabo/Xudabu units 1&2, AP 1000	2 × 1250 MWe
	Sanmen units 3&4, Ap 1000	2 × 1250 MWe
	Haiyang units 3&4, AP 1000	2 × 1250 MWe
	Lufeng units 1&2, AP 1000	2 × 1250 MWe
	Fanchenggang units 5 & 6, AP 1000	2 × 1080 MWe
	Bailong units 1&2, AP 1000	2 × 1250
	Huizhou units 1&2, AP 1000	2 × 1250
Shidaowan units 1&2, CAP 1400 or AP 1000	2 × 1400	
Tianwan units 5&5, VVER 1000 or AP 1000	2 × 1080	
Taishan units 3&4, EPR	2 × 1750	
Changjiang units 3&4, CNP 600	2 × 650	
Finland	Olkiluoto3, EPR	600 MWe
	Olkiluoto 4, EPR or ABWR or APR 14000 or EU-APWR or ESBWR	1450–1650 MWe
France	Hanhikivi-1, VVER or ABWR	1150–1300 MWe
	Flmanville 3, EPR	1750 MWe
	AREVA-MHI Atmea-1	1100 MWe

*(Continued)*

TABLE 14.14 (Continued)

Worldwide Nuclear Power Reactors Under Construction and Proposed

Country	Plant, Type	Nuclear Capacity
India	Kudankulam 2, VVER 1000	1000 MWe
	Kalpakkam PFBR, FBR	500 MWe
	Kakrapar 3, PHWR,	700 MWe
	Kakrapar 4, PHWR,	700 MWe
	Rajasthan 7, PHWR	700 MWe
	Rajasthan 8	700 MWe
Russia	Beloyarsk 4, BN 800 FBR	880 MWe
	Novovoronezh II -1, VVER 1200	1200 MWe
	Leningrad II -1, VVER 1200	1200 MWe
	Novovoronezh II -2, VVER 1200	1200 MWe
	Rostov 3, VVER 1000	1100 MWe
	Leningrad II -2, VVER 1200	1200 MWe
	Rostov 4, VVER 1000	1100 MWe
	Baltic -1, VVER 1200	1194 MWe
Saudi Arabia	18 Nuclear reactor, ABWR or ESBWR, or AP1000, or EPR	123 GWe
Slovakia	Mochovce 3, V-213	471 MWe
	Mochovce 4, V-213	471 MWe
	Bohunice New Block	1200–1750 MWe
	Keccerovce	1200 MWe
South Africa	Thyspunt, PWR	9600 MWe
South Korea	Shin Wolsong 2, OPR 1000	1050 MWe
	Shin Kori 3, APR 1400	1455 MWe
	Shin Kori 4, APR 1400	1455 MWe
	Shin Hanul/Ulchin 1, APR 1400	1455 MWe
	Shin Hanul/Ulchin 2, APR 1400	1455 MWe
	Shin Kori 5, APR 1400	1455 MWe
	Shin Kori 6, APR 1400	1455 MWe
	Shin Hanul/Ulchin 3, APR 1400	1455 MWe
	Shin Hanul/Ulchin 4, APR 1400	1455 MWe
	Shin Kori 7, APR 1400	1455 MWe
Turkey	Shin Kori 8, APR 1400	1455 MWe
	Akkuyu 1, VVER 1200	1200 MWe
	Akkuyu 2, VVER 1200	1200 MWe
	Akkuyu 3, VVER 1200	1200 MWe
	Akkuyu 4, VVER 1200	1200 MWe
	Sinop 1, Atmea 1	1150 MWe
	Sinop 1, Atmea 1	1150 MWe
	Sinop 1, Atmea 1	1150 MWe
Sinop 1, Atmea 1	1150 MWe	
UAE	Barakah 1, APR 1400	1400 MWe
	Barakah 2, APR 1400	1400 MWe
	Barakah 3, APR 1400	1400 MW MWe e
	Barakah 4, APR 1400	1400
Ukraine	Khmelnitski3, V392	1000 MWe
	Khmelnitski4, V392	1000 MWe

(Continued)

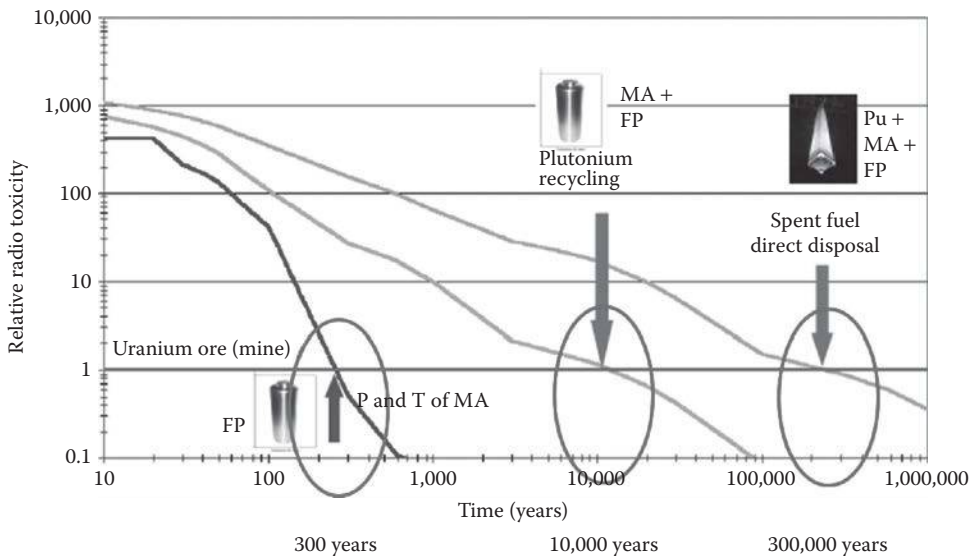


**TABLE 14.14 (Continued)**

Worldwide Nuclear Power Reactors Under Construction and Proposed

Country	Plant, Type	Nuclear Capacity
United Kingdom	Hinkley Point C-1, EPR	1670 MWe
	Hinkley Point C-2, EPR	1670 MWe
	Sizewell C-1, EPR	1670 MWe
	Sizewell C-2, EPR	1670 MWe
	Oldbury B, ABWR × 2 or 3	2760–4140 MWe
	Wylfa B, ABWR × 2 or 3	2760–4140 MWe
	Moorside, AP1000 × 3	Up to 3600 MWe
USA	Vogtle 3, AP1000	1250 MWe
	Vogtle 4, AP 1000	1250 MWe
	V.C.Summer 2, SC, Ap 1000	1250 MWe
	V.C.Summer 3, SC, AP 1000	1250 MWe
Vietnam	Ninh Thuan 1-1, VVER 1000	1060 MWe
	Ninh Thuan 1-2, VVER 1000	1060 MWe
	Ninh Thuan 1-3, VVER 1000	1060 MWe
	Ninh Thuan 1-4, VVER 1000	1060 MWe
	Ninh Thuan 2-1 Japanese Gen III	1000 MWe
	Ninh Thuan 2-2 Japanese Gen III	1000
	Ninh Thuan 2-3 Japanese Gen III	1000 MWe
	Ninh Thuan 2-4 Japanese Gen III	1000 MWe

Source: World Nuclear Association. December 2013. Available online at: <http://www.world-nuclear.org/info/Country-Profiles/Countries-A-Z/>.



**FIGURE 14.33**

Fast reactors: waste minimization. (Adapted from Bouchard, J. 2008. The GEN IV roadmap: The way towards sustainable development of nuclear energy. International Energy Agency Workshop on Energy Technology, Energy Technology Roadmaps, May 15–16, 08, IEA, Paris, France. Available online at: <http://www.bing.com/search?q=J.+Bouchard%2C+IEA%2C+Paris+2008&src=IE-SearchBox&FORM=IE8SRC.>)

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# 15

## *HWR Fuel Cycles*

Paul J. Fehrenbach and Alistair I. Miller

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One of the hallmarks of the heavy water reactor (HWR) is its high neutron economy. High neutron economy is a necessity for utilizing natural uranium equivalent (NUE) fuel but results in a reactor that has a very high degree of fuel-cycle flexibility. This enables a country, or utility, to optimize its nuclear fuel cycle based on its own unique circumstances. Thorium—as an alternative to uranium—has been given some emphasis because thorium burns particularly well in HWRs and has the potential to become an important nuclear fuel, initially regionally but ultimately as a huge additional energy resource as once-through uranium cycles are constrained by the availability of low-cost uranium ore. This chapter is a brief introduction to HWR fuel-cycle technology, which draws heavily on the more extensive treatment provided in an earlier IAEA report (IAEA Tec Report 407, 2002).

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## 15.1 Introduction

The nuclear fuel cycle starts with the ore being extracted from the earth and follows it through processing and use until a final waste form is placed back for permanent disposal. Both uranium and thorium exist in nature as minerals that can be mined; however, uranium will be used in this discussion because the use of thorium has not been extensively

developed. The so-called front end of the fuel cycle includes the processes required to prepare the uranium into a form that can be used as fuel in nuclear reactors. The processes include the following:

- Mining, milling, and conversion—See Chapter 10.
- Enrichment—See Chapter 11.
- Fuel Fabrication—See Chapter 12.

After the fuel is irradiated in the reactor, there are two divergent paths proposed to handle the used fuel. The first is to package it and dispose of it in a geologic repository as high-level nuclear waste. The alternative is to reprocess it to extract the remaining uranium and also recover the plutonium that was produced during irradiation. The waste from this process is the fission products. As seen in Chapter 14, there are several processes that can be used to preserve the useful uranium and plutonium.

The remainder of this chapter will briefly examine the recycle of used fuel in LWRs and in fast reactors and then provide a detailed discussion of a wide variety of fuel cycles proposed for HWRs.

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## 15.2 Fuel Resource Utilization

Beginning with 1000 kg of uranium ore containing 1% uranium  $U_3O_8$  means that there is only 8.5 kg of natural uranium containing 0.7%  $U^{235}$  (59.5 g of  $U^{235}$ ) in the material going to the enrichment process. Enriching that 8.5 kg to 5%  $U^{235}$  will yield about 0.85 kg of reactor grade fuel (see Table 11.2) and leave 7.65 kg of depleted uranium (DU). So, at the completion of the front end of the fuel cycle, only 10% of the original uranium in the ore makes it into the reactor as reactor fuel.

On the basis of the operation of a typical light water reactor (PWR or BWR), the process of producing electricity through the fission process consumes only about 5% of the contained uranium. The specific amount is dependent on the specific operating characteristics of the reactor. This leaves 95% of the original uranium in the used fuel and implies that less than 1% of the original uranium contained in the ore has been used in the production of power.

It becomes clear that that once-through concept with no reprocessing or recycle results in 99% of the original resource being disposed of as waste or very poor resource utilization. The various proposed recycle processes are attempts to use more of the resource to produce energy.

---

## 15.3 LWR Fuel Cycles

The fuel cycles used for LWRs utilize uranium enriched to between 3% and 5%. In the United States, there is no fuel reprocessing, so by default this implies a once-through fuel cycle. The used fuel is currently stored either in fuel storage pools or dry cask storage at the reactor sites. The recycle of used fuel has been studied since the early days of

nuclear power. Analytical studies have shown that recycle is possible and the testing of mixed oxide (MOX) fuels is underway in the United States. The MOX fuel facility is under construction at the USDOE Savannah river site. The purpose of this facility is to blend plutonium from the nuclear weapons program with uranium to manufacture commercial reactor fuel to decrease the inventory of plutonium.

Beginning in the early 1950s, recycled uranium from the DOE reprocessing facilities at Hanford, Idaho, and Savannah River was introduced into the enrichment cascades at Oak Ridge, Paducah, and Portsmouth. The introduction of this recycled material was the contamination of the cascade with neptunium and plutonium and technetium-99. The uranium arrived at Oak Ridge as 17,800 tons of uranium oxide and was converted to  $UF_6$ . About 5,800 tons was introduced to the Oak Ridge cascade from 1958 to 1962, from 1970 to 1974, and from 1976 to 1977. The remaining 12,000 tons was enriched at Paducah in the period from 1958 to 1977. About 1,300 tons of recycled uranium was fed into the Portsmouth cascade from 1955 to 1958 and in 1961, 1969, and 1974. These operations showed that it was possible to enrich recycled uranium; however contamination of the cascade with residual actinides and fission products created a long-term problem.

Studies in the 1970s looked at the possibility of using a reprocessing technique that did not separate the plutonium from the uranium during the reprocessing process (Pobereskin et al., 1978). The fuel cycle was demonstrated analytically and it involved the effective enrichment of the fuel by removing uranium after reprocessing, thus increasing the plutonium concentration. Using MOX fuel in LWRs requires well-controlled fabrication techniques so that the homogeneity of the fuel is assured to prevent hot spots during irradiation.

Outside of the United States where reprocessing is being performed, the use of reprocessed material is planned. The use of MOX fuel is common in Europe and Japan. Actually, all LWRs use energy from plutonium because the plutonium that is formed during normal operation starts with uranium fuel and builds up during operation. About 3% of the mass of used fuel is plutonium.

---

## 15.4 Fast Neutron Reactors

The first power-producing reactor in the United States was a sodium-cooled fast reactor, EBR-I. Fast neutron based reactors are able to utilize the  $U^{238}$  as well as the  $U^{235}$  as fuel. In addition, they are able to produce more fissionable plutonium than they burn. In a cover letter to a study commissioned by President Kennedy in 1962, Atomic Energy Commission Chairman Glenn T. Seaborg wrote the following:

*This restudy made it apparent that, for the long-term benefit of the country, and indeed of the whole world, it was time we placed relatively more emphasis on the longer-range and more difficult problem of breeder reactors, which can make use of nearly all of our uranium and thorium reserves, instead of the less than one per cent of the uranium and very little of the thorium utilized in the present types of reactors. Only by the use of breeders would we really solve the problem of adequate energy supply for future generations. (Seaborg, 1962)*

“Today about 20 Fast Neutron Reactors (FNR) have already been operating, some since the 1950s, and some supplying electricity commercially. About 400 reactor-years of operating



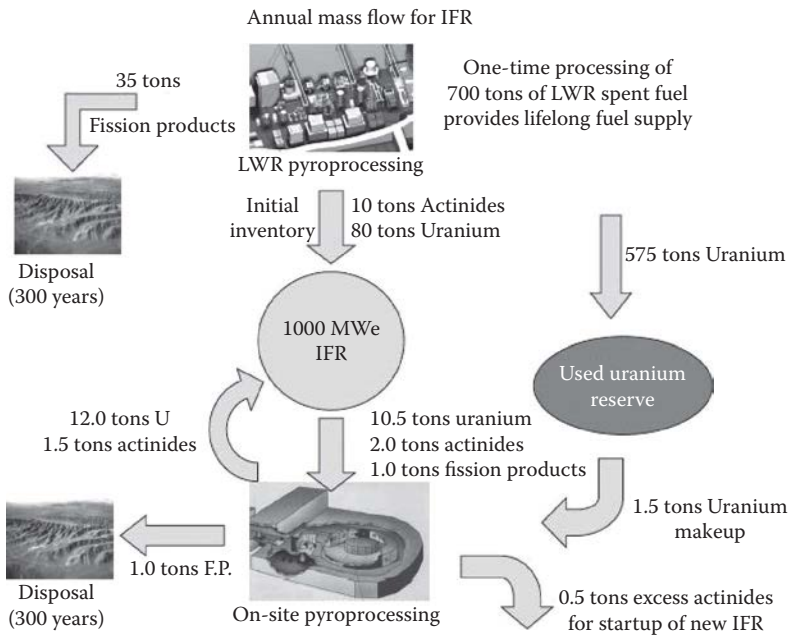
experience have been accumulated to the end of 2010. Fast reactors more deliberately use the uranium-238 as well as the fissile U-235 isotope used in most reactors. If they are designed to produce more plutonium than they consume, they are called Fast Breeder Reactors (FBRs). But many designs are net consumers of fissile material including plutonium. Fast neutron reactors also can burn long-lived actinides which are recovered from used fuel out of ordinary reactors” (World Nuclear Association, 2015). Fast reactors are covered in Chapter 6.

As part of developing the fast reactor in the United States, the DOE operated the EBR-II reactor in Idaho from the 1960s into the 1980s. During operation of the reactor, the full fuel cycle was demonstrated including the reprocessing system called “pyroprocessing” (Chapter 14). The system including the reactor and the reprocessing facility is referred to as the IFR (Integral Fast Reactor). The process, its development, and its integration with fast reactor operations is described in detail in the book *Plentiful Energy—The Story of the Integral Fast Reactor* (Till and Chang, 2011).

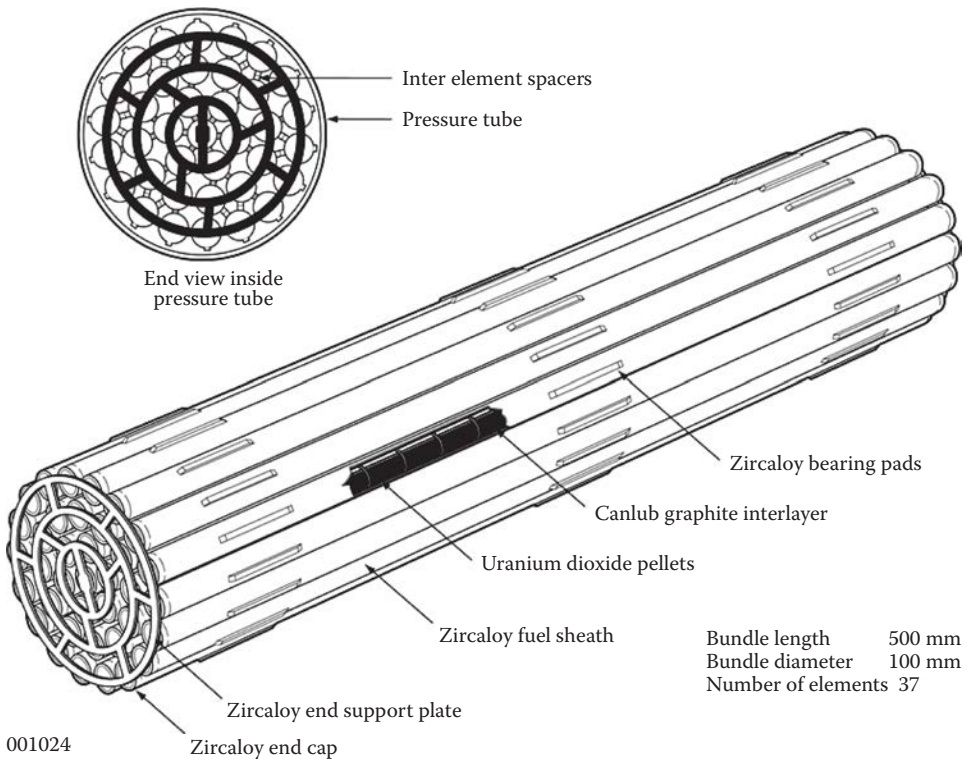
In a recent paper (SCGI Conference, 2012) titled “The Case for Near-term Commercial Demonstration of the Integral Fast Reactor,” the positive aspects of the IFR and its implementation are discussed. The following is a quote from the synopsis of the paper:

A whole–system evaluation by an international group of nuclear and energy experts, assembled by *The Science Council for Global Initiatives*, reached a consensus on the synergistic design choices: (a) a well-proven pool-type sodium-cooled fast reactor, (b) metal fuel, and (c) recycling using pyroprocessing, enabling the transmutation of actinides.

The system is shown in Figure 15.1. This system shows that it is possible to fuel an IFR with 700 tons of LWR-used fuel and operate it for its lifetime while also producing excess



**FIGURE 15.1** An integrated IFR system based on the use of LWR used fuel processed by a standalone pyroprocessing facility.



**FIGURE 15.2**  
CANDU 6, 37-element fuel bundle.

actinides that can be used for the startup of another new IFR. The processing of the used fuel completely changes the characteristics of the waste by greatly reducing its mass and average half-life.

Examination of Figure 15.2 reveals several interesting features of the proposed system. The first is a standalone LWR pyroprocessing facility. This facility will allow the recycle of 95% of the 700 tons of LWR fuel including all the uranium and the actinides. The remaining 5% is waste sent for disposal. Dealing with the used fuel from current nuclear plants is critical because the accumulating waste is likely the largest political problem facing the advancement of nuclear power.

The IFR system is designed to include a pyroprocessing facility for its own fuel so there will be no transportation required to process or store the fuel. Minimal on-site storage is required because the fuel is reprocessed within a matter of months after it is discharged from the reactor. The excess actinides can be combined with DU to produce fuel for the startup of a new IFR.

Because the actinides have been removed from the waste stream, the average half-life of the remaining material is greatly reduced. This means that an isolation period of 300 to 500 years will bring the activity level of the waste to less than that of the uranium that was used to produce it.

---

## 15.5 NUE Fuel Cycle

The HWR maintains very low excess core reactivity. Core reactivity characteristics change very little throughout the fuel residence in the core. This is made possible by on-power refueling because reactor criticality can be maintained indefinitely by replacing fuel on a daily or quasi-daily basis. This avoids the need for large amounts of soluble poison in the moderator or of burnable poisons in the fuel. Typically, the excess reactivity held down in an HWR is about 20 mk, compared with more than 200 mk at the start of a fuel cycle in a pressurized water reactor (PWR). Only small amounts of excess reactivity are needed in the light-water zone-control compartments for bulk reactivity and spatial control, and a desired level of xenon override capability can be designed into the adjuster rods. The small excess reactivity is a safety advantage, as the amount available to be added to the core in accidents is limited.

The HWR lattice features a long neutron migration length. This is a characteristic of the moderating properties of heavy water. It explains the relatively large HWR lattice pitch (28.6 cm for a CANDU reactor), which in turn makes possible the pressure tube design and the consequent capability for on-power refueling.

The large lattice pitch and the long neutron diffusion time in the HWR lattice yield a very long prompt-neutron lifetime  $l^*$ , approximately 1 ms in a CANDU reactor, which slows the evolution of power in transients, especially near-prompt-critical power excursions.

### 15.5.1 Fuel Design

The CANDU 37-element fuel bundle used in CANDU-6 reactors is a typical design, relatively small (0.5 m in length, 10 cm in diameter) and easily handled. It weighs about 24 kg, of which more than 90% is uranium oxide fuel, uses two main materials, Zircaloy and  $\text{UO}_2$ , and has a total of only seven components: pellets, sheaths, CANLUB coating, end-caps, spacers, bearing pads, and end plates. Hence, it is an easily manufactured product that countries using HWRs have found straightforward to localize. Fuel fabricators need only the following materials:

- Natural  $\text{UO}_2$  powder, for production of  $\text{UO}_2$  pellets
- Zircaloy wire, for the production of spacer pads and bearing pads
- Zircaloy strip for the production of end plates
- Zircaloy bar for the production of end caps
- Zircaloy tubing, for the production of fuel element sheathing
- Beryllium metal, for the production of braze joints
- Graphite and organic solution for the production of the CANLUB coating on the internal surface of the sheaths

The short, simple fuel assemblies for the HWR, called fuel bundles, are easily produced, and Korea, India, Argentina, Romania, and China all have independent fuel fabrication facilities sufficient to meet their demand. Furthermore, HWR fuel-cycle costs are low because natural uranium is relatively inexpensive, the uranium utilization (amount of energy produced from the mined uranium) is good, and the fuel bundle design and manufacture is simple.

Although several fuel bundle designs are in use in HWRs, a typical fuel bundle design is the CANDU 6, 37-element bundle (Figure 15.2). The following summarizes some prominent features of this fuel bundle design:

- High-density natural  $\text{UO}_2$  pellets, which ensure dimensional stability. This ensures bundle dimensional compatibility with the fuel channel and fuel handling systems.
- Thin-walled collapsible Zircaloy-4 cladding for neutron economy and improved heat transfer. Neutron economy ultimately leads to reduced electricity costs, and improved heat transfer leads to low temperature and high fission gas retention within pellets.
- High-integrity resistance-welding of end caps, resulting in good fuel reliability. The end caps serve three purposes to
  - Provide a seal for the contents of the element.
  - Provide effective element termination for attachment to the end plates.
  - Provide the structural component for interfacing with the fuel handling system.
- *No gas plenum.* Extensive operating experience confirms that no plenum is necessary in these 0.5-m long fuel elements to accommodate fission gases, thus maximizing the fissile content per bundle. This experience includes extended burnup experiments and postirradiation examination of HWR power reactor fuel.
- A thin CANLUB graphite interlayer between the  $\text{UO}_2$  pellets and Zircaloy cladding, which has eliminated fuel failures due to power ramping under normal operating conditions.
- Induction-brazed bearing and spacer pads, which respectively maintain spacing between the fuel bundle and the pressure tube and separation of the fuel elements from each other without the need of complex, expensive spacer grids (Indian HWR fuel uses welded appendages).
- *Simple bundle structure.* This is possible because the pressure tube supports the fuel bundle, and all reactivity control mechanisms are external to the fuel bundle and fuel channel

These features ensure low fueling costs, good uranium utilization, high capacity factors, and good fuel performance.

The HWR fuel design is qualified by out-reactor tests as well as test irradiation of elements and bundles in experimental loops. Various out-reactor qualification tests such as flow and heat transfer tests, strength tests, endurance tests, impact tests, and fuel handling compatibility tests are performed. Some of these tests were performed in test sections that are representative of full size fuel channels. The test conditions are designed to be more severe than those predicted in the reactor to allow for design margins. Subsequent examinations of the fuel and fuel channel components confirmed that the fuel design meets its design basis at the exit burnup. Experimental irradiations of prototype elements and bundles are also performed in in-reactor experimental loops at high powers to high burnups.

The present fuel bundle designs have evolved substantially from the 19-element design that was used in the early days in the Douglas Point prototype reactor in Canada. In addition to the CANDU 6, 37-element fuel bundles, there are the 28-element fuel bundles that are being used in the Pickering reactors. There is also another 37-element bundle design that is being used in the 850 MWe Bruce and Darlington reactors. The differences between

**TABLE 15.1**

Description of the CANDU 6, 37-Element Bundle, and the Indian 19-Element HWR Fuel Bundle

Fuel Material	CANDU 6, 37-Element Bundle	Indian 19-Element Bundle
	Natural UO <sub>2</sub>	Natural UO <sub>2</sub>
Number of fuel rods in a fuel bundle	37	19
Sheath material	Zr-4	Zr-4
Diameter of fuel pellet, cm	1.215	1.44
Outer diameter of the sheath, cm	1.3075	1.52
Clad thickness, cm	0.042	0.038
Pitch circle diameter of the first ring (6 pins), cm	2.977	3.30
Pitch circle diameter of the second ring (12 pins), cm	5.751	6.36
Pitch circle diameter of the third ring (18 pins), cm	8.661	
Pressure tube (Zircalloy) ID, cm	10.34	8.26
OD, cm	11.15	9.10
Calandria tube ID, cm	12.90	10.8
OD, cm	13.17	11.1
Material of the calandria tube	Zircaloy	Zircaloy

the two 37-element bundle designs are related to the end-cap profiles and the locations of the bearing pads, and are necessitated by differences in the fuel channel end fitting designs in these reactors.

In India, the present 19-element fuel bundle design used in the operating plants is an improved version of the wire-wrap fuel bundle that was used in the Douglas Point Reactor in Canada. This fuel was first installed in the Rajasthan Atomic Power Station-1 in 1972 (Das et al. 1995; Gupta et al. 1997; Jaya Raj et al. 1992).

A description of the CANDU 6, 37-element bundle and the Indian 19-element HWR fuel bundle is given in Table 15.1.

### 15.5.2 Fuel Performance

After more than 600 reactor-years of operation, the failure rate of CANDU fuel is very low, less than 0.1% bundle failure rate. Up to 1996, 1,400,000 fuel bundles had been irradiated in Canada, and more than 99.9% of the bundles have performed as designed. About half of the 0.1% defects could be attributed to a single cause: stress corrosion cracking (SCC) sheath failure caused by power boosts during the early refuelings of Pickering Units 1 and 2 in 1972, and by overpowering the core of Unit 1 in 1988. Since the introduction of the graphite CANLUB sheath coating in 1973, there have been very few confirmed power boost defects during normal operations. Improved fuel management and adjuster rod sequencing, developed through operational experience, are also partly responsible for reduction in the defect rate.

Table 15.2 shows the cumulative bundle defect rate over a 10-year period to 1996 for 37-element fuel bundles discharged from the 14 large CANDU reactors in Canada.

In 1997, the cumulative bundle defect rate for 37-element fuel in Canada was about 0.04%. Fuel performance continued to improve subsequently and for the period 2000–2005, the cumulative defect rate for Canadian CANDU fuel was less than 0.01% (Suk and Manzer 2006). Less than half of these failures are attributed to fabrication and

**TABLE 15.2**

CANDU 37-Element Fuel Defect Experience in Canada from About 1985 to 1995

Reactor	Bundle Defect Rate (%)
CANDU 6s (Point Lepreau and Gentilly 2 1983–1995)	0.06 (in 114,000 bundles)
Bruce A (1985–95)	0.04 (in 164,000 bundles)
Bruce B (1985–95)	0.04 (in 213,000 bundles)
Darlington (startup to 1995)	0.03 (in 74,000 bundles)
Total for above CANDUs	0.04 (in 565,000 bundles)

unknown causes. Each CANDU fuel bundle consists of 37 elements or rods, and usually, only one element per bundle failed. Therefore, over a 10-year period from 1985 to 1995, the annual defect rate for fuel elements among the 14 large CANDU reactors in Canada that use 37-element bundles is about  $0.65 \times 10^{-5}$  (Manzer 1996). For the period 2000–2005, this decreased to  $0.15 \times 10^{-5}$ .

The 1997 annual bundle defect rate for CANDU 6 reactors was 0.073% in 25,000 bundles discharged from the six operating CANDU 6 reactors (Point Lepreau, Gentilly 2, Wolsong 1 and 2, Cernavoda 1, and Embalse). About half of these bundle defects are attributed to the debris fretting among the newer units soon after they had been placed in service—construction debris in the primary circuit tends to become trapped within fuel bundles and cause defects. By comparison, the annual defect rate in the older Canadian CANDU 6 reactors at Gentilly and Pt. Lepreau was less than 0.02% in 2005. CANDU fuel reliability and experience has therefore been comparable with PWR. Furthermore, the operational implications of fuel defects are significantly less in CANDU reactors because on-power refueling (which is a routine operation) and the on-power failed fuel detection and location systems allow the removal of defected fuel without having to shut down the reactor.

The ability to locate the infrequent defects that do occur, and to remove the failed fuel during normal on-power refueling operations, minimizes coolant system contamination and the economic effect of fuel defects. This is in contrast to other reactor types, where a fuel defect must be left in the core for an extended period, or an extensive shutdown undertaken. Reactivity mechanisms are not part of the fuel bundle assembly, again simplifying fuel manufacture, and facilitating good fuel performance. Any dissolved neutron absorber that might be used for reactivity control is confined to the moderator, precluding the possibility of precipitation onto the fuel from the coolant.

### 15.5.3 Load-Following Capability

Canadian utilities normally meet the base-load requirements of the grid with the most inexpensive source of electrical energy, primarily with hydroelectricity and nuclear power. Because the fraction of electricity derived from nuclear power has increased in recent years for some utilities, HWRs may be expected to operate in various modes of load following in the future. These operations may require fuel to be power cycled on a weekly or daily or even hourly basis, depending on the grid. This requirement is especially important for those utilities whose reactors are connected to comparatively small grids.

One consideration in a postulated load-following operation is its effect on the performance and integrity of the fuel bundle. There have been previous compilations of

load-following performance of HWR fuel (Carter and Fehrenbach 1983; Hastings et al. 1990; Truant et al. 1988; Vinez et al. 1986). A current summary is given below, primarily featuring information from power reactors at Bruce B and Wolsong stations, from experimental irradiations at Chalk River Laboratories (CRL), and from analytical assessments that cover operational conditions that are beyond the existing database (Tayal et al. 1989, 1999).

For 9 months during 1986, the three commissioned reactors in the Bruce B station in Canada performed extensive load following. The frequency, duration, and nature of maneuvers varied considerably; typically, the frequency varied from zero to three per week, with the duration of reduced power being approximately 8 h. The station was operated with deep load following (power reduction about 40%) of up to 19 cycles, plus up to 65 comparatively shallower cycles (between 0% and 40% reduction in power) and 11 trips. If the trips are counted, the above maneuvers add up to a total of 95 cycles.

During the load-following period, the fission-product levels in the coolant increased in Bruce B Unit 6, and 17 "new" fuel defects were detected at the station. Each case of fuel failure was thoroughly followed up with root-cause determination in the spent fuel bay (and in hot cells in some cases). The investigations revealed that 15 of the failures were caused by debris fretting (not uncommon in a newly commissioned station); the remaining two failures were caused by manufacturing flaws (porous end caps). Thus, although fuel failures were detected during the load-following operation in Bruce B, their root causes were not directly related to the reactor power maneuvers. The above operating experience from Bruce B is very positive.

Load-following operations have also been conducted at the KANUPP reactor in Karachi, Pakistan (about 90 cycles), and at the Embalse reactor in Argentina (about 30 cycles). Fuel operating experience from the Wolsong 1 reactor in Korea shows that limited load following (e.g., many shallow cycles over a few days) does not cause fuel failures. Likewise, experimental irradiations at the CRL have shown no fuel failures during repeated power cycles: X-218 (490 cycles, 0–40 kW/m), X-411 (95 cycles, 0–70 kW/m), and U-900 (up to 103 cycles, power reduction greater than 25%). More recent power cycling tests of CANDU fuel by ICN in their TRIGA reactor at Pitesti, Romania, have confirmed this good performance (Abbas et al. 2008).

Ample evidence has accumulated to demonstrate that the fuel in CANDU 6 reactors can survive daily or weekly load following.

#### 15.5.4 Uranium Utilization

The excellent neutron economy of the HWR stems from several design features necessitated by the original desire to use NUE fuel. Neutron absorption is minimized by the choice and strategic use of reactor structural materials, and by the use of heavy water as a moderator and coolant. In addition, on-power refueling minimizes the excess reactivity required to maintain criticality. This eliminates the need for any neutron-absorbing materials needed to suppress the initial excess reactivity associated with refueling a LWR core.

A consequence of the high neutron economy of the HWR is high uranium utilization. Although natural uranium HWR fuel achieves an apparently low average burnup of only about 20% to 25% of that of enriched PWR fuel, HWRs extract about 38% more energy per Mg of  $U_3O_8$  compared with PWRs (OECD/NEA 1994). The HWR, therefore, makes efficient use of natural uranium resources. The use of NUE fuel also avoids the production of

**TABLE 15.3**

Uranium Consumption, Uranium Utilization, Equivalent NU Burnup, and Spent Fuel Arisings for Various HWR and PWR Fuel Cycles and HWR/PWR Systems

Fuel-Cycle Option	Natural U Consumption Mg U/GW(e) <sup>a</sup>	Uranium Utilization MW(e)·d/Mg U as U <sub>3</sub> O <sub>8</sub>	Equivalent Natural U Burnup MW(th)·d/kg NU	Spent Fuel Arisings Mg HE/GW(e) <sup>a</sup>
PWR with enriched U	218	1670	5.1	33
PWR with re-enriched RU recycled	182 (-16%)	2000 (20%)	6.1 (19%)	28
PWR with Pu recycled	185 (-15%)	1970 (18%)	6.0 (18%)	28
PWR with re-enriched RU recycled, and Pu recycled	160 (-27%)	2280 (37%)	7.0 (37%)	24
CANDU reactor with Natural U	157 (-28%)	2320 (39%)	7.5 (48%)	157
CANDU reactor with 0.9% SEU	119 (-45%)	3070 (84%)	9.9 (95%)	84
CANDU reactor with 1.2% SEU	116 (-47%)	3150 (89%)	10.2 (101%)	56
PWR/CANDU (2.8/1) <sup>a</sup> with RU recycled in a CANDU reactor	161 (-26%)	2280 (37%)	7.0 (38%)	25
PWR/CANDU (2.9/1) <sup>a</sup> with RU recycled in a CANDU reactor, and Pu recycled in a PWR	143 (-34%)	2550 (53%)	7.9 (55%)	22
PWR/CANDU (1.4/1) <sup>a</sup> with RU + Pu recycled in a CANDU reactor (TANDEM)	129 (-41%)	2830 (69%)	8.8 (73%)	20
PWR/CANDU (2.4/1) <sup>a</sup> with DUPIC fuel in a CANDU reactor	154 (-29%)	2370 (42%)	7.4 (45%)	24

Notes: Numbers in parentheses are percentage change from reference PWR. System uranium requirements and fuel disposal requirements for recycling options refer to an equilibrium system, in which the fresh fuel requirements of the "receiving" reactor (CANDU or PWR) are exactly met by the spent fuel discharge rate of the "supplying" PWRs.

<sup>a</sup> Ratio of PWR/CANDU electrical generation in equilibrium.

depleted-U tails. In the broader context of diverse reactor designs, HWRs can operate with no new input of uranium by burning fuel discharged from LWRs. Uranium utilization for various fuel cycles, and PWR/HWR reactor systems is shown in Table 15.3, while the fuel-cycle data used to calculate the characteristics are given in Table 15.4.

### 15.5.5 Fuel-Cycle Costs

Excellent uranium utilization and a simple fuel bundle design help minimize the HWR fuel-cycle unit energy costs, in both absolute terms and relative to other reactor types (OECD/NEA 1994). The bundle, with its few components, is easy to manufacture, and there are no criticality concerns associated with the handling or transporting of natural uranium. At the station, on-power refueling contributes to high capacity factors. Upon discharge from the reactor, criticality is again of limited concern, and the lower average burnup results in lower decay powers and shorter cooling periods. This translates into higher packing densities for ultimate disposal, which offset the greater amount of spent fuel produced (see Section 15.6). Fuel-cycle costs per unit energy for the HWR are about half those for LWRs (OECD/NEA 1994).



**TABLE 15.4****Fuel-Cycle Data and Formulae**

<i>CANDU Reactor</i>	
Net thermal efficiency	0.31 ( $\eta_1$ )
Burnup with NU, MW·d/kg HE	7.5 ( $B_1$ )
Burnup with RU from PWR spent fuel (0.9% U-235), MW·d/kg HE	13 ( $B_{1,ru}$ )
Burnup with 0.9%/1.2% SEU, MW·d/kg HE	14/21 ( $B_{1,SEU}$ )
NU feed for 0.9%/1.2% SEU, kg U/kg HE <sup>a</sup>	1.41/2.06 (F)
Burnup with RU and PU from PWR spent fuel (TANDEM), MW·d/kg HE	25 ( $B_{1,tandem}$ )
Burnup with DUPIC fuel	15 ( $B_{1,DUPIC}$ )
<i>PWR Reference</i>	
Net thermal efficiency	0.33 ( $\eta_2$ )
Burnup, MW·d/kg HE	33 ( $B_2$ )
U-235 content in fresh fuel, (%U-235 in U)	3.25
U-235 content in enrichment plant tails, (%U-235 in U)	0.25
NU feed, kg U/kg HE <sup>a</sup>	6.51 (F)
U-235 content in spent UO <sub>2</sub> fuel, (%U-235 in U)	0.92
U-236 content in spent UO <sub>2</sub> fuel, (%U-235 in U)	0.41
<i>PWR with Re-enriched RU</i>	
Burnup, MW·d/kg HE	33 ( $B_2$ )
U-235 content in re-enriched RU, %	3.62
U-236 content in re-enriched RU, %	1.23
U-235 content in enrichment plant tails, %	0.25
RU feed, kg RU/kg HE <sup>b</sup>	5.03 ( $NR_{ru}$ )
(also equal to the number of PWRs required to provide re-enriched RU for 1 PWR)	
<i>PWR with Recycled MOX</i>	
Number of PWRs required to provide Pu for 1 MOX PWR (33 MW·d/kg)	5.65 ( $NR_{pu}$ )
<i>Formulae: Equilibrium Natural U Consumption (Mg U / GW(e)a)</i>	
PWR or CANDU reactor with enriched U:	$365 * F / [\eta * B]$ , with the appropriate values of F, $\eta$ , and B
System, PWR with re-enriched RU:	$365 * NR_{ru} * F / [\eta_2 * B_2 * (NR_{ru} + 1)]$
System, PWR with plutonium recycle:	$365 * NR_{pu} * F / [\eta_2 * B_2 * (NR_{pu} + 1)]$
System, PWR with re-enriched RU & Pu recycle:	$365 * NR_{pu} * F / [\eta_2 * B_2 * (NR_{pu} + 1 + NR_{pu} / NR_{ru})]$
CANDU reactor with natural uranium:	$365 / [\eta_1 * B_1]$
System, PWR/CANDU, with RU recycled in a CANDU reactor: (the factor 0.956 is the fraction of uranium in the PWR spent fuel)	$365 * F / [0.956 * \eta_1 * B_{1,ru} + \eta_2 * B_2]$
System, PWR/CANDU, with RU recycled in a CANDU reactor and Pu recycled in PWR:	$365 * NR_{pu} * F / [0.956 * \eta_1 * NR_{pu} * B_{1,ru} + \eta_2 * B_2 * (NR_{pu} + 1)]$
System, PWR/CANDU (TANDEM): (the factor 0.966 is the fraction of uranium and plutonium in the PWR spent fuel) <sup>x</sup>	$365 * F / [0.966 * \eta_1 * B_{1,tandem} + \eta_2 * B_2]$
System, PWR/CANDU (DUPIC):	$365 * F / [0.985 * \eta_1 * B_{1,DUPIC} + \eta_2 * B_2]$

<sup>a</sup>  $F = (E - 0.25) / (0.711 - 0.25)$ , where E is the U-235 % (3.25% for PWR, 0.9% or 1.2% for CANDU).

<sup>b</sup> U-236 concentration in spent PWR fuel is assumed to be 0.41%, and in the enriched RU product is 1.23%. An additional 0.3% U-235 enrichment is needed for every 1% U-236 in the final product:  $NR_{ru} = [(3.25 + 1.23 * 0.3) - 0.25] / (0.92 - 0.25) = 5.03$ .

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## 15.6 HWR Fuel-Cycle Flexibility

### 15.6.1 Features Leading to Fuel-Cycle Flexibility

An inherent feature of the HWR is its very high degree of fuel-cycle flexibility better than that of LWRs, and with SEU or reuse of LWR fuel is even higher (Table 15.3).

While high neutron economy is required for the use of NUE fuel, it also allows the use of other low-fissile fuels such as SEU or reuse of LWR fuel (RU), and makes possible a unique synergism with LWRs that offers the potential of fuel recycling in HWR reactors that has a high degree of proliferation resistance, and even higher fuel utilization (Table 15.3) than with natural uranium, using simpler and potentially cheaper technologies than conventional reprocessing. High neutron economy also means that about double the thermal energy can be derived from burning fissile material in a CANDU reactor compared with a PWR, regardless of whether the fuel is enriched uranium, (uranium and plutonium) MOX, or recycled uranium (RU). High neutron economy also results in high conversion ratios, which can approach unity with the self-sufficient equilibrium thorium (SSET) cycle (meaning that as much fissile material is produced as is consumed). The high conversion ratio also results in a particularly strong synergism with FBRs in the future.

In a reactor design consisting of fuel channels that are separated by relatively large amounts of heavy water, the spectrum of neutrons entering a channel is very well thermalized, and largely insensitive to the fuel type. Reactivity control mechanisms are located in the low-pressure and low-temperature moderator. They are not part of the fuel bundle. This obviates various fuel performance issues that must be addressed with LWR fuel, such as deposition of dissolved boron in the coolant onto the fuel (the so-called “axial offset” phenomena); severe distortion of control rod guide tubes (resulting from having to push the burnup to very high levels to reduce fuel-cycle costs); hydriding of control rod guide tubes (due to large temperature gradients in the fuel assembly between the fuel and the guide tubes); and corrosion of the fuel sheath (again due to pushing fuel burnup to very high values).

On-power refueling contributes to HWR fuel-cycle flexibility by permitting very low excess core reactivity and core reactivity characteristics that change very little throughout the fuel cycle by replacing fuel on a daily or quasi-daily basis (see Section 15.5).

On-power refueling provides a great deal of flexibility in fuel management. Fueling is bidirectional, meaning that adjacent fuel channels are refueled in opposite directions. This method of fueling results in a flattening of the axial flux distribution, and a symmetrical axial flux distribution. The axial power distribution along the channel is mainly determined by the variation of reactivity along the channel, which itself is determined by the fuel type (particularly the initial enrichment), the fuel-management scheme, and the location of reactivity devices in the moderator (e.g., adjuster rods). The variation of reactivity along the channel can be controlled in the simplest instance by varying the rate of refueling; in most cases, this provides sufficient shaping of the axial power distribution and results in similar axial power profiles for a wide variety of fuel types. The consequence is that slightly enriched uranium (SEU), -MOX, thorium, and even inert-matrix fuels (containing no fertile material) can all be utilized in existing CANDU reactors.

Moreover, the axial power distribution with enriched fuels peaks toward that end of the channel in which a new fuel is added, and decreases along the length of the channel. For CANDU 6 reactors, in which fueling is in the direction of coolant flow, the peak bundle power occurs toward the coolant inlet end of the channel. This axial power distribution results in higher thermal–hydraulic margins than those obtained with the more symmetric axial power distribution arising from NUE fuel, and the declining power history with burnup facilitates good fuel performance.

Ultimately, bundles can be removed from the channel during on-power refueling and reshuffled, and reinserted in any order. This axial shuffling provides nearly unlimited capability for shaping the axial power distribution, if necessary. Adjuster rods are located interstitially between fuel channels, in the low-pressure moderator. They flatten the power distribution with NUE fuel, a function not required with enriched fuel, and provide xenon-override capability. With an enriched fuel, the adjuster rods can be easily replaced, if desired, or even eliminated, providing further flexibility in accommodating advanced fuel cycles.

The fuel-management scheme can also shape the radial channel power distribution across the core. With enrichment, the extra burnup potential can be traded off for an increased power in the outer channels by “flattening” the channel power distribution, obtaining more power from a given sized core without increasing maximum bundle and channel powers; alternatively, the flatter channel and bundle power distributions with enrichment can be used to reduce the peak fuel ratings for a given reactor power, increasing operating and safety margins. It has been proposed to utilize both of these approaches to some degree with the introduction of enrichment to the advanced CANDU reactor design (Torgerson et al. 2006) (see also Section 15.7.2). Fuel-management flexibility also provides many options in the transition from one fuel type to another.

An extensive array of in-core flux detectors has always been a feature of CANDU reactors, and this ensures that the flux and power distributions are well known, regardless of the fuel type and fuel-management strategy.

Finally, the basic CANDU fuel bundle design lends itself to fuel-cycle flexibility because the fuel composition can be easily varied from ring to ring within the bundle. Because of the low sensitivity of neutron spectrum to details of the fuel design, new fuels can be accommodated within operating reactors without changes to the fuel bundle or reactor core geometry.

### 15.6.2 Fuel-Cycle Drivers

The fuel-cycle path (Table 15.5) chosen by a particular country or utility will depend on many local and global factors, or criteria, key among them being:

- Overall energy economics
- Resource considerations
- Environmental impact
- National and international policies and goals

Public opinion and acceptance will also continue to influence fuel-cycle decisions. Safety is not listed as a criterion for selecting fuel-cycle options because any fuel-cycle strategy

TABLE 15.5

General Characteristics of Natural Uranium HWRs

	CANDU 6	Indian HWR 500 MWe	Indian HWR 220 MWe
Nominal reactor power (MW(th))	2061	1830	802
No. of fuel channels in the core	380	392	306
No. of fuel bundles per channel	12	12	10
Lattice pitch (cm)	28.575	28.575	22.86
Reference fuel: # of elements	37	37	19
UO <sub>2</sub> weight per bundle (kg)	21.8	21.6	15.2
U weight per bundle (kg)	19.1	19.0	13.4
<sup>235</sup> U weight per bundle (g)	135.6	135.4	95.1
Core average discharge burnup (MW·d/kg HE)	7.5	6.7	6.3
Maximum time-average channel power (kW)	6600	5500	3080
Maximum time-average bundle power (kW)	800	642	420
Maximum instantaneous channel power (kW)	7000	6215	3300
Maximum instantaneous bundle power (kW)	880	736	462
Zone control system worth/RRs (mk) <sup>a</sup>	7	6.7	5.7
Adjuster rod system worth (mk) (ARs)	16	11.5	9.0
Mechanical control rods/shim rod worth (mk)	11	10	7
SDS1 worth (mk) <sup>b</sup>	87	72	33.9
SDS1 (one or two rods not available) worth (mk)	57	50	28.4
SDS2 (liquid poison injection) worth <sup>c</sup>	>300	300	32.2
Fuelling scheme (bundles per shift)	8	8	8
Fuelling Rate (bundles per full power day)	15	15	10
Delayed neutron fraction	0.006	0.006	0.006

Source: Rozon, D. and W. Shen, A Reactor Physics Study of the Economic Penalty Associated with LVRF and Increased Pressure Tube Thickness, *Proc. 6th International Conference on CANDU Fuel*, September 26–30, Niagara Falls, Ontario, Canada, 1999. With permission.

<sup>a</sup> In the CANDU 6, the 7 zone controller units provide 14 compartments; in the Indian HWR, there are 4 absorber regulating rods (RR) which provide reactor regulation.

<sup>b</sup> In the 220 MWe IPHWR, 14 shim rods constitute SDS1.

<sup>c</sup> In the 220 MWe IPHWR, SDS2 comprises 12 liquid poison tubes.

must meet the highest safety standards. However, it is acknowledged that some advanced fuels or fuel cycles might be chosen specifically to address specific safety or licensing issues. For example, low void reactivity fuel (LVRF), discussed in Section 15.7.2, can reduce void reactivity; the CANFLEX fuel bundle features higher critical channel power, increasing the margins to fuel dryout; enrichment and new bundle designs can be used to reduce peak linear element ratings, increasing operating, and safety margins.

Thus, there are many factors that influence the choice of the fuel cycle, and various criteria upon which to optimize the cycle, as discussed by Kidd et al. (1998). The use of SEU in HWRs is an example of a fuel cycle that simultaneously optimizes several criteria; the more usual case is a tradeoff between conflicting criteria. The importance assigned to these criteria will vary from country to country, and with time. In particular, the relative weight assigned to each of the criteria will differ between “developing” and “industrialized” countries, and it is in the former that the greatest increase in energy demand (electricity in particular) will take place. Within a country, local electrical utilities will be largely driven by the short-term economic imperative to produce electricity at lowest cost,

particularly in the face of deregulation and privatization of the electricity supply industry. National organizations might have broader perspectives and agendas, encompassing international commitments to reduce greenhouse gases, and issues of national security and diversity of energy supply.

### 15.6.3 Flexible Designs for Diverse Fuels

There is thus no single fuel-cycle strategy that is optimal for all countries at all times. So, in parallel with the development of specific fuel cycles and advanced HWR fuel designs is the development of generic advancements in fuel design and performance that can be applied to any of these advanced fuels, including SEU, MOX, or thorium, or to natural uranium. These include the following:

- A generic high-burnup element design (some features of which might be applied to NUE fuel to enhance the load-following capability)
- An advanced CANLUB coating that provides protection against SCC at high burnup
- Enhanced thermal-hydraulic performance (lower pressure drop, higher critical channel power)
- Tailored reactivity coefficients (LVRF is one example)
- Low-temperature fuels (such as graphite-disk fuel and annular pellets)

Any particular fuel cycle might employ some, or all of these features, as needed. In addition, advanced characterization techniques will help to elucidate the relationship between fuel properties and fuel performance for advanced fuels. Examples of such innovative techniques include the measurement of thermal diffusivity, porosity, density, or the oxygen potential of irradiated fuel, use of advanced techniques for measuring the diffusion coefficient of fission gases (Hocking et al. 1998), and methods to accurately determine plutonium distribution in MOX fuel.

### 15.6.4 Canadian and International Experience with CANFLEX and Other Advanced Fuel Designs

#### 15.6.4.1 Canadian Experience and CANFLEX

Canadian CANDU fuel has evolved from 7-element fuel bundles in the NPD reactor, through 19 elements in the Douglas Point reactor, 28-elements in the Pickering Nuclear Generating Station, to the current 37-element bundle in CANDU 6 and Bruce and Darlington plants. Each evolution in design was accompanied by associated increases in fuel power and performance. The 43-element CANFLEX bundle is a logical extension in this evolution. AECL has been developing CANFLEX since 1986 (Hastings et al. 1989; Inch et al. 1998).

CANFLEX is a 43-element bundle design in which the outer two rings of elements have a slightly smaller diameter and the remaining central seven elements have a slightly larger diameter than the standard 37-element design. This feature flattens the radial bundle power profile so that for a given bundle power, the linear power and surface heat flux of the outer elements are significantly (~20%) lower than that for the 37-element design. This benefit can be realized in operation by keeping the bundle power the same with lower outer element fuel operating temperatures and heat flux, or by maintaining the operating temperature and heat flux margins and increasing the bundle power. Another important

feature of the CANFLEX bundle design is strategically located appendages to enhance coolant turbulence and increase the critical heat flux, again providing greater margins or higher permissible operating bundle power.

Since 1991, AECL and KAERI have pursued a collaborative program to develop, verify, and prove the CANFLEX design. New Brunswick Power began a demonstration irradiation of 24 CANFLEX fuel bundles at the Point Lepreau Generating Station (PLGS) in September 1998 that took place over a two-year period, as a final verification of the CANFLEX design prior to full-core conversion. Subsequently, KAERI led a similarly successful demonstration irradiation of Korean manufactured CANFLEX fuel.

The CANFLEX bundle design forms the basis of the both the fuel design adopted for LRVF (Section 15.7.2) and for the fuel design in the new 1000 MW Advanced CANDU Reactor.

#### 15.6.4.2 Indian Experience

India has developed a 22-element bundle design specifically for higher bundle powers than the 19-element fuel bundle currently in use. The bundle has three concentric rings of fuel pins, consisting of 1, 7, and 14 pins. The 14 pins of the outermost ring are of a smaller diameter than the remaining eight pins, which leads to reduced rating in the outer pins. A total of 522 fuel bundles of the 22-element bundle have undergone irradiation in Narora Atomic Power Station-1.

A 37-element bundle has also been developed for the 500 MW(e) HWR, which is now the HWR design being built in India. This has four concentric rings of fuel pins, consisting of 1, 6, 12, and 18 pins. A description is given in Table 15.6.

#### 15.6.4.3 Romanian Experience: Development of Romanian SEU-43 Fuel Bundle

In 1990, the Institute for Nuclear Research Pitesti (ICN) started a general research program aimed at developing a new fuel bundle for extended burnup operation (Horhoianu 1998). It adopted a staged strategy, in which each step was based on the results obtained

**TABLE 15.6**

Description of the Indian 37-Element Fuel Bundle

Fuel Material	Natural UO <sub>2</sub>
Number of fuel rods in a fuel bundle	37
Sheath material	Zr-4
Diameter of fuel rod, cm	1.22
Outer diameter of the sheath, cm	1.30
Clad thickness, cm	0.038
Diameter of the first ring (6 pins), cm	2.97
Diameter of the second ring (12 rods), cm	5.73
Diameter of the third ring (18 rods), cm	8.63
Pressure tube (Zircaloy) ID, cm	10.4
OD, cm	11.3
Air gap thickness, cm	0.65
Calandria tube ID, cm	12.6
OD, cm	12.9
Material of calandria tube	Zircaloy

in the preceding steps (Horhoianu 1992; Horhoianu et al. 1991; Laslau and Serghiuta 1991). This process resulted in a 43-element bundle design consisting of two fuel element sizes: 11.78-mm diameter elements in the outer ring, and 12.4-mm diameter elements in the intermediate, inner, and center rings. The peak linear element ratings are reduced by 16% in comparison with the standard 37-element bundle (Laslau and Serghiuta 1991). The larger-diameter elements in the inner rings of the bundle compensate for the fuel volume lost due to the smaller-diameter outer elements. To maintain compatibility of the new bundle with the existing CANDU 6 reactor and fuel handling systems, the overall dimensions of the SEU-43 fuel bundle were designed to be the same as those of the 37-element bundle. The detailed design features of the bundle have continued to evolve as a result of ongoing design analysis and thermal-hydraulic testing. Test programs demonstrated the hydraulic characteristics of the bundle and the irradiation behavior of the fuel elements (Inch et al. 2001).

#### **15.6.4.4 Argentinean Experience: CARA Bundle**

Argentina has two HWR reactor types: the Embalse 648 MWe CANDU reactor and the Atucha-1, 360 MWe pressure-vessel HWR. While both reactors use 37-element, NUE fuel, the two fuel types are quite different. The CANDU fuel is the standard 37-element bundle, 0.5-m long, simple to fabricate with low fueling costs. The Atucha-1 fuel is 6-m long, somewhat more complex, with higher manufacturing and fueling costs. Both fuels are made domestically, by the fuel manufacturing company CONUAR.

To reduce the overall nuclear fuel-cycle costs in Argentina, a new fuel bundle is being developed that can be used in both reactors: Combustible Avanzado para Reactores Argentinos (CARA) (Florido et al. 1999). This is a good example of exploiting the synergism between two reactor types (through employing a common fuel bundle in both reactors) and of optimizing the overall fuel cycle (not just one component of it).

The objectives of the CARA fuel bundle are to

- Use the same fuel for both reactors.
- Increase the heat transfer area.
- Use a single fuel rod diameter.
- Decrease the fuel center temperature.
- Decrease the ratio of zirconium-to-uranium masses.
- Keep the higher uranium density of the CANDU fuel.
- Maintain the same hydraulic pressure drop per channel in each reactor.
- Achieve higher burnup using SEU.
- Do not exceed the fabrication cost of the 37-element CANDU fuel in Argentina.

While the CANFLEX bundle achieves some of these objectives, it has two different rod diameters, and hence does not meet objective 3. Hence, a new design was developed.

If the number of fuel rods is increased, keeping the core pressure drop constant, the frictional pressure drop will also increase due to the smaller equivalent hydraulic diameter. To compensate for the higher frictional pressure drop, a double-length bundle (1-m long) was adopted. This avoids the sizeable pressure drop due to the endplate and bundle junctions.

A 52-element bundle geometry was chosen due to good symmetry and compactness of the array. The reduction in the number of endplugs and endplates, due to the double-length bundle, gives a uranium credit that can be used to increase the bundle uranium mass. The pellet diameter is very similar to that of the smaller CANFLEX elements, but with a slightly greater clad thickness and gap clearance compared to the standard 37-element bundle.

The project is well advanced with a strong commitment of the utility (NASA) and the fuel manufacturing company CONUAR.

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## 15.7 SEU and RU

### 15.7.1 Introduction

The ability to use NUE fuel distinguishes the HWR from other commercial reactor types. Natural uranium fuel can enhance security and independence of fuel supply for countries in which these are important fuel-cycle or energy drivers. Its use can also increase the localization of the fuel supply by eliminating reliance on foreign sources of enrichment. It must be understood, however, that excellent neutron economy in the HWR *enables* the use of NUE fuel, and is not a *result* of its use. The high uranium utilization using NU in HWR reactors (as measured, for instance, by the amount of energy derived from the mined uranium) is even further improved through the use of SEU fuel. Furthermore, enrichment is readily available on the open market, from a wide variety of fully safeguarded sources.

The use of SEU (or recycled uranium, RU, from reprocessing spent PWR fuel) in current HWR designs offers many benefits, and in many countries having such HWRs one would anticipate that enrichment will be introduced in the near future.

The inherent differences in the neutronics, and the low fabrication cost of HWR fuel, mean that the optimal enrichment in HWRs is much lower than in PWRs: between ~0.9% and 1.2% in current HWRs, corresponding to burnups between ~14 MW d/kg HE and ~21 MW d/kg HE. This is in contrast with LWR fuel, where the much higher fuel fabrication cost drives the optimal enrichment to as high a level as can be achieved. This pushes LWR fuel technology toward life-limiting fuel performance limits. In the current designs of HWR, most of the benefits of SEU can be realized at an enrichment level near 0.9%. This represents a small incremental step technically in the evolution of HWR fuel cycles.

While enrichments around 0.9% can likely be successfully introduced using the existing 37-element fuel bundle, it is likely that SEU will be incorporated into an advanced bundle design, such as CANFLEX or CARA, which would provide greater confidence in maintaining the excellent fuel performance achieved with natural uranium fuel, as well as other operational benefits, such as improved thermal-hydraulic performance. A likely fuel-cycle scenario in some countries would be the introduction of CANFLEX with natural uranium fuel to take advantage of higher thermal-hydraulic margins. Once there is experience and confidence with the new carrier, SEU (or RU) could then be introduced using CANFLEX.



### 15.7.2 Benefits of Enrichment in the HWR

The use in current HWRs of SEU with enrichment between 0.9% and 1.2% may very well be a *compelling* product, providing many benefits. This section summarizes some of those benefits.

The already high uranium utilization (in terms of electrical energy derived from the mined uranium) is increased by 32% and 36% compared with natural uranium fuel, for 0.9% and 1.2% SEU, respectively (Table 15.2). As lower enrichment plant tails become economical through advances in enrichment technology, the improvements in uranium utilization with SEU will become even larger: 43% for 0.9% SEU, and 56% for 1.2% SEU with 0.1% enrichment plant tails, relative to natural uranium. Lower enrichment-plant tails pushes the optimal enrichment level higher.

The inverse of uranium utilization is uranium consumption. Relative to a PWR, natural uranium requirements in an HWR are ~30% lower with natural uranium fuel. Enrichments to 0.9% SEU and 1.2% SEU would increase the fuel burnup by a factor of ~2 and ~3, respectively, relative to natural uranium fuel. This amounts to ~45% lower uranium consumption with 0.9% SEU. The reduction in mined uranium requirements also has environmental benefits at the front end of the cycle, which will become even more important in the coming decades as cheaper, higher-grade uranium ore resources are depleted, requiring the mining of greater volumes of lower-grade ores.

The economics of switching from natural uranium to SEU depend on the cost of uranium, the cost of enrichment, and the burnup achieved by the SEU fuel. Use of slight enrichment is even more attractive if the enriched material is RU.

While fuel-cycle costs are currently a small fraction of the total unit energy cost for HWRs, there is already significant utility interest in exploiting any opportunity for reducing OM&A costs to meet competition in an increasingly deregulated electricity supply industry. Enrichments around 0.9% are below the threshold at which criticality considerations result in restrictions and complications in fuel fabrication, fuel handling, and fresh or spent fuel storage and would not constrain the fuel fabrication process.

In operating reactors that have surplus heat removal capacity, or in which this can be provided in a cost-effective manner during a planned outage (such as during retubing), SEU can be used to flatten the channel power distribution across the core (see Section 15.6.1) and hence to uprate the reactor power without increasing the limits on maximum bundle or channel power. Power uprating can provide a large economic benefit to operating plants, and enhance the case for plant-life extension.

Alternatively, enrichment could be used to flatten the channel and bundle power distributions in the core without increasing reactor power. Using CANFLEX, for example, as the carrier for SEU, peak linear element ratings could be reduced to below 40 kW/m, resulting in virtually no fission gas release during normal operation, and increased operational and safety margins.

Flattening the channel power distribution does cause a trade-off of some of the burnup potential of SEU because higher power in the peripheral channels increases neutron leakage from the core. Flattening the channel power distribution across the core can be achieved using a single fuel type, or with two or more fuel types (enrichments) across the core (Dastur and Chan 1993). Using single fuel enrichment, the flattening is achieved through differential burnup across the core, reducing the burnup (and dwell-time) of the fuel in the outer part of the core, relative to the unflattened reference case, to increase the power there. Using 0.9% SEU to flatten the channel power distribution in the core without

increasing bundle or channel power limits would yield ~1100 MW(e) from a 480-channel, Darlington-size core, nominally rated at 935 MW(e).

SEU also offers greater flexibility in fuel-bundle design, providing the means to achieve specific operating benefits, or reductions in capital cost. For example, reactivity coefficients can be tailored to meet specific objectives. The LVRF concept provides a means of optimizing the value of void reactivity and burnup (Boczar et al. 1992; Rozon and Shen 1999). It uses a neutron absorber (e.g., dysprosium) in the central elements of the bundle and enriched fuel (SEU or MOX) in the outer rings of fuel. By independently varying the concentration of poison and the level of enrichment, the desired value of void reactivity and burnup can be achieved. Hence, void reactivity can be considered as a design variable in new plants. Reduced, or even negative, void reactivity might offer opportunities for capital cost reduction in new plants by reducing the number of coolant loops, by eliminating the need for interlacing the feeders, or by allowing an increase in the maximum bundle power before safety limits are encountered. These benefits would have to be evaluated against the higher fuel cost. AECL has conducted an extensive qualification of LVRF designs, including thermal-hydraulic measurements of critical heat flux in freon, reactor physics measurements of void reactivity and fine flux distributions through the bundle in the ZED-2 zero-power reactor, and irradiation of prototype bundles in the NRU reactor. A demonstration irradiation of LVRF has been completed in one of the Bruce Station reactors where the use of LVRF to increase LOCA margins is being considered (Liska and McArthur 2003).

Operational considerations for existing HWR reactor designs are easily met with enrichments in the range of interest (0.9%–1.2% SEU), with no changes to the reactor. The HWR's on-power refueling offers flexibility in fuel management that facilitates the use of SEU and other advanced fuel cycles. This flexibility extends from the equilibrium core where, for example, different fuel management strategies could be used to accommodate different levels of enrichment to the transition from one fuel type (such as natural uranium) to another (such as SEU). For new designs, such as the Advanced CANDU reactor (ACR), the use of enrichment in the range of 2%–3% permits significant reduction in capital costs and tailoring of core reactivity coefficients.

In summary, SEU offers compelling advantages to operating and new HWRs. It is anticipated that SEU will be introduced into operating plants and will be included in the reference design of new plants within the next decade. It is likely that in this time period a reactor optimized for the use of SEU would still be able to revert to the use of natural uranium fuel if desired (although at some penalty).

### 15.7.3 Use of SEU in the Advanced CANDU Reactor

As noted in Section 15.7.2, fuel enrichment can also be used to provide reactor design flexibility. In the ACR, substantial reductions in capital cost were made possible through the use of enrichment (e.g., Advanced CANDU Reactor) (Boczar et al. 1996; Torgerson 2002; Torgerson et al. 2006). Studies indicated a potential reduction in capital costs of more than 20% relative to the latest CANDU plants. By using light water as coolant, the lattice pitch could be significantly reduced, reducing the overall size of the core for a given power and further reducing the heavy water inventory. These design features allowed the elimination or simplification of a number of other reactor systems and resulted in a more compact, simplified reactor design with an associated significant decrease in capital cost. A high degree of passive safety could also be incorporated into such a design.

Enrichment has also been used to optimize AECL's ACR design by increasing the pressure-tube thickness, thereby extending pressure tube lifetime, and enabling the upgrade of the heat transfer system (HTS) conditions to achieve higher thermodynamic efficiency.

An LVRF type of fuel design was also proposed in the ACR design to achieve a slightly negative core void reactivity throughout the design life of the reactor. The fuel design is referred to as CANFLEX ACR, and consists of 43 fuel elements of uniform 2.2% enrichment surrounding a central Zr alloy clad rod of ZrO<sub>2</sub> pellets containing a burnable neutron absorber. The use of a neutron absorber to manage void reactivity requires a slightly greater degree of enrichment to achieve the same power and burnup. The fuel design for the initial startup ACR core is designed to achieve a burnup of 20,000 MWd/teU. To further improve uranium utilization and reduce fueling costs, transition to a reference core with a burnup in the range of 30,000 MWd/tU would begin during the first several years of operation.

#### 15.7.4 Higher Burnup Fuel Design and Performance Experience

To facilitate the achievement of extended burnups of interest in the near term (two to three-times natural uranium burnups), AECL and KAERI have developed the CANFLEX bundle (Section 15.6.4.1) (Inch et al. 2001; Jo and Suk 2008); ICN in Romania has developed the "SEU-43" bundle design (Section 15.6.3); and Argentina has developed the CARA bundle (Section 15.6.4). These designs all feature greater bundle subdivision (more elements in all cases, and two element sizes in the case of CANFLEX and Romania's SEU-43 bundle). This reduces peak linear element rating, as well as peak and average fuel temperatures, hence lowering fission gas release. Other minor changes to the pellet or element design can be included to accommodate higher burnup:

- Optimization of the chamfer shape, pellet length-to-diameter ratio (L/D), and pellet dish. The chamfer shape at the ends of the pellets influences the size of the interpellet, circumferential sheath ridges, as does the L/D ratio. The chamfer, as well as the dishes at the pellet ends, also provide space to accommodate fission gas release.
- Optimization of radial and axial clearances; provision of plenum voidage at the ends of the element to provide volume to accommodate fission gas release.
- Optimization of endcap shape to avoid stresses in that region.
- Optimization of cladding thickness for extended burnup.
- Provision of natural uranium fuel pellets at the pellet stack ends to reduce end-flux peaking.
- Use of pore-formers in the pellets to enhance dimensional stability.
- Alternate appendage attachment (such as resistance brazing) to reduce the extent of the heat-affected zone.
- Increased thickness of CANLUB, or alternate CANLUB coating (to enhance protection against SCC failures at extended burnup).
- Use of large-grain pellets to reduce fission gas release into the free inventory.

The ongoing SEU fuel development program at AECL and elsewhere involves the irradiation of elements that incorporate design improvements to enhance extended burnup fuel

performance. For even higher burnups in the longer term, other design changes have been tested in the past that would significantly reduce gas release by lowering fuel temperatures, such as graphite disks between pellets to provide a radial heat conduction path from the fuel to the sheath, and/or annular fuel (Hastings and MacDonald 1984; MacDonald 1970; MacDonald and Hastings 1984).

AECL's experience with SEU and natural uranium  $\text{UO}_2$  fuel irradiated to extended burnup in AECL research reactors and CANDU power reactors is summarized elsewhere (IAEA Tec Report 407, 2002). More than 3000 CANDU bundles have been irradiated to above average burnup in power reactors, with a few to a maximum of 30 MW-d/kg HE (Floyd et al. 1992; Hains and Novak 1989). About 150 bundles have experienced burnups above 16.5 MW-d/kg HE, mainly in Ontario Power Generation reactors. The extensive experience, both from research reactor and power reactors, demonstrates that the current 37-element bundle design will operate successfully up to about 19 MW-d/kg HE, at power and burnup levels representative of the 0.9% SEU/RU fuel cycles (Boczar et al. 1993). The lower linear element ratings with CANFLEX fuel will further increase confidence in good fuel performance at extended burnups, and most importantly, provide additional operating benefits, such as improved thermal-hydraulic performance.

### 15.7.5 Mixed Natural Uranium-SEU Core

An interesting variation that can be considered is the use of a mixed natural uranium/SEU core. One such configuration would have SEU in the perimeter of the core, away from the adjuster rods, and natural uranium in the center of the core. This is one option for increasing the power in the peripheral channels, without switching to a full core of SEU, and without unduly increasing the fueling machine usage and sacrificing the burnup, as would be the case if radial flattening were pursued using natural uranium fuel only. This could also be an intermediate configuration in a transition from an all natural uranium core, to an all-SEU core.

Studies of mixed cores show that, in equilibrium, they have many advantages over the natural uranium core. The SEU is used to flatten the radial channel power distribution in the core, with the peak channel power in the mixed core 7% lower than in the natural uranium reference core. Powers in the outer channels of the mixed core are 20%–35% greater than the powers of the corresponding channels in the natural uranium core; in the center of the core, the channel powers in the mixed core are about 9% lower than in the natural uranium core. The quantity of spent fuel produced is reduced by 40%; core-average burnup increased by ~70%. The refueling rate in channels per day remains about the same as with natural uranium, but the refueling rate in bundles per day falls from 18.5 to 11. The mixed core reduces the annual uranium requirements by 24%, with significant savings in fuel-cycle costs. The axial bundle power profile with the SEU fuel peaks toward the inlet end of the channel and decreases along the length of the channel. In a CANDU 6 reactor, with coolant flow and refueling in the same direction, this power shape will improve the critical channel power.

Transition from the equilibrium natural uranium core to the mixed core would be particularly straightforward because the adjuster rods do not distort the axial power profile in the SEU-fueled channels. In those channels, the axial power profile evolves smoothly, with no significant power boosting at extended burnups.

Because the first application of enrichment in the HWR will likely be SEU at enrichment levels around 0.9%, a mixed-core with 0.9% SEU in the peripheral channels is a

particularly attractive option for introducing SEU into an HWR, while uprating the reactor power. Chan and Dastur (1987) suggest that uprating of up to 12% could be achieved in this manner.

### 15.7.6 Transition to SEU: Indian Perspective

There can be two ways of introducing SEU into HWRs. The first is starting from a NUE-fueled core and then converting to SEU by gradually replacing NUE fuel bundles by SEU fuel bundles during refueling. While online refueling with the HWR provides this flexibility, the fuel management strategy over the transition period must be carefully considered. Alternately, a new reactor (or a reactor that is returning to power after a maintenance or refurbishment outage) can be started fully loaded with SEU fuel. In this case, fuel management during the transition to the equilibrium core is simpler, but the reactivity of the initial core will be high, and has to be suppressed.

In the Indian HWR, with the whole core loaded with 1.2% SEU fuel,  $k_{\text{eff}}$  is found to be 1.254. The easy way of suppressing this excess reactivity is to use boron as the moderator. Apart from the fact that this negates the benefit of the improved uranium utilization of the SEU in the initial core, this approach will give rise to severe power peaking, and an increase in void reactivity during a loss of coolant accident (LOCA). The power peaking would be worse than in the case of the natural uranium core because the channel flow distribution of the Indian HWR has been designed to match the power distribution in the equilibrium core. As fuel burnup proceeds, the bundle powers become acceptable, but the coolant outlet temperatures in the peripheral channels increase beyond their rated values.

An interesting, and potentially better, alternative is to use thoria ( $\text{ThO}_2$ ) bundles to suppress the excess reactivity of a fresh core of SEU. A distribution of thoria bundles that will suppress the  $>200$  mk of initial reactivity without requiring derating of the reactor was determined for the Indian HWR. More details of this approach can be found in IAEA Tech Report No. 407 2002.

### 15.7.7 Use of RU in an HWR

RU is one of the products of conventional chemical reprocessing of spent uranium oxide fuel. In this context, RU is simply a subset of SEU, acquired on the open market, as is SEU or natural uranium, and its use is not linked to the reprocessing of the spent fuel of the utility acquiring the material. It is considered simply as an alternate source of enrichment (anticipated to be much cheaper) compared with SEU from enriching fresh uranium and can be substituted by SEU of equivalent enrichment. The enrichment level is around  $\sim 0.9\%$ , with the actual  $^{235}\text{U}$  content and composition depending on the initial enrichment and burnup of the spent fuel from whence it was obtained. In general, reactor physics and fuel performance will be comparable with that for 0.9% SEU.

The use in HWRs of RU from reprocessed spent PWR fuel can be considered as an illustration of HWR/PWR synergism on a global basis.

The cumulative quantity of RU from the reprocessing of European and Japanese spent fuel by 2000 was around 25,000 Mg. This RU, which is owned by the utilities or reprocesors, is an alternative fuel source to new natural uranium for use in LWR and CANDU reactors. Each country and utility will determine its strategy for RU based on local factors. Theoretically, this 25,000 Mg would provide sufficient fuel for 500 CANDU 6 reactor-years of operation because the initial core load of uranium for a CANDU 6 reactor is 85 Mg, and

annual refueling requirements for a RU fuel burnup of 13 MW·d/kg HE are around 50 Mg/a. From another perspective, two 1000-MW LWRs discharge enough spent fuel to fuel one CANDU 6, which would achieve a burnup of double that with NUE fuel.

The dose fields associated with RU are higher than those associated with natural uranium. The isotopic composition and activity of unenriched RU  $\text{UO}_2$  powders depend *inter alia* on the reactor type, initial enrichment and discharge burnup of the PWR fuel, the time between spent PWR fuel discharge and reprocessing, the route chosen to convert the UNL to  $\text{UO}_2$ , and the delay until fuel fabrication.

Studies have been undertaken of the impact on personnel dose during fuel manufacturing operations from the increased specific activity of the RU compared to natural uranium. These studies have shown that this impact can be readily minimized without significant cost penalty to the acceptable levels recognized in modern standards for fuel manufacturing operations. The increase of external activity is unlikely to require any change to current fuel manufacturing or fuel receipt and handling procedures at the reactor. Nonetheless, gamma dose rates for RU powder will increase linearly for several years after conversion to  $\text{UO}_2$ , which will require attention to the time delays between conversion, manufacturing, and use in-reactor.

The extra isotopes in RU have minimal effect on the reactor physics characteristics in an HWR compared with SEU of the same  $^{235}\text{U}$  content. Spent PWR fuel contains typically 0.4 wt%  $^{236}\text{U}$  with a range from 0.2% to 0.7%, having a strong resonance at 5.5 eV, originating from neutron capture in  $^{235}\text{U}$  in the original PWR fuel. Because of the softer neutron spectrum in an HWR, the absorption worth of the  $^{236}\text{U}$  is an order of magnitude lower in an HWR than in a PWR. Therefore, the main determinant in HWR reactor physics with RU is the  $^{235}\text{U}$  level.

There are no significant nonproliferation or safeguards considerations with the use of SEU or RU in an HWR, compared to natural uranium (considering RU as available on the open-market).

#### 15.7.7.1 Use of RU with DU in Chinese HWR Reactors

An interesting way of introducing the use RU in HWRs is to blend it with DU, another underutilized uranium resource resulting from the uranium enrichment process. Since 2008, CANDU Energy and TQNPC, the CANDU Utility in China, have partnered to convert the Qinshan CANDU reactors to use recycled and DU-based fuels, blended together to produce a fuel with an equivalent fissile content to that of NUE fuel. (Cottrell et al. 2013). One advantage of the NEU concept is that it provides the simplest and lowest cost approach to utilization of RU or SEU fuel cycles in existing CANDU reactors, since it does not require modifications to the existing licensing basis.

The composition of NUE is about 70% RU and 30% DU, with the RU containing about 0.85–0.99 wt%  $^{235}\text{U}$  and the DU about 0.2–0.3 wt%  $^{235}\text{U}$ . The resulting fissile content of the NUE is then very similar to NUE fuel. Of course, the  $^{235}\text{U}$  content of the RU and DU will vary depending on the specific source from which they are obtained, and the ratios of RU and DU will need to be adjusted accordingly to obtain the desired NUE fissile composition.

Following manufacture of 24 NUE CANDU fuel bundles by CNNFC in China, a one-year demonstration irradiation was successfully conducted during 2010–2011 in two of the higher power fuel channels of Unit 1 of the Qinhsan CANDU PHWR reactor. Monitoring during the test and postirradiation examination confirmed that the NUE fuel performance met the established test criteria CNNC is now preparing for full core implementation of NUE in Qinhsan Unit 1 with fuel loading scheduled for early 2014.

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## 15.8 DUPIC Fuel Cycle

### 15.8.1 Introduction

Direct use of spent PWR fuel in CANDU (“DUPIC”) is one group of options for recycling spent PWR fuel into CANDU fuel using only dry thermal/mechanical processes; wet chemical processing is not employed. Furthermore, there is no selective chemical element removal in the DUPIC process. This, along with the high radiation fields associated with the fuel, offers a very high level of proliferation resistance. AECL, KAERI, and the US State Department have collaborated since the early 1990s on an assessment of these dry-recycle options (Keil et al. 1992; Sullivan et al. 1999). The IAEA has also participated in the safeguards aspects of this program.

Several DUPIC variants have been identified. One option is to simply cut the PWR fuel elements into CANDU lengths (~50 cm), straighten them, and weld new end-caps to the ends (optionally, the elements could be double-clad). The smaller diameter of PWR elements would enable the use of a 48- or 61-element fuel bundle, which would significantly reduce the linear element ratings compared with those of a 37-element bundle. The low linear element ratings would enhance fuel performance, and would help to accommodate the variation in fissile content between elements.

Another option is to vibratory-pack (VIPAC) ground PWR pellets into fresh CANDU sheaths. Yet another option is the Oxidation and REduction of OXide fuels) process (“OREOX”), a series of oxidation/reduction cycles that convert used PWR pellets into a ceramic-grade powder after the cladding has been removed. The powder would be pressed and sintered as “new” CANDU pellets, and loaded into standard sheaths that would be assembled into fresh bundles. All options were judged to be technically feasible, and the last option was the focus of a collaborative technical feasibility assessment program between AECL, KAERI, and the US State Department.

In the nominal DUPIC fuel cycle, spent PWR fuel, having a nominal burnup of 35 MW d/kg HE, would be processed into CANDU fuel. An additional burnup of >15 MW d/kg HE would then be obtained through irradiation in CANDU.

The DUPIC fuel cycle offers several benefits to a country that has PWRs and CANDU reactors including

- A very high degree of proliferation resistance throughout the entire fuel cycle, fuel conversion and fabrication costs that are expected to be lower than for conventional fuel-reprocessing and MOX fabrication
- A significant improvement in uranium utilization and reduced quantities of spent fuel per unit of electrical energy produced
- A simple disposal concept combined with a reduction in the cost of geological disposal

### 15.8.2 Fuel Design, Fabrication, and Performance

At the heart of the reference DUPIC fuel cycle is the OREOX process. During this dry process, uranium from spent PWR fuel is sequentially oxidized and reduced to a fine powder, which forms the starting material for fabrication of DUPIC fuel pellets. The powder is conditioned to improve sinterability, pressed into pellets, sintered to a high density, ground to final size, and seal-welded within Zircaloy sheaths. Because the fuel remains highly radioactive, all the processing must be done in hot cells.

The entire fabrication process involves the following steps:

- Decladding the spent PWR fuel
- Exposing the fuel to thermal cycles of oxidation and reduction, to break the fuel into a fine powder (OREOX)
- Milling (or any subsequent powder conditioning) to improve the sinterability of the powder
- Fabricating CANDU-quality fuel pellets from that powder
- Loading the pellets into sheaths
- Assembling the CANDU bundles
- Disposing of irradiated PWR assembly hardware
- Trapping and disposing of volatile fission products released during the decladding

The CANFLEX bundle is the reference for the DUPIC cycle, having 20% lower peak linear element ratings than a 37-element bundle, and improved thermal-hydraulic performance.

The fuel from different PWR assemblies would be blended to ensure that each bundle has equivalent (within specified tolerance) neutronic properties.

The OREOX dry reconstitution process employed in the DUPIC fuel cycle is potentially a much simpler process than conventional wet chemical reprocessing. This process does not involve chemical separation of the spent fuel. The final fuel product (DUPIC bundles) is fabricated in the same facility in which the spent fuel is processed. This process contrasts with conventional wet reprocessing, in which the fuel is separated into uranium, plutonium, and fission product/actinide waste, with a very high decontamination factor. MOX PWR fuel made from recovered plutonium is fabricated remotely in a different processing facility. Moreover, the CANDU fuel bundle is very simple in design and fabrication (with only seven distinct components), and smaller (~0.5-m long) compared with a PWR assembly (~4.1-m long), which facilitates remote fabrication. Thus, even though fabrication must be done in a dedicated shielded facility, DUPIC fuel processing and fabrication is expected to be less expensive than conventional reprocessing and PWR MOX fuel fabrication. While preliminary assessments indicate that this is the case, more technical work is required to specify the process details before more definitive costs can be confirmed. A comparison of indicative DUPIC and reprocessing/PWR MOX fabrication costs is shown in Table 15.7 (Choi et al. 1996).

### 15.8.3 Reactor Characteristics

One of the key criteria in the final selection of the DUPIC variants for further assessment was compatibility of the fuel and fuel cycle with existing reactors, the objective being no major reactor changes required to utilize DUPIC fuel. Hence, the OREOX option utilizes an existing element and bundle design (CANFLEX), and a pellet design that meets current CANDU specifications. Obviously, changes must be made in storing, handling, and loading the fresh DUPIC fuel into the refueling machines but, aside from that, other changes to the reactor are expected to be minimal.

**TABLE 15.7**

Indicative DUPIC and Reprocessing/PWR MOX Costs

Process Stage	DUPIC (\$/kg)	Reprocessing/PWR MOX (\$/kg)
Spent fuel separation		1000
Fuel fabrication	600	800
Total cost of fuel	600	1800



Detailed reactor physics assessments have been performed for the DUPIC fuel, including lattice studies, detailed time-dependent fuel management simulations, and LOCA analysis (Choi et al. 1997). These studies confirm that DUPIC fuel can be accommodated within existing CANDU reactors. Some of the key results of these studies are summarized below.

In this analysis, the Korean 17-17 optimized fuel assembly (KOFA) was used as the reference PWR fuel, from whence the composition of the DUPIC fuel was derived. The reference PWR fuel has an initial enrichment of 3.5 wt%  $^{235}\text{U}$ , and a burnup of 35 MW-d/kg HE, which is representative of the 950 MW(e) PWR plant Yonggwang 1. At discharge, the fuel contains 0.92 wt%  $^{235}\text{U}$ , 0.56 wt%  $^{239}\text{Pu}$ , and 0.08 wt%  $^{241}\text{Pu}$ : a total fissile content of 1.56 wt%. It was assumed for the physics calculations that during fuel processing (OREOX, and sintering), all of the ruthenium and technetium were removed, and 98% of the krypton, iodine, xenon and cesium. None of the high neutron-absorbing rare-earth isotopes were assumed to be removed during processing.

Because DUPIC fuel has a faster dynamic response than  $\text{UO}_2$  due to the lower delayed neutron fraction and lower prompt neutron lifetime for plutonium isotopes, the same positive reactivity insertion will result in a faster response than for  $\text{UO}_2$ . To compensate for this in a postulated large LOCA, void reactivity for the DUPIC fuel was reduced by adding a small amount of neutron absorber to the center element (26 g of natural dysprosium). This is an embodiment of the LVRF concept and reduces void reactivity sufficiently so that the power pulse with the DUPIC fuel is lower than that with natural uranium. The downside, of course, is reduced burnup: without the neutron poison the burnup of the DUPIC fuel is ~18.8 MW d/kg HE; with the burnable poison, the burnup drops to 15 MW d/kg HE, which is still double that of NUE fuel (Table 15.8).

**TABLE 15.8**

Characteristics of Different DUPIC Core Performance with Various Spent PWR Fuel Types

Initial Enriched in PWR (% $^{235}\text{U}$ )	Batch Size in PWR	DUPIC Fuel Types	Equilibrium Discharge Burnup (MW-d/kg HE)			Instantaneous Calculation Peak Powers (kW)		CPPF
			In PWR	In CANDU	Total	Channel	Bundle	
NU	–	–	0	7.45	7.45	6799	866	1.074
3.2	1/2.5	1	30	17.71	47.71	6821	857	1.074
3.2	1/3	2	32.5	15.10	47.60	6754	837	1.062
3.2	1/4	3	34.6	13.00	47.6	6741	806	1.056
3.5	1/2.5	4	33.9	17.75	51.65	6814	853	1.072
3.5	1/3	5	36.5	15.08	51.58	6759	837	1.062
3.5	1/4	6	38.9	12.72	51.62	6718	794	1.054
3.8	1/2.5	7	37.3	18.24	55.54	6813	852	1.072
3.8	1/3	8	40.3	15.19	55.49	6754	831	1.061
3.8	1/4	9	42.9	12.66	55.56	6703	793	1.053
4.2	1/2.5	10	42.0	18.63	60.63	6812	849	1.072
4.2	1/3	11	45.4	15.20	60.60	6743	820	1.060
4.2	1/4	12	48.3	12.42	60.42	6690	799	1.050
4.5	1/2.5	13	45.3	19.07	64.37	6817	849	1.072
4.5	1/3	14	48.9	15.46	64.36	6741	818	1.059
4.5	1/4	15	52.0	12.51	64.51	6684	802	1.049

Source: Rozon, D. and W. Shen, A reactor physics Study of the Economic Penalty Associated with LVRF and Increased Pressure Tube Thickness, *Proc. 6th International Conference on CANDU Fuel*, September 26–30, Niagara Falls, Ontario, Canada. 1999. With permission.

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## 15.9 Other HWR/PWR Fuel Cycles

### 15.9.1 HWR-MOX with Pu from Spent PWR Fuel

The PWR/HWR recycling options discussed above are nonconventional, based on not separating the uranium and the plutonium from the fission product and minor actinide waste. If a country has access to conventional reprocessing, then a PWR/HWR two-reactor system opens the possibility of recycling the MOX fuel from reprocessed PWR fuel back into HWRs. This would have potential benefits compared with recycling in PWRs. A full MOX core could be used in existing HWRs. Although MOX fuel fabrication will be much more expensive than natural uranium, the simplicity of the HWR fuel bundle and its short length will result in cheaper MOX fuel fabrication costs compared with PWR MOX.

### 15.9.2 Actinide Burning in an Inert Matrix

The other product of reprocessing is the minor actinide and fission product waste. There is worldwide interest, particularly in Europe, in investigating the feasibility of annihilating certain long-lived, carcinogenic actinides produced during normal reactor operation, and concentrated by reprocessing of the fuel. The intent is to make an inert-matrix fuel incorporating these actinides ( $^{237}\text{Np}$ ,  $^{241}\text{Am}$ ,  $^{244}\text{Cm}$ ), together with some of the plutonium that is separated during reprocessing. The plutonium provides a major fissile component and the neptunium, americium and curium are transmuted or fissioned to shorter-lived, less carcinogenic nuclides. Without  $^{238}\text{U}$  in the fuel matrix, further production of these actinides does not occur. The CANDU reactor is particularly suited for this application because no major modifications to reactor design are required and because the neutron economy of CANDU reactors means the annihilation of plutonium/actinide waste can be more complete. This application would utilize the third product of reprocessing (the others being recycled uranium and plutonium), and is another manifestation of the synergism between CANDU reactors and PWRs.

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## 15.10 Burning Thorium

The thorium fuel cycle in HWRs is of strategic interest for several reasons (Boczar et al. 1998).

The abundance of thorium in the earth's crust is about three times that of uranium. Hence, the thorium fuel cycle ensures a long-term nuclear fuel supply. For countries with abundant thorium reserves, the thorium fuel cycle in HWRs would enhance the sustainability of nuclear power and the degree of energy independence using a single reactor type.

In thorium fuel,  $^{233}\text{U}$  is produced in-reactor through neutron capture in  $^{232}\text{Th}$ , and subsequent beta-decay of  $^{233}\text{Th}$  and  $^{233}\text{Pa}$ . The concentration of  $^{233}\text{U}$  in the spent fuel is about five-times higher than that of  $^{239}\text{Pu}$  in spent natural uranium  $\text{UO}_2$  fuel. This isotope of uranium is a very valuable fissile material because of the many neutrons produced per neutron absorbed ( $\zeta$ ) in a thermal neutron spectrum.

The thermal conductivity of  $\text{ThO}_2$  is about 50% higher than that of  $\text{UO}_2$  over a large temperature range, and its melting temperature is  $340^\circ\text{C}$  higher than that of  $\text{UO}_2$  (Belle

and Berman 1982; Fink et al. 1981). As a consequence, fuel operating temperatures will be lower for ThO<sub>2</sub> than for UO<sub>2</sub>, and all thermally activated processes, such as creep and diffusion of fission gas from the fuel, will be slower. Fission-gas release from the fuel should be lower for ThO<sub>2</sub> than for UO<sub>2</sub> operating under similar ratings.

ThO<sub>2</sub> is chemically very stable and does not oxidize—a benefit for normal operation, postulated accidents, and waste management.

<sup>232</sup>Th produces fewer minor actinides than <sup>238</sup>U. The resultant lower radiotoxicity of spent thorium fuel may be viewed as a benefit in waste management. However, in the Canadian concept for engineered geological disposal, the actinides contained in used fuel are not a significant contributor to radiological risk (Goodwin et al. 1994), and this benefit is judged to be small in that context.

The amount of energy that can be extracted from mined uranium can be significantly extended using thorium fuel cycles, and a wide range of thorium cycles is feasible in HWRs. In the limit, the SSET cycle is independent of natural uranium, and of any external supply of fissile material (Critoph et al. 1976; Milgram 1984). The high neutron economy of HWRs makes this fuel cycle theoretically achievable. Hence a single reactor technology can provide short-term and long-term assurance of fuel supply. High conversion ratio HWRs utilizing thorium would also be synergistic with more expensive, fast breeder reactors (FBRs), supplying fissile material.

The once-through thorium (OTT) cycle in HWRs provides an evolutionary approach to exploiting some of the energy potential of thorium without recycling. The optimal OTT cycle is economical today in terms of money and in terms of uranium resources. This cycle creates a mine of valuable <sup>233</sup>U, safeguarded in the spent fuel, for future recovery if desired.

Because commercial thorium fuel recycling facilities have not been built, there is an opportunity to establish a new, proliferation-resistant technology for recycling.

## 15.10.1 Fuel Design, Fabrication, and Irradiation Performance

### 15.10.1.1 Canadian Experience

AECL's past experience in fabricating ThO<sub>2</sub>, (Th, U)O<sub>2</sub>, and (Th, Pu)O<sub>2</sub> fuels includes fabrication routes, that is, coprecipitation, dry-powder blending/milling, wet-powder blending/milling, sol-gel microspheres, and extrusion. The latter two processes are very experimental and are not discussed here.

There are several ways of coprecipitating UO<sub>2</sub> and ThO<sub>2</sub> (Belle and Berman 1984; Radford and Bratton 1975). AECL's experience in coprecipitation has focused on the addition of ammonia to nitrate solutions. In this process, uranium and thorium are dissolved into a nitrate solution to form UO<sub>2</sub>(NO<sub>3</sub>)<sub>2</sub> and Th(NO<sub>3</sub>)<sub>4</sub>. Ammonia is added to the solution to precipitate (NH<sub>4</sub>)<sub>2</sub>U<sub>2</sub>O<sub>7</sub> (ammonium diuranate (ADU)) and Th(OH)<sub>4</sub> (thorium hydroxide). The precipitate is calcined to form blended UO<sub>2</sub> and ThO<sub>2</sub> powder, which is subsequently processed into fuel pellets. The microstructure and quality of pellets made by this process are generally very good.

Dry blending can be done by several methods, including mixing in a V-blender, dry-ball milling, vibratory milling, and jet milling. These tend to be dusty processes. There can be problems with uniformity on a microscopic scale, and intensive mixing is required. AECL investigated dry blending, using previously ground thoria powders that were mixed with highly enriched uranium powders and dry-ball milled before pressing and sintering. The microstructure of fuel pellets made by this route was quite uniform and devoid of cracks and large pores.

As is the case with dry blending, there are several wet mixing methods, including wet-ball milling and attrition milling. Wet processes have the advantage of not being dusty. In AECL's experience, it was found that product quality was poor and not reproducible with wet processes. The causes of the variations from batch to batch were not well understood. The fuel pellets used as reference fuels in the Whiteshell Reactor 1 (WR 1) irradiations were manufactured using a process of attrition milling, pan drying (forming cakes), granulation, pressing, and sintering. This process yielded pellets with nonuniform microstructures and evidence of residual granules from the fabrication process.

AECL conducted an extensive program to develop thoria fuels for use in CANDU reactors from the mid-1970s to the mid-1980s. Test irradiations were performed in the WR 1 reactor at AECL's Whiteshell Laboratories. Fission-gas release from standard reference (Th, U)O<sub>2</sub> pellets in these irradiations was significantly higher than from other thoria fuel irradiations, and was generally comparable with UO<sub>2</sub> under similar conditions.

Thoria is expected to have fission-gas retention superior to the fission-gas retention of UO<sub>2</sub>, especially at higher powers (see Section 15.10.6.4). Very recently, AECL investigated the fabrication of ThO<sub>2</sub> (with or without admixed enriched UO<sub>2</sub>), and discovered that the primary cause of the nonuniform microstructure was the lubricant that was blended with the granulated fuel before final pressing. By removing the admix lubricant and using a wash to lubricate the pellet die, pellets with uniform microstructure free of residual granules and with high sintered density could be fabricated. Although the admix-lubricant technique for lubricating dies has been very successful in UO<sub>2</sub> fuel fabrication, it does not appear to work well with ThO<sub>2</sub>-based fuels.

As a result of AECL's investigation into factors controlling microstructure, a program is underway to fabricate and irradiate thoria fuels with controlled microstructures, to demonstrate the effect of microstructure on fuel performance and the superior performance of thoria fuels over UO<sub>2</sub>.

In addition to the WR 1 irradiations, AECL also irradiated four 19-element bundles in the NPD reactor between 1977 and 1987. Two of these bundles contained 2.6 wt% UO<sub>2</sub> (enriched to 93 wt% <sup>235</sup>U) in ThO<sub>2</sub>; the other two contained 1.45 wt% UO<sub>2</sub> in ThO<sub>2</sub>. The irradiation was done at low powers (<30 kW/m) to discharge burnups of approximately 40 MW d/kg HE. The fission-gas release from these fuels was typically <1%, demonstrating good performance at low powers.

## 15.10.2 Indian Experience

### 15.10.2.1 Thorium Fuel Fabrication Experience

Thorium reactors are an important part of India's future nuclear program, so India has been constantly working in this area and making steady progress (Balu et al. 1998; Jagannathan and Lawande 1998). Indian experience with thorium dates back to the days when a 40-MW research reactor CIRUS was commissioned and thorium rods ("J-rods") were introduced into the reflector outside of the reactor vessel. Fabrication experience in making thoria and thoria-based J-rods was further broadened when about 2.5 tons of high-density sintered thoria pellets were fabricated. Since then, this technology has been transferred to the commercial plant, the Nuclear Fuel Complex. A considerable quantity of thoria elements were fabricated there and supplied for use as a blanket in the LOTUS facility in Switzerland. These elements have now been transferred to France where they will be used for carrying out reactor physics experiments.

Several ThO<sub>2</sub> fuel bundles were fabricated and loaded into the two units of the Kakrapar Atomic Power Station, and more recently into the second unit of the Rajasthan Atomic Power Station (2.5 Mg). The characteristics for thorium fuel pins have been taken to be essentially the same as for UO<sub>2</sub> fuel.

A major difficulty in the use of <sup>233</sup>U fuels is the high radiation and dose problem. <sup>233</sup>U is always associated with traces of <sup>232</sup>U, which gives rise to highly penetrating gamma-emitting daughter products. Unlike natural uranium and thorium, fuel containing <sup>233</sup>U must be fabricated under remote conditions, with proper containment inside shielded glove boxes or hot cells to keep the operator exposure at low levels.

The R&D work on thorium fuel fabrication in India is two-pronged. One route is the powder metallurgy process, and the other is the sol-gel microsphere pelletization process (SGMP). In the development of sol-gel processes, India started from what was developed at Juelich in Germany for the pebble bed HTGR. The process that was used by KFA, Juelich was modified in many ways. One major change was the addition of carbon-black to the sol so that the microspheres produced were more porous. The highest density that could be achieved without adding the carbon was only about 83% TD. Another modification was the addition of about 1% calcium nitrate in the feed solution in order to improve the sinterability of the calcined microspheres.

SGMP is ideally suited for remote and automated fabrication of thorium-based highly radiotoxic oxide fuels. Here sol-gel-derived calcined oxide microspheres, rather than fine powder-derived granules, are used for pellet pressing and sintering. Thus, radiotoxic dusts and aerosols are avoided and process losses are minimized. This process is amenable to remote handling because of the dust-free, free-flowing nature of microspheres. A high degree of microhomogeneity is attained in fuel pellets because of mixing of heavy metals in their nitrate solution forms. Finally, fuel pellets of high density with desirable microstructure are obtained by this SGMP route. The sol-gel process is increasingly used for making fine and homogeneous ceramic powders for a variety of applications and can produce high-quality microspheres. A recent innovation of the process involves the preparation of soft thorium oxide or thorium uranium microspheres and using these for making fuel pellets. Superior microstructure and porosity distribution of the pellets obtained through this route have the potential for superior performance in a reactor.

Thorium fuel has also been produced along two alternative routes, the cold pelletization and sintering route, and the pellet impregnation method used for (Th + U) MOX. This latter method uses an interesting technique in which ThO<sub>2</sub> pellets of low density are suspended in a high-temperature uranyl nitrate bath where they soak up <sup>233</sup>U. The conventional powder-pellet route involves simultaneous mixing-grinding of ThO<sub>2</sub> powder with UO<sub>2</sub> or PuO<sub>2</sub> powders, granulation, cold-pelletization of granules at ~350 MPa and high temperature (~1973 K) sintering of pellets in an Ar + 8% H<sub>2</sub> atmosphere. ThO<sub>2</sub> and PuO<sub>2</sub> powders are produced by air calcination of the oxalates. These powders are extremely fine (<1 micron) and have poor flowability. Likewise, the UO<sub>2</sub> powder produced via the ADU route is also fine and not free-flowing. Further, the platelet morphology of oxalate-derived ThO<sub>2</sub> powder causes problems in achieving homogeneity while mixing with UO<sub>2</sub> or PuO<sub>2</sub> powders.

Some modifications were made in the conventional powder-pellet route to fabricate ThO<sub>2</sub>-UO<sub>2</sub> and ThO<sub>2</sub>-PuO<sub>2</sub> pellets of high density and good homogeneity. A thorium nitrate feed solution was doped with ~1 wt% magnesium sulfate or nitrate before precipitation of thorium oxalate in order to get ~0.4 wt% MgO as "sintering aid" in the calcined ThO<sub>2</sub> powder. Proper homogenization of oxide powder mixtures is ensured by premilling the oxalate-derived ThO<sub>2</sub> powder to break the platelet morphology before comilling with

UO<sub>2</sub> or PuO<sub>2</sub>. About 0.25 wt% Nb<sub>2</sub>O<sub>5</sub> is mixed with the ThO<sub>2</sub>-UO<sub>2</sub> and ThO<sub>2</sub>-PuO<sub>2</sub> powders for improving their sinterability.

With MgO-doped ThO<sub>2</sub> powder, it was possible to achieve high pellet density with ThO<sub>2</sub>-UO<sub>2</sub> and ThO<sub>2</sub>-4% PuO<sub>2</sub> pellets at relatively low sintering temperature (<1500°C). The Th<sup>4+</sup> is partially replaced by Mg<sup>2+</sup> ions, causing the formation of oxygen or anion vacancies, which is thought to enhance the volume diffusion of thorium ions leading to rapid densification of ThO<sub>2</sub>-UO<sub>2</sub> or ThO<sub>2</sub>-PuO<sub>2</sub> at relatively low sintering temperatures.

Cermet fuel is one of the advanced concepts that has been considered in the case of thorium. It envisages kernels of uranium oxide coated with nickel and chromium dispersed in a matrix of thorium metal. A certain level of success has been achieved in coating uranium oxide microspheres with nickel.

Other R&D work that has been done at Trombay on the use of thoria in power reactors is related to evaluating the thermal diffusivity and hot hardness of ThO<sub>2</sub> plus 4% PuO<sub>2</sub> and ThO<sub>2</sub> plus 2% UO<sub>2</sub>. These data will be necessary to estimate the maximum heat rating that these fuel elements can stand.

#### 15.10.2.2 <sup>233</sup>U Recycle Experience

The Thorex process uses the organic solvent tributylphosphate. As already mentioned, thorium rods (commonly referred to as the “J-rods”) have been loaded into the CIRUS reactor from the early 1960s. A facility was set up at Trombay during the late 1960s to separate <sup>233</sup>U from the irradiated J-rods. Chemical reprocessing yields essentially pure <sup>233</sup>U with traces of <sup>232</sup>U. Though the ppm level concentration of <sup>232</sup>U in the fresh <sup>233</sup>U recovered by chemical processing is not by itself of significance, a serious problem arises due to the decay products of <sup>232</sup>U. As a result of these decay products, the gamma activity of separated <sup>233</sup>U increases for about 10 years. Because of the very high activity due to the build-up of <sup>228</sup>Th it is very undesirable to reprocess irradiated thorium, recover the <sup>233</sup>U, and then store the recovered <sup>233</sup>U. It is essential to minimize the time interval between the separation of <sup>233</sup>U and the refabrication of the <sup>233</sup>U into fuel elements. Buildup of <sup>228</sup>Th necessitates not only shielding for the high-energy gamma rays, but is also necessary to provide neutron shielding. Alpha particles emitted from the decay of <sup>233</sup>U, <sup>232</sup>U, and <sup>228</sup>Th interact with light elements like carbon and oxygen, releasing neutrons in the process. Thus, it is important to refabricate the <sup>233</sup>U as early as feasible after it is recovered. In other words, store the discharged fuel and wait for reprocessing until it is time for fabricating the <sup>233</sup>U into fuel elements. Another aspect of thorium reprocessing is related to the refabrication of the thorium separated from irradiated fuel. The separated thorium contains enough <sup>228</sup>Th to make the refabrication of the thorium as complex as the fabrication of <sup>233</sup>U-bearing fuel. It has been observed that if the thorium is stored for about 10–16 years, the activity would fall to acceptable levels.

#### 15.10.2.3 Thorium Irradiation Experience in Research Reactors

The performance targets for thoria fuel will depend on the specific fuel cycle that is intended. The requirements for a low burnup, near-breeder type of cycle will be different from those to be met by high burnup thorium fuel.

India has been following a policy of irradiating thoria whenever possible for many years. The first scheme of this kind, as discussed previously, was fabricating thorium rods and placing them in the reflector of the research reactor CIRUS. The 40-MW research reactor has 64 positions for loading thorium fuel rods. Several thorium metal rods and many

thorium oxide rods have been fabricated and irradiated in the CIRUS reactor. Because the only neutron flux that they were seeing was the leakage flux from the reactor, the  $^{233}\text{U}$  build-up was quite slow, but they did not form a load on the reactor, so that the  $^{233}\text{U}$  was produced at no loss. These rods were subsequently reprocessed to recover the  $^{233}\text{U}$  contained in them.

The CIRUS reactor has an in-core loop for fuel irradiation. This loop, called the pressurized water loop (PWL), has been extensively used for testing thorium fuel. A fuel assembly containing  $\text{ThO}_2$  and  $\text{PuO}_2$  was irradiated in the PWL from May 1985 to January 1988. The highest heat rating that the fuel pins experienced was 38.5 kW/m. The power was raised with no restriction until the heat rating reached 32.5 kW/m. Subsequently, it was raised in steps of 0.4 kW/m with an interval of 2 hours between successive power increases. The fuel cluster was subjected to about 100 power cycles as well. When the cluster was removed, the thorium-plutonium MOX fuel pins had undergone a burnup of about 18.5 MW·d/kg HE.

Further to this, a string of two six-pin clusters has been loaded into the CIRUS PWL in the last week of 1992. One of these clusters is of six pins of unenriched thoria. The other is a composite cluster of two pins of  $\text{ThO}_2\text{-PuO}_2$ , two of  $\text{UO}_2\text{-PuO}_2$ , and two natural uranium pins. The Pu content is about 4%, and the heat rating of the  $\text{ThO}_2\text{-PuO}_2$  pins is 50 kW/m. The objective of this irradiation is to prove the design of  $\text{UO}_2\text{-PuO}_2$  and  $\text{ThO}_2\text{-PuO}_2$  pins of HWR design at high burnups, to improve the understanding of the fuel behavior for advanced fuel cycles, and to benchmark the fuel design codes. The next assembly to be installed in the PWL in CIRUS will be similar to this one. The difference will be that the pellets will be made by different fabrication routes (low temperature sintering, sol-gel). A few  $\text{ThO}_2$  pellets have been used in the 100 MW research reactor at Dhruva.

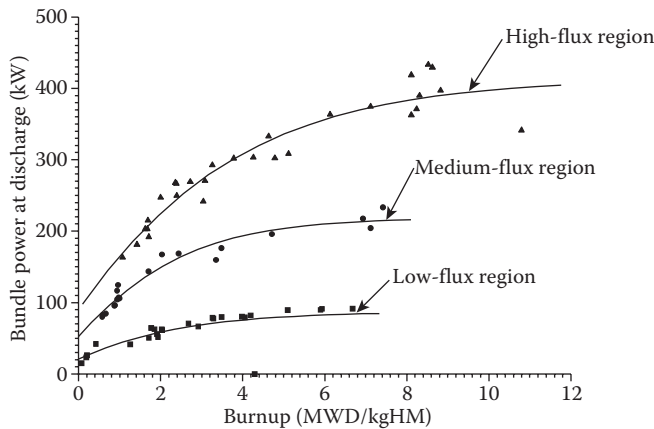
#### **15.10.2.4 Thorium Irradiation Experience in Power Reactors**

The Indian HWR is so designed that during the major part of its life, there is some power flattening in the central part of the core, achieved through a differential burnup distribution. In a newly started reactor, where the entire core is at zero burnup, the reactor will have to be derated unless some other measures can be taken to obtain a similar effect. Advantage was taken of this fact to use thoria bundles for power flattening in the initial core of new reactors.

The first regular use of thoria fuel in a power reactor in India was at the Kakrapar Atomic Power Station. About 500 kg (35 bundles) of thorium fuel was used for power flattening in the initial core of Unit-1, which was commissioned in 1993. The core has already seen 200 full-power days of operation, and all thoria bundles behaved well. The same scheme of power flattening has been employed in the second unit as well. Thus 500 kg of thoria are also present in Kakrapar Unit-2, which attained criticality in December 1994. All earlier HWRs in the world used DU in the initial core.

Before this extensive use of thoria in Kakrapar was started, performance testing of thorium bundles in power reactors was done by loading four thorium bundles in the Madras Atomic Power Station in May 1985. These bundles were unloaded during the usual refueling operation. By then the thoria bundles had been in the core for 280 effective full-power days. The highest burnup was 1.7 MW d/kg HE.

The Rajasthan Unit-2 has been restarted after retubing. In this reactor, about 250 kg of thorium was loaded for initial power flattening. The thoria bundle distribution is different here from that of Kakrapar Units 1 and 2. This is because Kakrapar uses shutoff rods for reactor scram, and so the thoria bundle locations have to be chosen in such a manner as



**FIGURE 15.3**  
Irradiation data for Indian ThO<sub>2</sub> bundles irradiated in power reactors.

not to cause reduction in the worth of the primary or secondary shutdown systems. The Rajasthan reactor depends on moderator dump for shutdown and so the placement of thorium bundles has to only consider power flattening.

ThO<sub>2</sub> bundles have seen a maximum of 700 full-power days of operation, and there have been no fuel failures. A plot showing the discharge burnup versus the bundle power at discharge is shown in Figure 15.3.

### 15.10.3 Thorium Fuel-Cycle Options

Because thorium itself does not contain a fissile isotope, neutrons must be initially provided by adding a fissile material, either within or outside the ThO<sub>2</sub> itself. How the neutrons are initially provided defines a variety of thorium fuel-cycle options in HWRs that will be examined in this section. These include the following:

- OTT cycles, where the rationale for the use of thorium does not rely on recycling the <sup>233</sup>U (but where recycling remains a future option)
- Direct self-recycle of irradiated thoria elements following the OTT cycle (no reprocessing)
- Other recycling options, ranging from reprocessing, to selective removal of neutron absorbing fission products
- SSET cycle, a subset of the recycling options in which there is as much <sup>233</sup>U in the spent fuel as is required in the fresh fuel
- High-burnup open cycles
- Pu/ThO<sub>2</sub> cycle, which can also be considered as an option for plutonium dispositioning

#### 15.10.3.1 OTT Fuel Cycles

The OTT cycle produces a mine of valuable <sup>233</sup>U in the spent fuel, at little or no extra cost, available for future recovery, predicated by economic or resource considerations.

High neutron economy, on-power fueling, the channel design, and simplicity of the fuel bundle provide a great deal of flexibility in approaches to the OTT cycle. In the original

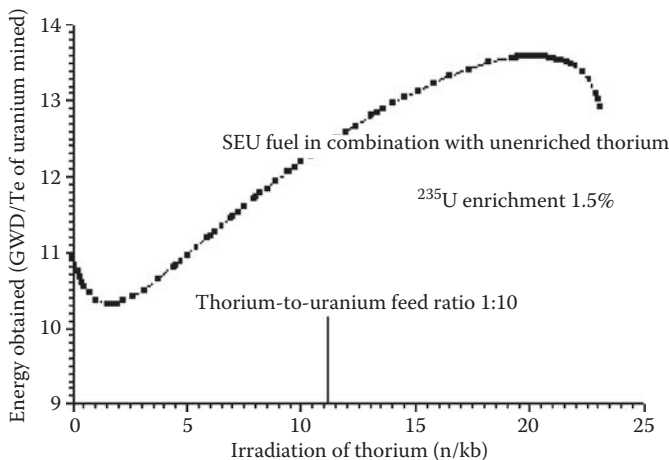


OTT concept proposed by Milgram (1982), called in this document the “mixed-channel” approach, channels would be fueled either with  $\text{ThO}_2$  bundles or with “driver” fuel, typically SEU. The driver fuel would provide the neutrons required to convert  $^{232}\text{Th}$  to  $^{233}\text{U}$  in the thoria fuel. In such a system, the thoria would remain in the core much longer than would the driver fuel.

At low burnups, the thorium is a load on the uranium, and so the presence of thorium causes a reduction in the energy obtained from uranium. With increasing thorium burnup, the  $^{233}\text{U}$  that builds in produces power and the sum total of energy extracted from the SEU and thorium can become larger than what is achievable with SEU alone. At still higher burnups, the accumulated fission product poisons cause the energy extracted to fall once again. This variation can be clearly seen in Figure 15.4. The total energy extracted will be the sum of the energy obtained from the thorium and the SEU. As the residence time of the thorium in the core increases, the energy obtained from a unit of mined uranium will first decrease then, after passing through a minimum, will start increasing again, and finally it will become higher than it would have been had no thorium been present at all.

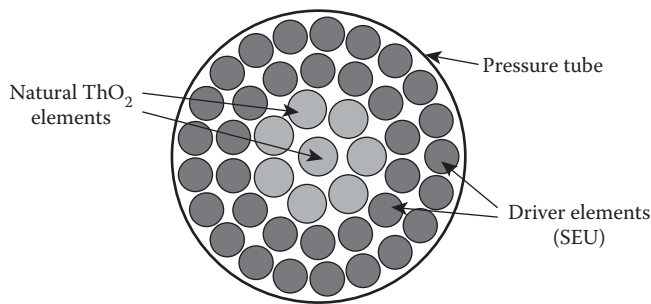
In the optimal mixed-channel approach to the OTT cycle, a combination of feed rates, burnups, uranium enrichment, and neutron flux level would be chosen so that the cycle is economical compared with natural uranium or SEU (in terms of resource utilization or money), without taking any credit for the  $^{233}\text{U}$  produced. Simple scoping studies using a lattice code have shown that such OTT cycles do indeed exist (Milgram 1982), although their implementation would pose technical challenges in fuel management because of the disparity in reactivity and power output between driver channels and thorium channels. Other driver fuels could also be considered, such as DUPIC fuel from recycled PWR fuel or MOX fuel (Dastur et al. 1995).

An alternative approach has been developed in which the whole core would be fueled with mixed-fuel bundles, which contain thorium and SEU fuel elements in the same bundle. Figure 15.5 shows a CANFLEX “mixed-bundle” containing  $\text{ThO}_2$  in the central eight elements, with SEU in the outer two rings of elements. This mixed-bundle approach is a more practical means of utilizing thorium in existing HWRs, while keeping the fuel and the reactor operating within the current safety and operating envelopes established for



**FIGURE 15.4**

Once-through thorium (OTT) cycle in HWR (n/kb or neutron per kilobarns is a measure neutron flux averaged over time).



**FIGURE 15.5**  
CANFLEX "mixed-bundle."

the NUE fuel cycle. Compared with natural uranium fuel, this option has better uranium utilization, comparable fuel-cycle costs, lower void reactivity, higher thermal-hydraulic margins, a simpler fuel management scheme, and lower bundle and channel powers. However, the uranium utilization and fuel-cycle costs are not as low as for SEU, or for an "optimized" OTT cycle using the mixed-channel approach. This mixed-bundle option is a practical means of utilizing thorium in operating HWRs, within the current safety and operating envelopes, and with no significant hardware changes.

### 15.10.3.2 Direct Self-Recycle

Further energy can be derived from the thorium by recycling the irradiated thoria fuel elements containing  $^{233}\text{U}$  as-is without any processing into the center of a new mixed bundle (Boczar et al. 1999). Recycling of the central 8 thoria elements results in an additional burnup of  $\sim 20$  MW-d/kg HE from the thoria elements for each recycle. The reactivity of these thoria elements remains remarkably constant over irradiation for each recycle. This direct, self-recycle results in a significant improvement in uranium utilization compared with the OTT: after the first recycle, the uranium requirements are  $\sim 35\%$  lower than for the natural uranium cycle, and  $>10\%$  better than the optimal SEU cycle, and remain fairly constant with further recycling. The cumulative uranium requirement averaged over a number of cycles is  $30\%$ – $40\%$  lower than a NUE-fueled CANDU reactor.

### 15.10.3.3 Other Recycling Options

The burnup, and energy from the thoria elements could be increased even further by removing the rare-earth, neutron-absorbing fission products from the spent fuel, before recycling. At the one end of the spectrum is conventional reprocessing as applied to spent thoria fuel (the so-called "thorex" solvent-extraction process, which separates the uranium and the thorium from the fission products and other actinides). The radiation fields caused by the presence of  $^{232}\text{U}$  (which emits copious  $\alpha$ -particles) and its daughter products (particularly  $^{208}\text{Tl}$ , which emits a 2.6-MeV  $\beta$ -ray), provide a degree of self-protection and increase the proliferation resistance of recycled fuels containing  $^{233}\text{U}$ . However, the absence of a commercial thorium recycling industry opens up the opportunity for developing a novel, simpler, more proliferation-resistant recycle technology.

For example, AECL has conceived a simple means of removing neutron absorbing, rare-earth fission products that have a higher degree of proliferation resistance than the conventional thorex process, and which would be much cheaper. The spent fuel would be

dissolved in nitric acid, and the pH adjusted. Uranium and thorium precipitate at similar values of pH, while the rare-earth fission products remain largely in solution until a much higher pH. By adjusting the pH of the solution, the uranium, thorium, and some radioactive fission products could be removed from solution, leaving the parasitic, neutron-absorbing, rare-earth fission products behind.

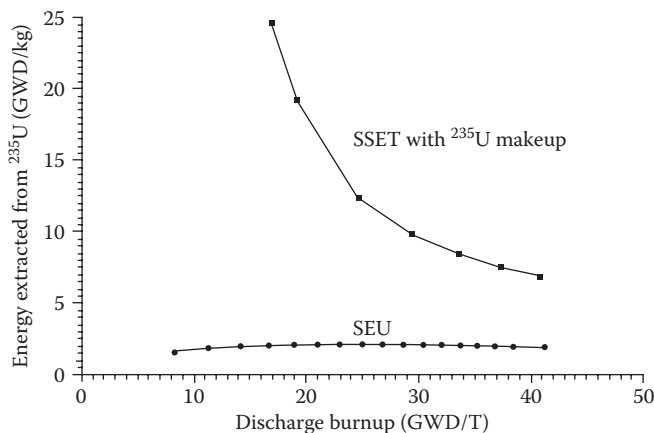
The resultant fuel would be highly radioactive and this processing (as well as the subsequent fuel fabrication and handling) would be done remotely. This would greatly enhance the proliferation resistance of the fuel because it would have a distinct radioactive signature and would be difficult to access. Moreover, the simplicity of the CANDU fuel bundle design would facilitate processing, remote fabrication, and handling, and reduce the cost relative to more complex fuel designs. Of course, this recycle option would be more expensive than the simple OTT cycle, or the direct, self-recycle of irradiated thoria fuel elements into new bundles.

Thorium reactors are an important part of India's future nuclear program and the recycle of  $^{233}\text{U}$  an essential part of assuring a long-term energy supply.

#### 15.10.3.4 SSET Cycle

The ultimate uranium-conserving fuel cycle would be an SSET cycle in which no fissile topping (and hence, no natural uranium) would be required in equilibrium (Critoph et al. 1976), that is, the  $^{233}\text{U}$  concentration in the recycled fresh fuel matches the  $^{233}\text{U}$  concentration in the spent fuel. Further improvements in neutron economy would be required to achieve this: reducing the fuel rating to lower the flux and hence neutron capture in  $^{233}\text{Pa}$ , increasing the moderator purity, removing the adjuster rods from the core, and enriching the Zr used in the pressure and calandria tubes to remove most of the high cross section isotope,  $^{91}\text{Zr}$ . It is questionable whether all of this could be economic and, in any case, a major shortcoming of the SSET cycle is its low burnup (10–15 MW·d/kg HE), which will not be economical in a cycle that requires reprocessing and remote fabrication of the  $^{233}\text{U}$ -bearing fuel. A practical approach is addition of a small amount of  $^{235}\text{U}$  to each cycle. With this, the burnup can be increased as desired.

Figure 15.6 shows the energy production in a  $^{233}\text{U}$  "near-sustaining" cycle with  $^{235}\text{U}$  makeup in the form of 0.9% SEU. Because there is no net destruction of  $^{233}\text{U}$ , it can be



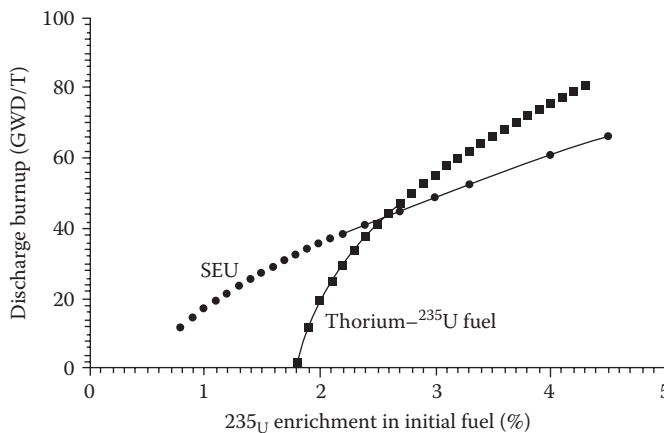
**FIGURE 15.6**  
Comparison of SSET and SEU fuel cycle.

assumed that all the energy is coming from the destroyed  $^{235}\text{U}$ . As the discharge burnup increases, the energy extracted decreases. Also shown in the same figure is a similar curve for SEU, and the immense advantage of the thorium near-sustaining cycle is clearly evident.

### 15.10.3.5 High-Burnup Open Cycle

The high-burnup thorium open cycle avoids the issues related to closing the fuel cycle with reprocessing. In this cycle, the burnup is increased by trading off the conversion ratio. The thorium is enriched with  $^{235}\text{U}$  to give whatever burnup the fuel can achieve. The spent fuel is not recycled (although this option would not be precluded). High burnup is of course, equally possible with SEU, but the advantage of thorium over SEU lies in the fact that for very high discharge burnups, the initial fissile content required is lower with thorium fuel. Figure 15.7 shows Indian calculations of the discharge burnup as a function of initial  $^{235}\text{U}$  content for SEU fuel, and for thorium topped with  $^{235}\text{U}$ . For low enrichments, SEU gives higher discharge burnup for a given  $^{235}\text{U}$  enrichment, but for very high discharge burnups, the enrichment required for the thorium fuel is lower than that required for SEU. In this theoretical assessment, pure  $^{235}\text{U}$  is added to the thorium.

The main advantage of this thorium cycle as compared with an equivalent enriched uranium cycle stems from the fact that as  $^{235}\text{U}$  is burnt,  $^{233}\text{U}$  is built up, and  $^{233}\text{U}$  being a superior fissile material than  $^{235}\text{U}$ , the reactivity vs. burnup curve falls off slower with thorium than with enriched uranium. This means that to attain the same discharge burnup, the initial  $^{235}\text{U}$  content can be lower in the thorium cycle. Consequently, power peaking problems are easier to manage with the thorium cycle. For a discharge burnup of around 66 MW·d/kg HE, SEU requires an enrichment of 4.5%, whereas thorium needs only 3.5% (in total heavy element). Add to this the fact that thermal neutron absorption in thorium is about three-times that in  $^{238}\text{U}$ , we can see that the initial reactivity in the thorium core will be much below that of the SEU core for the same discharge burnup. This leads to lower reactivity swings, which is a definite operational advantage. This cycle is also an attractive cycle for plutonium annihilation, as discussed in Section 15.10.3.6, having very high plutonium destruction efficiency.



**FIGURE 15.7**  
Comparison of thorium and uranium cycles for high burnup.

### 15.10.3.6 Pu-Thorium as a Plutonium-Dispositioning Option

A special application for thoria that has recently received attention is its use as a matrix material for the annihilation of weapon-derived plutonium (Boczar et al. 1997; Chan et al. 1997). This is a responsible, forward-looking strategy that uses plutonium to convert  $^{232}\text{Th}$  to  $^{233}\text{U}$  and would be available as a future energy resource as and when needed. The  $^{233}\text{U}$  would be safeguarded in the spent fuel, with all the proliferation-resistant features of spent  $\text{UO}_2$  or MOX fuel. As noted above, the radiation fields caused by the presence of  $^{232}\text{U}$  (which emits copious  $\alpha$ -particles) and its daughter products (particularly  $^{208}\text{Tl}$ , which emits a 2.6 MeV  $\gamma$ -ray), provide a high degree of self-protection and render  $^{233}\text{U}$  unattractive as a weapons material. The  $^{233}\text{U}$  could be recovered in the future using a proliferation-resistant technology, when warranted by the price of uranium and other factors.

A different approach was taken in designing the Pu-ThO<sub>2</sub> fuel bundle for this application. To maximize the destruction of the plutonium, good neutron economy was desired. A reduction in void reactivity was also sought to compensate for the faster dynamic behavior of the fuel (shorter neutron lifetime, and smaller delayed-neutron fraction). To achieve these two objectives, the central elements in a CANFLEX bundle were replaced with a large central graphite displacer. Plutonium at 2.6% (354 g per bundle) was mixed with thorium in the remaining 35 elements in the outer two fuel rings of the CANFLEX bundle. Enrichment grading in the outer two fuel rings would result in peak element ratings that are comparable to those in a 37-element bundle with NUE fuel. The resultant burnup was 30 MW-d/kg HE, a burnup for which there is research reactor experience with Pu-ThO<sub>2</sub> fuel. Void reactivity was 8.6 mk, which is judged to be acceptable with the current shutdown system. Computer simulations also showed that using SiC instead of graphite in the central displacer reduces the magnitude of the void reactivity somewhat.

Addition of a small amount of burnable poison to the central displacer would further reduce void reactivity, increase the plutonium loading per bundle as well as the absolute amount of plutonium destroyed, but would decrease the plutonium destruction efficiency. The plutonium destruction efficiency would be reduced from about 77% to 71%, by poison addition that reduces void reactivity from about 8.6 mk to zero.

### 15.10.3.7 Power Plant Optimization for Thorium Use: The Indian Advanced HWR

India is presently working on the design of an advanced HWR (AHWR), designed specifically for utilizing thorium, with reprocessing of spent thorium and refabrication of (Th, $^{233}\text{U}$ )O<sub>2</sub> fuel. The main constituent of the core is fuel made of MOXs of thorium and  $^{233}\text{U}$ . The reactor will be of the pressure tube type, with boiling light water as coolant. The discharge burnup of the ( $^{232}\text{Th}$ , $^{233}\text{U}$ )O<sub>2</sub> can be increased later on, depending upon fuel performance limits. The  $^{233}\text{U}$  enrichment in the thorium is so adjusted that the system will be self-sustaining in  $^{233}\text{U}$ . Because the burnup of the ( $^{232}\text{Th}$ , $^{233}\text{U}$ )O<sub>2</sub> is planned to be increased to fuel performance limits, and the  $^{233}\text{U}$  enrichment is determined by self-sustaining considerations, it is clear that this lattice would be subcritical. To make the reactor critical, a few driver zones are provided. These regions will consist of highly reactive (Th, Pu)O<sub>2</sub> fuel. This region forms a part of each cluster. To get negative void reactivity, and also to obtain the benefit of a hard spectrum for plutonium, the lattice pitch has been reduced.

This reactor also has 100% heat removal by natural circulation, and passive safety is ensured by engineered safety features like gravity-driven water pool, isolation condenser, and large volumes of water that can totally submerge the core in the event of an accident.

Briefly, the design objectives of the AHWR are

- About 75% of the power produced by the AHWR should be contributed by thorium.
- The system should have a negative void coefficient of reactivity.
- Discharge burnup of the fuel should be more than 20 MW-d/kg HE.
- Plutonium consumption and initial plutonium inventory should be as low as possible.
- The system should be self-sustaining in  $^{233}\text{U}$ .
- The total thermal power of the reactor should be 750 MW(th).

The fifth point indicates that the starting point will be the SSET cycle described in Section 15.10.3.4. To extend the burnup of the SSET, plutonium is used as makeup fissile material.  $^{239}\text{Pu}$  has a high capture-to-fission ratio in the soft spectrum of the HWR. To put plutonium in a more favorable spectrum, the lattice pitch is reduced, resulting in an undermoderated spectrum compared with the HWR.

Thorium oxide has very good fuel performance characteristics and is capable of achieving very high burnups. Because this has to be matched by reactivity considerations, the initial plutonium enrichment could be very high. This would have the undesirable consequence of too high a fraction of the power coming from plutonium. This has been addressed by concentrating the plutonium in a small number of pins. In the AHWR cluster, which is a 54-rod cluster, only 24 pins contain plutonium, the other 30 being  $(^{232}\text{Th}, ^{233}\text{U})\text{O}_2$ .

The thermal absorption of thorium is three times that of  $^{238}\text{U}$ . Due to this, the deleterious effects of parasitic absorption are less in thorium systems, and one can consider the use of light water coolant. This opens the way to in-core boiling. The reactor then has to be vertical, and then it becomes possible to design for 100% heat removal by natural circulation and passive safety. The possibility of positive void coefficient of reactivity has been countered by the lattice being under moderated with a burnable absorber in the fuel cluster.

To summarize, the Indian nuclear power program aims at maximizing the energy potential of the indigenous resources. Various fuel-cycle-related development programs in India address this objective. Because India has abundant thorium reserves, efforts are being made to increase the energy extracted from thorium. Thorium utilization in Indian HWRs has already been initiated with the use of  $\text{ThO}_2$  bundles for initial flux flattening in PHWRs. The proposed AHWR is another step toward the establishment of thorium cycles.

## 15.10.4 Waste Management Aspects

### 15.10.4.1 Physical Properties

Thoria-based fuels are appealing from a waste management perspective because  $\text{ThO}_2$  is chemically stable and almost insoluble in groundwater. By far the most important chemical difference between  $\text{ThO}_2$  and  $\text{UO}_2$  is the fact that thorium is present in its maximum oxidation state, Th(IV), whereas uranium is not. Therefore, oxidative dissolution of the matrix is not an issue with thoria fuel. Redox conditions could affect the leachability of  $^{233}\text{U}$  from irradiated thoria, but this would be limited to surface dissolution and is unlikely to be a major concern.

The inertness of thoria to oxidation is also relevant to interim dry storage of irradiated fuel before geological disposal. In contrast with  $\text{UO}_2$ , air oxidation of the fuel matrix in defective elements is not an issue with thoria-based fuels. Moreover, the thoria structure can easily accommodate oxidation of minor solid-solution components such as U and Pu. Thus fuel oxidation is unlikely to be a concern during dry storage of thoria-based fuels, and hence the maximum storage temperature would be limited by some other factor, probably cladding degradation (EPRI 1989).

The solubility of crystalline thoria in aqueous solution at  $25^\circ\text{C}$  and  $\text{pH} > 5$ , in the absence of complexing agents, is extremely low. The release of actinides and those fission products that are retained by the thoria matrix is expected to be limited by the solubility of  $\text{ThO}_2$ . Such release would be exceedingly slow in an engineered disposal vault of the type envisaged for CANDU  $\text{UO}_2$  fuel. No credible aqueous or geochemical process has been identified that would greatly accelerate  $\text{ThO}_2$  fuel-matrix dissolution under disposal conditions (Taylor et al. 1996).

#### 15.10.4.2 Radiotoxicity

One of the advantages often cited for the thorium fuel cycle is a significant reduction in the production of long-lived transuranic actinides. It has lower “radiotoxicity” than uranium-based fuels, and so the source term in the waste management vault will be lower.

However, in assessing the risk to man and the environment posed by radioactive spent fuel, it is important to consider not only the source term but also the pathways to exposure. Thus, in the Canadian concept for geological disposal of spent fuel, the actinides are not the major contributor to risk after tens or hundreds of thousands of years. This is because they are relatively insoluble in the reducing environment of the vault and are transported only slowly if they do dissolve.

Nonetheless, many proponents of the thorium cycle promote this feature, and indeed, many detractors of the nuclear industry cite the long-term radiotoxicity in spent uranium-based fuels as a major issue.

#### 15.10.5 Nonproliferation and Safeguards Considerations

In general, thorium cycles are all quite proliferation resistant for a variety of reasons. Prominent among those are that separated  $^{233}\text{U}$  is always accompanied by  $^{232}\text{U}$ .  $^{232}\text{U}$  is a copious emitter of alpha particles, an undesirable feature for a weapon. Moreover, the decay chain following the alpha emission of  $^{232}\text{U}$  to produce  $^{228}\text{Th}$  includes  $^{212}\text{Bi}$  and  $^{208}\text{Tl}$ , both hard gamma emitters. The presence of these two isotopes makes  $^{232}\text{U}$  a difficult material to handle and thereby to divert (although the  $^{208}\text{Tl}$  daughter product builds in with a half-life of several years, and so would not be present immediately in chemically separated uranium). Additionally, if thorium is mixed with  $^{238}\text{U}$  to begin with, the  $^{233}\text{U}$  that is produced will be contaminated with  $^{238}\text{U}$ , from which it cannot be separated by chemical separation. The proportions of thorium and  $^{238}\text{U}$  can be so adjusted that the final uranium does not have a fissile content exceeding 12%, or whatever other value is felt to be acceptably unsuitable for weapons. It is difficult to come up with a similar solution for the plutonium produced in the uranium cycle. If the thorium cycle involves addition of “topping” material to maintain a desired burnup, the use of “denatured” material would be another feature that would enhance the proliferation resistance of the cycle.

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## 15.11 Spent Fuel Disposal

### 15.11.1 Introduction

The simple fuel design of the HWR offers many benefits in spent fuel handling, storage and disposal. Over 20 years' experience in development and application of medium-term storage has resulted in technology that is well-proven and economical, with an extremely high degree of public and environmental protection. In fact, short-term (water-pool) and medium-term (dry canister) storage of spent HWR fuel is lower in cost per unit of energy than for other reactor types.

AECL's extensive research in permanent spent-fuel disposal has resulted in a defined concept for Canadian spent fuel disposal in crystalline rock. This concept was recently confirmed as "technically acceptable" by an independent environmental review panel (CEAA 1998). It is also now the basis of the spent fuel management policy adopted by the Government of Canada using an adaptive phased management approach, leading to placement in a centralized deep geological vault located in plutonic rock, with continuous monitoring and the potential for retrieval up to a future decision on closure of the repository (CNWMO 2005). Much of the technology behind the Canadian concept can be adapted to permanent land-based disposal strategies chosen by other countries. In addition, the Canadian development has established a baseline for HWR spent fuel permanent disposal costs. Canadian and international work has shown that the cost of permanent CANDU fuel disposal is similar to the cost of LWR fuel disposal, per unit of electricity produced.

### 15.11.2 Wet Storage of Spent HWR Fuel

After an HWR NUE fuel bundle is discharged from the reactor, after about one year of irradiation, it is removed to a pool system for interim storage. The water in the pool removes the residual heat produced by the spent fuel and provides radiation shielding for workers. The compact design of the HWR fuel bundle, and the impossibility of criticality for HWR natural uranium and low-enriched SEU spent fuel bundles under storage conditions in water pools (Frost 1994; Tsang et al. 1996) make for extremely simple pool storage.

To ensure protection of the environment and public health, the fuel spent pool in CANDU reactors has double concrete walls designed such that any leakage through the inner wall would enter drains between the walls and flow to the cleanup system. The radiation emitted by the spent fuel is shielded by the water, and the heat generated by the radioactive decay is transferred to the water. The water is cooled by circulating it through heat exchangers and is purified by filters and ion exchange systems that remove any dissolved and suspended radionuclides.

### 15.11.3 Dry Storage of Spent HWR Fuel

After spent HWR fuel has been out of the reactor for about 7 years, its activity and rate of heat generation (~5.4 W per bundle) have decreased sufficiently to allow the fuel to be transferred to dry storage, if desired. Compared with wet storage, dry storage is considered to have the advantages of

- Reduced amounts of radioactive waste, such as filters
- Smaller potential for contamination of the storage facility



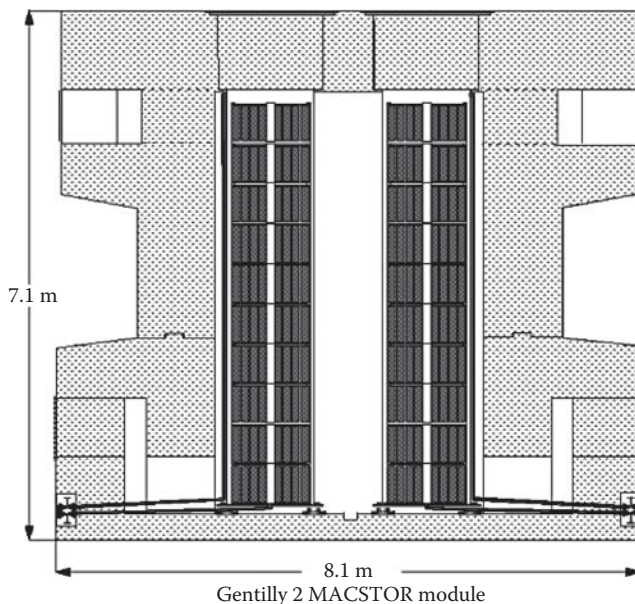
- Little or no corrosion of fuel sheaths
- Less radiation exposure to operating personnel (with triple containment of the spent fuel radioactivity)
- Minimal maintenance
- Low operating costs

Dry storage is simple to implement and modules can be added as needed.

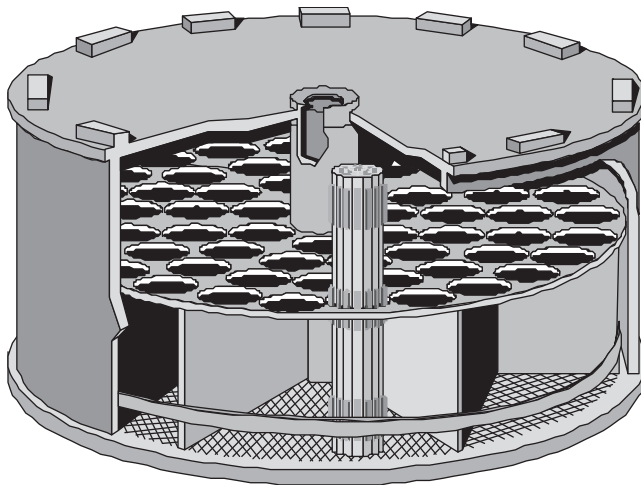
AECL began to study dry storage for spent nuclear fuel in the early 1970s. Silo-like structures called concrete canisters were first developed for the storage of research reactor enriched uranium fuel and then perfected for spent CANDU NUC fuel. By 1987, concrete canisters were being used for the safe and economical storage of all spent fuel accumulated during the operation of AECL's decommissioned prototype reactors. Each canister contains a stack of spent fuel baskets.

The same basic technology was then applied to on-site dry storage of spent fuel generated by operating CANDU power generating plants. The spent fuel baskets have air as the fuel cover gas. The basket is dried and welded in a shielded work station and transferred by a shielded transfer flask into the dry storage canister. The concrete canisters typically hold 9 baskets of 60 bundles with four or five canisters containing one year's worth of spent fuel.

In 1989, AECL began development (in cooperation with Transnuclear, Incorporated) of a monolithic, air-cooled, concrete structure for dry storage called MACSTOR (Figure 15.8). MACSTOR modules require less land area than concrete canisters for the same amount of spent fuel and are also suitable for storage of spent fuel assemblies from other reactor types (PWR, BWR, VVER) as well as CANDU. The MACSTOR modules store 12,000 bundles in 20 storage cylinders, each holding 10 baskets of 60 bundles (Figure 15.9).



**FIGURE 15.8**  
MACSTOR facility.



**FIGURE 15.9**  
Basket holding 60 CANDU spent fuel bundles.

In India, the HWR fuel bundles are also stored in the spent fuel storage bay after irradiation. These bays are designed to store the discharged fuel for 10 years. The water in the pool acts a cooling medium and also provides the required shielding. The fuel bundles from Rajasthan-1 reactor have been under such storage for more than 25 years, and show no signs of degradation. The storage of fuel bundles in concrete casks in dry air atmosphere is now being practiced at Rajasthan to enhance the spent fuel storage capacity. Normalized to a gross power of 1000 MW(e), dry storage costs for spent fuel from a CANDU reactor design have been compared with those for a typical LWR (Beaudoin and Pattantyus 1997). The indicated costs for the CANDU system were about 30% lower than the LWR system.

Like other countries, Canada is basing its plans for long-term management of spent nuclear fuel on placement in deep geologic structures—in the Canadian case, in stable crystalline rock of the Canadian Shield. In common with the approach adopted in other countries, the concepts developed by AECL and the Canadian Nuclear Waste Management Organization (CNWMO) entail isolating the spent nuclear fuel from the biosphere by a series of monitored engineered and natural barriers (AECL 1994). These barriers include: the spent fuel itself (in particular, the ceramic  $\text{UO}_2$  matrix); sealed, long-lived containers in which the spent fuel is placed; buffer materials to separate the containers from the surrounding rock and to control the movement of water to, and corrosion products away from, the container; monitoring of the geologic and engineered barrier performance for a period of time before eventual closure, and the use of seals and backfill materials to close the various openings, tunnels, shafts, and boreholes; and the rock mass in which the repository is located (geosphere).

#### 15.11.4 Summary

The HWR design is backed by well-established, economical means of short- and medium-term storage, and by more than 20 years' experience in defining and proving concepts for permanent disposal. For HWRs, wet pool storage has the advantages that fuel criticality concerns are eliminated (for natural and slight enrichments), and that pool fuel storage baskets can be moved directly to dry storage facilities. HWR dry spent fuel costs are

expected to be lower than for PWRs based on current studies, partly as a consequence of the more easily handled HWR fuel bundles. For permanent disposal, the Canadian concept of adaptive phased management leading to ultimate disposal provides an international demonstration of feasibility. It also establishes cost benchmarks, showing that permanent disposal costs for CANDU fuel would be comparable or less than the costs for PWRs.

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## *Waste Disposal: Transuranic Waste, High-Level Waste and Spent Nuclear Fuel, and Low-Level Radioactive Waste*

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Kenneth D. Kok, Joseph Heckman, and Murthy Devarakonda

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## 16.1 Introduction

Nuclear waste management (and the permanent disposal of nuclear waste in particular) is a complex and contentious issue. Classes of radioactive materials for which permanent disposal is a substantial challenge include transuranic (TRU) wastes (Section 16.2) and high-level wastes (HLWs) and spent nuclear fuel (SNF) (Section 16.3). The subject of low-level radioactive waste (LLRW) disposal is discussed in Section 16.6.

The process of determining a permanent disposal solution involves social, political, economic, and technical issues that compete and coalesce to determine the specific end state. Characteristics that make permanent disposal of TRU and HLW complex include contamination with long-lived radionuclides (e.g., plutonium-239 has a half-life of approximately 24,000 years) and high concentrations of radioactivity. While less contentious, LLRW disposal also presents challenges, as discussed in Section 16.6. The discussion in this section is primarily applicable to TRU and HLW disposal.

To ensure the protection of the health and safety of the public and the environment, a permanent disposal option for these wastes requires isolation for several thousand years, with a reasonable degree of predictive certainty as to the performance of the option during this multigenerational timeframe. In the United States, as early as 1957, the National Academy of Sciences (NAS) recommended the disposal of long-lived radioactive wastes in deep geologic formations (NAS-National Research Council, 1957). Other options for meeting the challenge of protecting the public and the environment from the risks of long-lived radioactive wastes have also been studied. For example, the development of reliable surface storage facilities may be used to isolate wastes provided that the resources required to operate and to maintain such complex facilities exist. This approach requires that the custody of storage facilities be passed on to generations that are committed to the continued maintenance of the facilities until a future permanent solution is identified. Disposal by ejection into space and emplacement under the Arctic ice cap have been considered but not seriously pursued due to the associated high risks and costs and unproven technologies (Institute for Energy and Environmental Research, 1992). While successful nuclear transmutation processes have been demonstrated as capable of reducing quantities of long-lived radioactive wastes by creating innocuous waste forms, waste modification is not a complete or practical solution with respect to all long-lived radionuclides at this time. Following two decades of study, the U.S. Department of Energy (DOE) concluded that disposal in a deep underground repository, as recommended by the NAS, is the preferred approach for TRU wastes as well as HLW and SNF. The result of this decision is a U.S. repository program composed of Waste Isolation Pilot Plant (WIPP) and the Yucca Mountain site. The WIPP is a DOE facility that has been operational since 1999 for defense-related TRU wastes (Public Law 102-579, as amended by Public Law 104-201). Plutonium-239 is the predominant radionuclide in wastes generated from defense-related activities. The Yucca Mountain site was designated by the U.S. Congress in 2002 as the site for an HLW and SNF repository (House Joint Resolution



87 2002), in accordance with Section 114 of the *Nuclear Waste Policy Act of 1982* (Public Law 97-425). DOE is currently preparing an application for review by the U.S. Nuclear Regulatory Commission (NRC) to proceed with the Yucca Mountain repository construction.

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## 16.2 TRU: Case Study of the WIPP

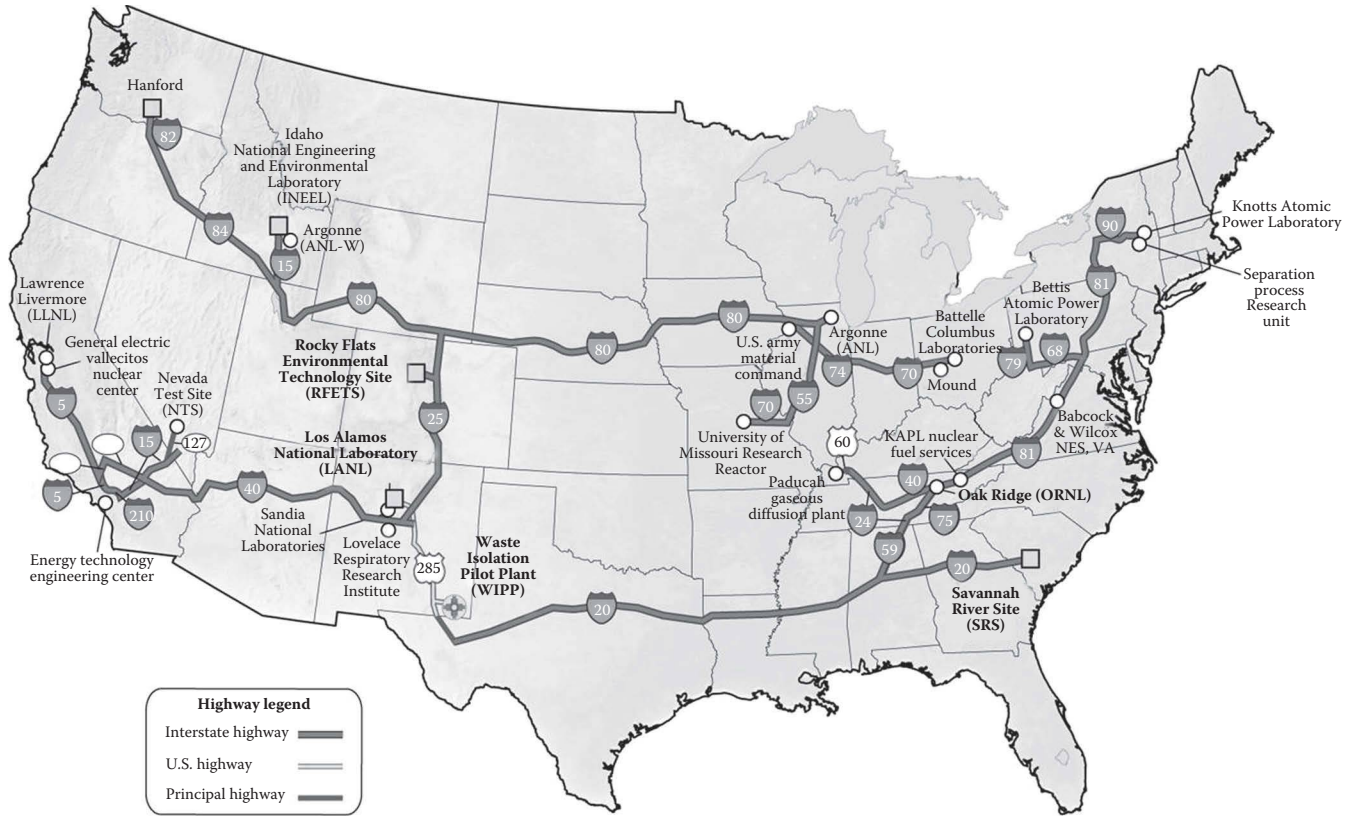
The underground WIPP repository is located in the southeastern part of the State of New Mexico, 41.84 km southeast of the city of Carlsbad. The site selected for WIPP is a 25.75 km<sup>2</sup> tract of federal land. The WIPP repository is located 655 m below the surface in a bedded salt formation of Permian age known as the Salado Formation. The U.S. Congress authorized the development of WIPP in 1979 to demonstrate the safe disposal of radioactive waste resulting from defense programs of the United States, including weapons production and research and development. The WIPP is authorized for the disposal of 175,500 m<sup>3</sup> of defense-related TRU wastes (Public Law 102-579, as amended by Public Law 104-201) and has an operational life of 35 years (until 2033).

TRU waste is defined as radioactive waste contaminated with alpha-emitting radionuclides with half-lives of >20 years and atomic numbers of >92, in concentrations >100 nCi/g of waste. TRU waste is produced during nuclear fuel assembly, nuclear weapons research, production, and cleanup, and as a result of reprocessing SNF. The waste generally consists of solid or solidified materials such as protective clothing, tools, glassware, and equipment contaminated with radioactive materials (Figure 16.1). Figure 16.2 shows the DOE sites that generate or store TRU waste around the country and the routes used to transport waste to WIPP (some sites in Figure 16.2 have been cleaned up).

The most common container configuration for the waste is a 0.208 m<sup>3</sup> drum, with some waste stored in larger drums or boxes. Within these containers, the waste is typically packaged in plastic bag layers for contamination control. TRU wastes are classified as contact handled (CH) or remote handled (RH) depending on the radiation dose rate at the



**FIGURE 16.1**  
Examples of TRU waste.



**FIGURE 16.2**  
DOE TRU waste sites.

surface of the waste container. CH-TRU waste containers have an external radiation dose rate  $\leq 2$  mSv/h at the surface of the container; RH-TRU waste containers have an external radiation dose rate  $>2$  mSv/h at the container surface. CH-TRU waste constitutes the vast majority (~97 vol. %) of the overall DOE TRU waste inventory anticipated for WIPP disposal.

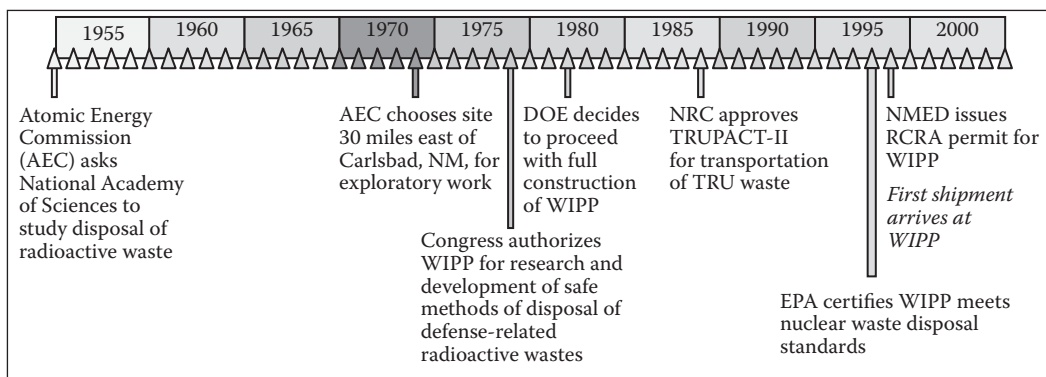
The WIPP, which was officially opened and first received a shipment of defense-related CH-TRU waste on March 26, 1999, has been operational for 10 years. WIPP has received over 7000 shipments of CH-TRU waste from nine DOE waste-generating and storage sites. Currently, WIPP is receiving and disposing of approximately 25 shipments per week, including RH-TRU waste.

The complex chronology of key decisions required to successfully obtain the current operational status of the WIPP repository showcases the influence of various stakeholders in determining the end state of the repository program. Figure 16.3 presents the historical decisions for achieving WIPP operational status on a timeline spanning more than four decades. As evident from this extensive timeline, the opening of WIPP was achieved only after the completion of technical research to the satisfaction of technical oversight groups and professional societies, the demonstration of compliance with governing regulations, the satisfactory negotiation of organizational agreements, the settlement of various lawsuits challenging regulatory authority, and the input and opinions made available by public comment periods. Figure 16.4 depicts the key stakeholders for the WIPP repository program. A detailed discussion of the evolution of the WIPP program is provided below and is organized by the following areas:

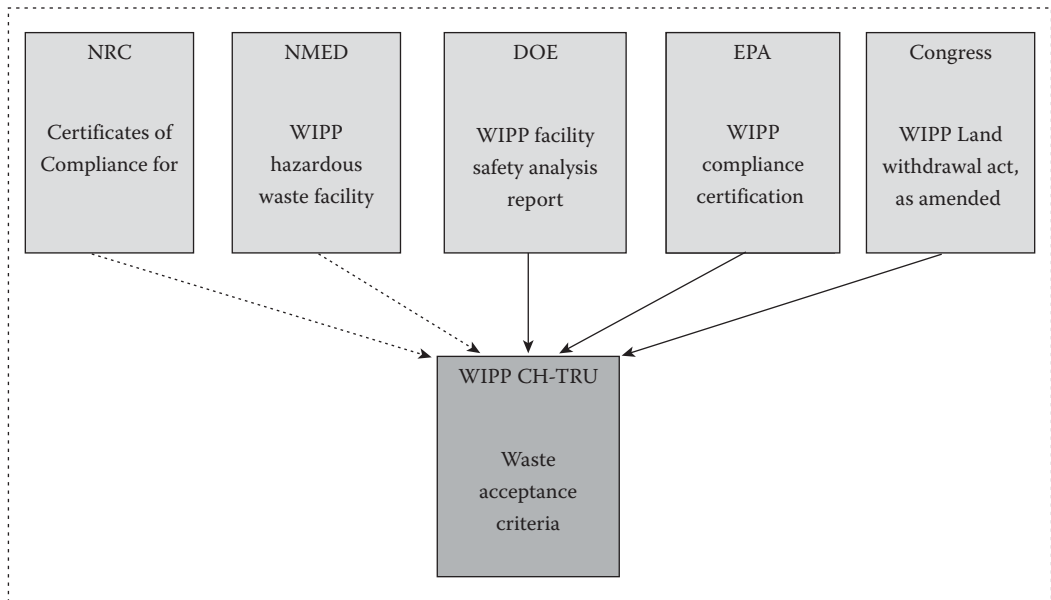
- Siting
- Land withdrawal
- Construction
- Regulatory process and approvals

### 16.2.1 Siting

Following the NAS recommendation for disposal of long-lived radioactive wastes in geologic formations (NAS-National Research Council, 1957), the Atomic Energy Commission (AEC) sponsored several years of research (1957–1961) at the Oak Ridge



**FIGURE 16.3**  
Historical decisions for achieving WIPP operational status.



**FIGURE 16.4**  
Historical decisions for achieving WIPP operational status.

National Laboratory (ORNL) on phenomena associated with the disposal of radioactive waste in salt formations. In 1962, the U.S. Geological Survey (USGS) reported on the distribution of domestic salt deposits that may be suitable for radioactive waste disposal. The Permian Basin, which includes the Delaware Basin in southeastern New Mexico and large areas in the States of Kansas, Texas, and Oklahoma, was one of the areas identified by the USGS. The research at ORNL was then expanded to include a large-scale field program in which simulated waste (irradiated fuel elements) supplemented by electric heaters was placed in an existing salt mine at Lyons, Kansas. The research of the Project Salt Vault field program (1963–1967) (Holdoway, 1972) led to the 1970 AEC selection of the Lyons site as a potential location for a radioactive waste repository. However, in 1972, the Lyons site was judged unacceptable due to the area's uncertain geology/hydrology and previously undiscovered drill holes that could lead to extensive dissolution of salt.

Following the investigation of several other potential Permian Basin sites, in 1973, the AEC, USGS, and ORNL selected a portion of the Permian Basin in southeastern New Mexico as best meeting their site selection criteria (Wentz, 1999). A location 48.28 km east of Carlsbad, New Mexico, was chosen for exploratory work and extensive field investigations. Based on results of initial investigation, in 1975, an area approximately 11 km southwest of the initial location was recommended by the USGS for further examination. Detailed site characterization and engineering design programs were initiated and continued for several years (Sandia Laboratories, 1978).

The key stakeholder interactions in the siting phase of the WIPP repository program were between the scientific community (NAS) and DOE, with the local community in the city of Carlsbad being supportive of the site selection process.

### 16.2.2 Land Withdrawal

The U.S. Bureau of Reclamation Land Withdrawal Handbook defines the term “withdrawal” as the withholding of an area of federal land from settlement, sale, location, or entry, under some or all of the general land laws. The withdrawal may be for the purpose of limiting activities under those laws to maintain other public values in the area or reserving the area for a particular public purpose or program or transferring jurisdiction over an area of federal land, other than “property” governed by the *Federal Property and Administrative Services Act*, as amended, from one department, bureau, or agency to another department, bureau, or agency. All withdrawal actions require coordination with the U.S. Department of the Interior (DOI) Bureau of Land Management (BLM).

To proceed with the WIPP site characterization, the U.S. Energy Research and Development Administration, which was a predecessor agency of DOE, filed an application in 1976 with the BLM for the withdrawal of 69.61 km<sup>2</sup> of land in Eddy County, New Mexico, for the WIPP project. The application effectively segregated the identified lands from public entry for a period of two years from the date of its noticing (*Federal Register*, 1976). This segregation was continued by BLM based on a subsequent DOE application made in 1978 (*Federal Register*, 1978).

Based on a later DOE application (*Federal Register*, 1980), the BLM issued Public Land Order (PLO) No. 6232, withdrawing 36.26 km<sup>2</sup> of federal land (and 51.8 km<sup>2</sup> of state trust land, if acquired by the federal government) for the purpose of conducting the Site and Preliminary Design Validation (SPDV) program at WIPP. This administrative withdrawal was effective from 1982 to 1990 (*Federal Register*, 1982). When DOE published the results of the SPDV program (USDOE, 1982), comments from the State of New Mexico cited various unresolved issues, including compensation for the State’s loss of mineral revenues. In 1988, the BLM issued to the State of New Mexico a land exchange conveyance document, which conveyed 10.2 km<sup>2</sup> of federal land in Eddy County (surface and mineral estate) in exchange for 5.18 km<sup>2</sup> of state trust lands (surface and mineral estate) located within the WIPP withdrawal area. As a result, all acreage (~41.44 km<sup>2</sup>) at the WIPP site was given to federal control administered by BLM (*Federal Register*, 1988). The existing mineral leases within the WIPP withdrawal area were settled in 1990 when DOE reached agreement with a New Mexico business regarding the purchase of its leasehold interests in a federal potash lease.

As a precursor to the construction of WIPP, in 1983, the BLM issued PLO No. 6403, withdrawing the same tract of land (~41.44 km<sup>2</sup>) for the purpose of DOE construction of full facilities at the WIPP site. This administrative withdrawal, also effective for an 8-year period (1983–1991), prohibited the “... use or occupancy of the lands hereby withdrawn for the transportation, storage, or burial of any radioactive materials ...” (*Federal Register*, 1983a, b).

In 1991, BLM issued PLO No. 6826 to modify PLO No. 6403, including the expansion of the order’s purpose to include the conduct of a Test Phase at WIPP using retrievable TRU waste (*Federal Register*, 1991a, b). PLO No. 6826 was challenged by *State of New Mexico v. Watkins* (Civil Action No. 91-2527 filed in U.S. District Court for the District of Columbia). The issue of whether DOI/BLM had violated the *Federal Land Policy and Management Act* by issuing a WIPP administrative withdrawal order was resolved in the State’s favor. A motion for summary judgment was granted on the basis that “... the Secretary of Interior cannot extend a withdrawal of WIPP for a new purpose not required by the purpose of the original withdrawal.” The original PLO (No. 6403) withdrew federal lands at the WIPP site for “construction” of the facility and prohibited the transport or emplacement of any

radioactive waste at WIPP. The Interior and Insular Affairs Committee of the U.S. House of Representatives approved a resolution aimed at nullifying the PLO No. 6826 citing environment, safety, and public health concerns. As a result, DOI/BLM modified PLO No. 6826 for the purposes of prohibiting until June 30, 1991, the transportation or emplacement of any radioactive waste within WIPP (*Federal Register*, 1991c).

Subsequently, committees of the U.S. House of Representatives and the U.S. Senate passed various versions of a WIPP land withdrawal bill. On October 30, 1992, the WIPP legislation was signed into law. Among its key provisions, the *WIPP Land Withdrawal Act* (Public Law 102-579) established prerequisites for initial receipt and permanent disposal of TRU wastes at WIPP. On September 23, 1996, the *National Defense Authorization Act for Fiscal Year 1997* (Public Law 104-201) was signed into law. This law contained amendments (Amendment #4085) to the 1992 *WIPP Land Withdrawal Act*, the import of which is discussed in Section 16.2.4.2.

The land withdrawal phase of the WIPP repository program involved active interaction among DOE, the U.S. government, and the State of New Mexico. The end result and ensuing conditions of land withdrawal were arrived at not necessarily by consensus but by active negotiation and dialogue between the stakeholders.

### 16.2.3 Construction

On December 29, 1979, the U.S. Congress approved the *Department of Energy National Security and Military Applications of Nuclear Energy Authorization Act of 1980* (Public Law 96-164). Section 213(a) of the *Act* authorized WIPP

for the express purpose of providing a research and development facility to demonstrate the safe disposal of radioactive wastes resulting from defense activities and programs of the United States. ...

In January 1981, DOE issued a Record of Decision to proceed with WIPP construction (*Federal Register*, 1981). In May 1981, the New Mexico Attorney General filed suit in the U.S. District Court against DOE and DOI, alleging violations of federal and state law in connection with the development of WIPP (Civil Action No. 81-0363 JB filed in the U.S. District Court [Albuquerque]). As a result, a Federal Court Order was issued that stayed all proceedings in the state lawsuit in accordance with a Stipulated Agreement, signed by the New Mexico Attorney General, DOE, and DOI, which required DOE to perform additional geotechnical studies at the WIPP site. Upon resolution of these issues, DOE announced its decision to proceed with full facility construction of WIPP (*Federal Register*, 1983b).

Underground excavation of the WIPP repository in the Salado formation salt beds began in the early 1980s. During siting and initial excavation activities, four shafts were drilled as discussed below.

- On July 14, 1980, DOE initiated drilling of a 3.66-m (diameter) exploratory shaft. Following extensive hydrological and geological evaluations, DOE redesigned the proposed repository with the disposal area relocated approximately 1828 m south of its original location. Salt from the underground excavation of the repository was transported to the surface via the exploratory shaft, also known as the construction and salt handling shaft.
- On December 22, 1981, DOE initiated drilling of a 1.83-m (diameter) ventilation shaft at the site. This second WIPP shaft reached a total depth of 671.5 m on February 22, 1982.

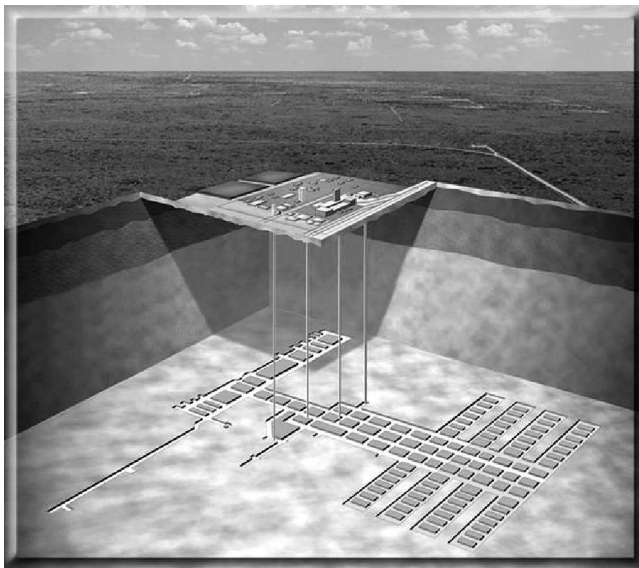
- On September 23, 1983, DOE initiated drilling of a 4.27-m (diameter) exhaust shaft. This third WIPP shaft reached a total depth of approximately 670.6 m on November 22, 1983.
- On May 1, 1988, DOE initiated drilling of a second ventilation shaft. This shaft was completed on April 17, 1989.

The repository disposal areas excavated in the salt beds of the formation are referred to as “panels” and “rooms.” As shown in Figure 16.5, there are a total of eight panels, each consisting of seven rooms and associated access drifts. Typical room dimensions are 10 m (width) × 4 m (height) × 91 m (length). To date, five panels (panels 1 through 5) have been excavated and three panels (panel 1 through 3) have been filled with waste. After disposal of the waste in the storage rooms, closure of the repository occurs due to the creep (plastic flow) of the surrounding salt formation. This creep is in response to the pressure gradient that exists between the far-field pressure away from the repository (referred to as the lithostatic pressure or the pressure at the depth of the repository due to overlying rock) and the pressure in the repository (which is initially at atmospheric pressure).

## 16.2.4 Regulatory Process and Approvals

### 16.2.4.1 Transportation

Initially, to transport CH-TRU waste to WIPP from the waste-generating and storage sites across the United States, the TRUPACT-I was designed by DOE in 1985 as a single-contained, vented rectangular packaging. To ensure compliance with the governing regulations at the time, the packaging was redesigned by DOE to incorporate double containment and to eliminate the venting feature. The redesigned packaging, the TRUPACT-II, was a right-circular-cylinder design with unvented double containment. The TRUPACT-II design was consistent with the Second Modification to the Consultation and Cooperation Agreement



**FIGURE 16.5**  
Schematic of WIPP.

(State of New Mexico-DOE, 1987). This modification required all waste shipments to WIPP to be completed in packages that NRC has certified for use. Similarly, the *WIPP Land Withdrawal Act* states that TRU waste may be transported to WIPP only in packagings certified by NRC (Public Law 102-579, as amended by Public Law 104-201).

The NRC regulations require that Type B packages remain leak-tight even under severe accidental conditions (10 CFR 71). The NRC approval process for these packages involves testing, analysis, definition of the waste inventory, and imposition of limits based on the safety analysis. The TRUPACT-II was certified as a Type B package by NRC in 1989 (NRC, 2005a). Since the opening of WIPP in 1999, NRC has certified as Type B the following additional transportation packagings for the shipment of TRU waste (NRC, 2005b, c, 2006):

- HalfPACT (first certified in 2000)
- 72-B Cask (first certified in 2000)
- 10-160B Cask (first certified in 2001)

These packagings are shown in Figure 16.6.

Specific programs have been established at each DOE waste-generating and storage site expected to ship waste to WIPP to ensure compliance with the transportation requirements imposed by NRC for the packagings. These programs are reviewed, audited, and formally approved by DOE prior to granting authorization to a site to ship to WIPP. The WIPP transportation program has a proven track record with >13.3 million safe transport km and a fleet of >100 Type B packagings. In addition to using only NRC-certified



**FIGURE 16.6**  
NRC licensed packagings for transportation of TRU waste.



transportation packages, all shipments are monitored by a satellite tracking system. Emergency response preparedness, as mandated by the *WIPP Land Withdrawal Act* (Public Law 102-579, as amended by Public Law 104-201), is a high priority in the WIPP program. DOE has trained emergency response teams in several states, under a variety of programs and in cooperation with other government agencies. The WIPP transportation program holds periodic transportation accident exercises to test the emergency response capabilities of DOE, Washington TRU Solutions, LLC (WIPP management and operating contractor), and the local police and fire departments along the transportation routes.

#### 16.2.4.2 Licensing

In September 1985, the U.S. Environmental Protection Agency (EPA) promulgated its “Environmental Radiation Protection Standards for Management and Disposal of SNF, High-Level and TRU Radioactive Wastes” (*Federal Register*, 1985). These standards were subsequently codified in Part 191 of Title 40 of the Code of Federal Regulations (40 CFR 191). In 1987, the U.S. Court of Appeals for the First District (Boston) vacated and remanded to EPA for reconsideration of Subpart B of 40 CFR Part 191. This action, deciding a legal challenge to the EPA standards by the Natural Resources Defense Council and others, impacted WIPP in that there were now no repository standards applicable to the project. However, the Second Modification to the Consultation and Cooperation Agreement (State of New Mexico-DOE 1987) required DOE to continue its performance assessment planning as though EPA’s 1985 repository disposal standards remained applicable.

In 1991, DOE notified DOI that WIPP was ready to begin the Test Phase. This notification certified that “...all environmental permitting requirements have been met by DOE for WIPP” as required under PLO No. 6403, as modified by PLO No. 6826. The DOI authorized DOE to transport and emplace wastes (after June 30, 1991) at the WIPP site under these PLOs (*Federal Register*, 1991d). Upon notification of the first shipment of waste to WIPP, the New Mexico Attorney General filed a lawsuit against DOE and DOI to stop the planned shipment (Civil Action No. 91-2527 filed in the U.S. District Court for the District of Columbia). The lawsuit alleged violations of the *National Environmental Policy Act*, the *Federal Land Policy and Management Act*, and the *Administrative Procedure Act* and provided documentation in support of a motion for both a Temporary Restraining Order and a preliminary injunction. The U.S. District Court issued an order granting the preliminary injunction and directed DOE to cease all activities relating to the WIPP Test Phase insofar as they involved the introduction or transportation of TRU waste into New Mexico. A subsequent 1992 order imposed a permanent injunction prohibiting the transport or disposal of any TRU waste at WIPP. DOE decided to conduct the Test Phase activities using radioactive waste in laboratories rather than underground at the WIPP site.

In 1992, the *WIPP Land Withdrawal Act* (Public Law 102-579) established prerequisites for initial receipt and permanent disposal of TRU wastes at WIPP. The *Act* also specified the statutory, regulatory, and other requirements and restrictions applicable to the WIPP facility and its operations. Significantly, EPA was designated as a primary independent regulator at WIPP with authority to determine whether the repository is suitable as a long-term disposal facility.

In 1993, EPA issued “Environmental Radiation Protection Standards for the Management and Disposal of SNF, High-Level and TRU Radioactive Wastes” codified at 40 CFR Part 191 (*Federal Register*, 1993a, b). In 1996, EPA issued a Final Rule establishing criterion for use in certifying whether WIPP complies with the standards set forth in 40 CFR Part 191

(*Federal Register*, 1996) codified at 40 CFR 194. These regulations established a definitive set of standards and criteria for use in determining the suitability of WIPP as a disposal facility.

On April 8, 1996, the New Mexico Attorney General filed a petition for the review of 40 CFR 194 (Civil Action No. 96–1107 filed in the U.S. Court of Appeals for the District of Columbia Circuit). This petition was ultimately consolidated with two other similar petitions filed by two environmental groups and two individuals (Civil Action No. 96–1108) and the Texas Attorney General (Civil Action No. 96–1109). The petitions alleged violations by EPA of the *WIPP Land Withdrawal Act* and the *Administrative Procedure Act* in promulgating the WIPP compliance criteria. The U.S. Court of Appeals for the DC Circuit denied the petitions and the final criteria remained as promulgated.

In 1998, under the regulations of 10 CFR Parts 191 and 194, EPA issued its final certification decision based on a DOE application that WIPP will comply with EPA's radioactive waste disposal regulations and is safe to contain TRU waste for 10,000 years (*Federal Register*, 1998). Following EPA certification, DOE notified the U.S. Congress that WIPP was ready to begin disposal operations and petitioned the U.S. District Court for the District of Columbia to lift the permanent injunction remaining from 1992, which barred the transport or introduction of any TRU waste at WIPP. Following the removal of this injunction, WIPP was officially opened and received the first shipment of TRU waste on March 26, 1999.

The EPA certification of WIPP imposes limits on the radionuclide inventory of the TRU waste as well as on specific waste components that affect the long-term properties of the repository (e.g., gas generation). As specified in the *WIPP Land Withdrawal Act*, EPA will continue to regulate WIPP until it closes (Public Law 102-579, as amended by Public Law 104-201). EPA will conduct a recertification every five years until closure to determine whether WIPP continues to comply with the radioactive waste disposal standards. DOE is required by regulation to report changes in activities or in conditions that have the potential for any releases from the repository. EPA may conduct inspections of activities at the WIPP repository and at other WIPP-related facilities (laboratories, waste generating, and storage sites) to verify continued compliance with the radioactive waste disposal standards.

#### **16.2.4.3 Mixed Wastes and the Complexity of Dual Regulation**

In 1986, EPA determined that mixed wastes, which are wastes containing hazardous and radioactive contaminants, are subject to regulation under the *Resource Conservation and Recovery Act* (RCRA; *Federal Register*, 1986). As such, for mixed wastes, the conventional hazardous waste component is regulated by EPA under RCRA and the radioactive waste component is separately regulated by NRC under the *Atomic Energy Act*. A significant portion of the wastes destined for WIPP disposal are mixed wastes and are subject to dual regulation under RCRA and the *Atomic Energy Act*.

RCRA establishes a comprehensive “cradle-to-grave” scheme for regulating hazardous wastes. As amended by the Hazardous and Solid Waste Amendments of 1984, the centerpiece of RCRA is an ambitious set of land disposal restrictions (LDRs). The LDRs prohibit land disposal of particular wastes as of specified dates unless such disposal is carried out in accordance with regulations issued by EPA. In 1996, DOE submitted a final No-Migration Variance Petition to EPA seeking a waiver from the LDR to dispose of mixed waste at WIPP. However, the 1996 amendments to the *WIPP Land Withdrawal Act* (Public Law 104-201) included a provision exempting WIPP mixed waste from the LDRs. This provision obviated the need for DOE to receive EPA approval of the pending petition.

Aside from the application of the LDRs to WIPP, the New Mexico Environment Department (NMED) was authorized by EPA in 1990 to regulate mixed wastes in the State of New Mexico in accordance with the RCRA implementation program defined by the *New Mexico Hazardous Waste Act* (20.4.1 NMAC). This state regulatory authority extends to the management, storage, and disposal activities to be conducted at the WIPP facility to meet requirements set forth in 20.4.1.500 NMAC (incorporating 40 CFR 264.13) (*Federal Register*, 1990).

As required by RCRA and the *New Mexico Hazardous Waste Act* (20.4.1 NMAC), DOE submitted an RCRA Part B Permit application to NMED for review in 1995. A draft RCRA Part B Permit for the storage and disposal of TRU mixed waste at WIPP was issued by NMED in 1998 for public comment. While the finalization of the WIPP RCRA Part B Permit by NMED was pending, EPA issued its final certification decision (*Federal Register*, 1998) and DOE notified the State of New Mexico that it intended to use WIPP to dispose of TRU waste that was “nonmixed.” DOE and NMED reached an agreement on a schedule and process for making a determination regarding whether a selected waste stream from the Los Alamos National Laboratory (LANL) was nonmixed waste. Based on confirmatory sampling and analysis results (DOE-LANL, 1998), NMED determined that the LANL waste stream was nonmixed waste. Consequently, the first shipment to WIPP consisted of nonmixed TRU waste and was completed prior to the NMED issuance of the final WIPP RCRA Part B Permit.

The final WIPP RCRA Part B Permit was issued by NMED in November 1999 and specified the characterization requirements and applicable limits for CH-TRU waste disposal at WIPP. Since this time, both mixed and nonmixed TRU wastes destined for WIPP disposal have been required to be characterized in accordance with the Permit established to ensure compliance with RCRA.

In addition to the requirements imposed by the repository licensing activities, the WIPP facility safety analysis established requirements for worker safety that have been incorporated into the repository waste acceptance criteria (WAC). The licensing phase of the WIPP repository program involved the most stakeholder participation, especially from the federal and state regulatory agencies, the public-at-large, and the State of New Mexico local community. These interactions defined conditions for WIPP operations.

### 16.2.5 WIPP Operation

WIPP received the first shipment of TRU on March 26, 1999. The material was shipped from Alamos. The first panel in the mine was filled in 2003. The final shipment from the DOE Rocky Flats site was received in 2005 and panel 2 of the mine was filled. In 2007, the shipment of RH-TRU (Remote Handled TRU) was begun and emplacement of this material in the repository was started. By the end of 2014, WIPP had received 11,894 shipments and emplaced 90,983.42 million cubic feet of waste.

On February 5, 2014, a haul truck, used to move salt from mining operations to the hoist which carried the mined salt above ground for storage, caught fire. The truck was dumping its load and the operator noticed an orange glow between the engine and the dump section of the truck. After expending the on-board extinguisher, the operator activated the fire suppression system on the vehicle which failed to extinguish the fire. All personnel were evacuated from the underground safely. Postevent evaluation showed several deficiencies related to equipment maintenance, emergency procedures, and system design.

On February 14, 2014, a second event occurred. This was a failure of an emplaced waste leading to a release of radioactive material to the environment outside of the mine. Late in

the evening a CAM monitor alarm indicated that there had been a release of radioactive material. Ventilation of the mine continued through HEPA filters and samples were collected from these filters on the surface. No personnel were working in the underground. Analysis of the filters showed high levels of alpha and beta radiation downstream of the HEPA filters confirming a release to the environment.

Investigation showed that the release was caused by the failure of a waste drum containing TRU waste from Los Alamos. It is postulated that a chemical reaction occurred between the liquid in the drum and the absorbent added to the drum prior to the drum being sealed and shipped to the WIPP facility. This reaction allowed a buildup of gases in the drum that overpressurized and failed leading to the ejection of radioactive material.

As a result of these events, the WIPP has not received or emplaced any waste since February 2014 and is not expected to resume normal operations until 2016 or later depending on the completion of facility modifications.

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### 16.3 HLW and SNF

HLW is defined as follows (10 CFR 63.2):

1. The highly radioactive material resulting from the reprocessing of SNF, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations
2. Irradiated reactor fuel
3. Other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation

#### 16.3.1 Proposed Repository for HLW

Even though the potential barrier of Congressional approval has been overcome with the U.S. Senate casting the final vote approving the development of the Yucca Mountain repository in 2002, several technical, regulatory, political, and societal challenges remain in the road to achieving operational status for the Yucca Mountain repository. The opening of WIPP in 1999 was an important step toward demonstrating the successful licensing and operation of a permanent geological repository for TRU waste in the United States. In opening WIPP, the DOE successfully navigated the complex maze of technical, regulatory, and societal requirements for regulatory approval and provides a valuable baseline for the evaluation of the proposed Yucca Mountain repository in technical, regulatory, and political terms. While the geologic characteristics and source terms differ, the principles of geologic disposal provide some basis for comparison of WIPP to Yucca Mountain. More significantly, the regulatory framework for the two programs is similar, and stakeholder concerns and involvement are expected to be comparable, given the “radioactive” and “permanent disposal” nature of both programs.

The role of the key agencies regulating Yucca Mountain and the status of the Yucca Mountain regulatory process are provided in Table 16.1. The Yucca Mountain project is at the stage where the regulatory landscape has just been defined, whereas WIPP is at the stage where all the regulatory approvals have been obtained and the operations are at full capacity.

**TABLE 16.1**  
Key Agencies Regulating Yucca Mountain

Agency	Role	Status
DOE	Determines suitability of a proposed disposal site; develops, builds, and operates site	In mid-November 2001, DOE amended the policies under the NWPA of 1982 for evaluating the suitability of Yucca Mountain as a site for development of a nuclear waste repository. On June 3, 2008, DOE submitted a license application to the NRC seeking approval to construct the repository. In 2010, DOE filed a motion to withdraw its application but this motion was denied. DOE began shutting down the Yucca Mountain project in 2009 and funding was terminated in 2011.
NRC	Licenses repository after determining whether DOE's proposed repository site/design comply with EPA standards	In early November 2001, NRC published licensing criteria for disposal of SNF and HLW in the proposed geologic repository at Yucca Mountain. In June 2008, NRC received the license application from DOE. The NRC began winding down activity related to Yucca Mountain in October 2010 and ceased review activity at the end of September 2011. The NRC resumed work on its technical and environmental reviews of the Yucca Mountain application using available funds in response to an August 2013 ruling by the U.S. Court of Appeals for the District of Columbia Circuit. The staff completed and published the final volumes of the safety evaluation report in January 2015.
EPA	Develops environmental standards to evaluate safety of repository	In June 2001, EPA published final radiation protection standards for the potential Yucca Mountain repository

## 16.4 Parameters Governing Public Acceptance of Repositories based on the WIPP Example

Based on the experience from the WIPP repository, the critical parameters that govern public and stakeholder interactions in the nuclear waste debate can be identified as follows:

- Demonstration of technical soundness
- Legal resolutions to stakeholder conflicts
- Economic impacts
- Cultural issues
- Drivers for need of program

### 16.4.1 Demonstration of Technical Soundness

One key lesson from the WIPP program that is applicable to the nuclear arena is that technical soundness, as demonstrated by compliance with all governing regulations, is absolutely essential, but entirely insufficient for gaining stakeholder confidence and public acceptance. The WIPP program is governed by multiple regulations, administered by NRC, EPA, NMED, and DOE, addressing various safety aspects of waste transportation, characterization, handling, and disposal. Each set of regulations includes provisions for ensuring public and stakeholder participation through public comment periods and other communications to

apprise the stakeholders of key milestones in the regulatory approval processes. While the technical aspects of the WIPP repository were generally concurred upon by project scientists, technical oversight groups, and regulators relatively early in the program, numerous stakeholders including various citizens' groups, the State of New Mexico, and the public at large continued to voice concerns pertaining to the environment and human health and safety during public comment opportunities. As an example of the level of stakeholder and public interest, the EPA received 1400 written and oral comments from more than 380 stakeholders at eight public hearings held around the United States during the compliance certification process for the WIPP repository (Biedscheid and Devarakonda, 2002).

In the interest of public acceptance, it is often necessary as part of the demonstration of technical soundness to perform activities that are not directly traceable to technical safety bases. WIPP repository program development decisions could not be based solely on technical grounds given the importance of public perception and stakeholder involvement. In some cases, initial conditions and requirements, although not justifiable solely on technical or economic grounds, may be perceived by impacted stakeholders as a commitment to perform the activity, which is solely related to their "comfort level" with the repository operations. For example, the application of dual regulations to the mixed waste (radioactive as well as hazardous) disposed at WIPP can be debated as being redundant, given the fact that the hazard from the radioactive nature of the waste overwhelms any impacts from the hazardous nature of the waste (as defined by RCRA) (Silva et al., 2003). While the dual regulation that mixed waste is subjected to may not be meaningful solely on a risk basis, the regulation of the waste under RCRA, as administered by the State of New Mexico, is crucial to the acceptance of the repository by stakeholders with local and state interests and is built into the regulatory process. Debate of technical issues alone has been inadequate in addressing stakeholder expectations, which in many cases are based on an opinion that more waste characterization results in safer conditions.

#### **16.4.2 Legal Resolutions to Stakeholder Conflicts**

As evidenced by the four decades of WIPP program evolution, legal resolutions to conflicting positions among stakeholders are to be expected and should be factored in as part of the decision-making process. For example, the land withdrawal phase of the WIPP repository program, an essential element in siting, constructing, and operating the repository, involved active negotiation among DOE, the U.S. government, and the State of New Mexico, with legal recourse sought by the participants on multiple occasions. The end result and ensuing conditions of land withdrawal were arrived at not necessarily by consensus but by interface and dialog between the stakeholders and the intervention of the judicial process. Similarly, a Cooperation and Consultation Agreement between the DOE and the State of New Mexico formalized the role of the State of New Mexico and included the negotiated settlement of key "off-site concerns" especially as related to transportation (e.g., emergency response, highway upgrading, transportation monitoring, and accident liability). Informal agreements without enforcement terms were not effective in seeking public acceptance of WIPP.

#### **16.4.3 Economic Impacts**

Following the demonstration of technical soundness and the realization of a threshold "comfort-level" among the public and stakeholders regarding project safety, economic interests can influence public acceptance. The opportunity for education and dissemination of

project knowledge is greatest at the local level. As a result, WIPP project communications are most effective in its local community of Carlsbad, New Mexico, and the supportive local community position of the project has been shaped by direct and frequent interactions with project representatives. The economic interests of Carlsbad, in the form of employment opportunities, augmentation of existing industries, and the addition of a new industry, influenced local public acceptance of the repository concept in the community. The local community also benefited from experience with a previous nuclear industry related project (McCutcheon, 2002) and continues to be informed by an active communications program for the project within the local community.

#### **16.4.4 Cultural Issues**

In pursuing public acceptance, attention and consideration must be given to “cultural issues.” In the U.S. nuclear repository and nuclear energy programs alike, cultural issues amount to the commonly negative public perception that exists for “all things nuclear,” a distrust of the government and the industry and, recently, fears of security risks and associated safeguards issues. These cultural issues are a function of the information provided to the public (through industry, citizens’ groups, new media, and entertainment). Changing these perceptions completely is an unrealistic and potentially undesirable end goal. The objective of public acceptance is not to get all stakeholders in complete agreement, but to support a process that results in an end-state that is inclusive and rigorous enough to withstand technical, political, legal, and public scrutiny. “Legitimacy” of a program may be realized through efforts to reach a “threshold level of stakeholder comfort” that include transparent decision making, encouraging stakeholders’ contributions, and conceding to perform activities that are important to stakeholders.

#### **16.4.5 Drivers for Need of Program**

The greater the need for a solution, the greater the likelihood of public acceptance for a solution. The need for the WIPP repository was driven by the fact that radioactive waste generated in the United States national interest from defense programs required a national solution, which imposed responsibilities on the national decision makers to ensure relevant legislative and executive actions occurred in pursuit of a defense waste disposal solution. This national solution incentive associated with WIPP-directed key decisions that established the program and compelled its progress.

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### **16.5 Nuclear Power and Public Acceptance**

The public and stakeholder input during the evolution of the WIPP program from conception to construction and their continuing participation in the operational phase of the project offer valuable insights to the issue of public acceptance of nuclear power and future power plant projects in the United States. Arguably, the time and environment are favorable to the reexamination of the nuclear energy option from a “need” and “desire” perspective. Reliance on energy imports is seen as a definite negative, and the rising prices of oil and gas place the “finite reserves” issue squarely in the public domain.

Unlike United States' experience with nuclear power prior to the 1979 Three Mile Island accident, the future exploration of nuclear energy will not be dominated by issues based solely on technical grounds. Nuclear power is a proven technology, but safety and reliability concerns of the public must be addressed as part of the public acceptance strategy. The public perception of nuclear power risks is the legacy of the Three Mile Island accident. Concessions and efforts not necessarily directly linked to safety issues may be necessary and should be part of the framework and mindset of any such program.

Legal resolution of conflicts and divergent positions should be expected and factored in as an essential decision-making tool. In addition to the necessary involvement of the judicial branch of the government, legislation supporting emission-free and sustainable energy would provide an essential foundation for promoting the future of nuclear energy and fostering public involvement and acceptance.

Finally, cultural issues frequently ingrained with beliefs and opinions that are nontechnical should not be ignored given the charged nature associated with the nuclear topic. It is unreasonable to expect or desire complete agreement among the stakeholders and unconditional acceptance by the public. The desired end result should be a nuclear energy program and policy that are perceived by all parties involved as legitimate. A public perception of legitimacy will provide the desired "threshold level of stakeholder comfort" to allow the complex process to move forward.

The time-scale framework for nuclear waste disposal projects is unparalleled. The technical basis for the program must continue to be evaluated and preserved during the multigenerational time frame as the composition of the administrators, technical experts, and public change. It is unlikely that those who initiate the planning of projects in the nuclear waste arena will be the same ones seeing these to completion. What is essential in a nuclear waste project from a technical perspective is knowledge preservation and maintenance of project history. Given the timeframes involved, this requires formalized efforts to transfer information across experts and generations. If continuity of technical knowledge is not maintained and lessons learned are not preserved, project progress may be negatively impacted. While programs will identify specific methods for conserving knowledge within the given program structure, beneficial efforts may include debriefing of retirees to address the aging workforce issues, training programs to transition technical knowledge, and maintaining a qualified core workforce. Equally important is the preservation of knowledge of factors that provide the basis for decision-making. These include the social, political, economic, and cultural realities that were in force at any given point in time.

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## 16.6 Overview of LLRW

LLRW is generated by nuclear power plants, educational institutions, industries, medical (hospital and research) facilities, and government entities. Typical LLRW consists of contaminated protective clothing, rags, mops, rubber, wood, concrete, steel, debris, filters, water treatment residues, equipment, tools, luminous dials, medical supplies, swabs, injection needles, syringes, laboratory animal carcasses, animal tissues, liquid scintillation vials, accelerator targets, and sealed sources. The levels of radioactive content can range from slightly above natural background levels to very high concentrations of radioactive materials and can involve virtually any radionuclide.



Prominent forms of LLRW generated by the nuclear power industry include

- Compactable trash (dry active waste such as paper and plastic)
- Noncompactable trash (glass and metal components such as pipes and tools)
- Ion exchange resins (from water treatment)
- Filter sludge (from water treatment)
- Concentrated liquids (from evaporation of wet reactor waste streams)
- Nonfuel irradiated components (e.g., reactor internals)
- Large equipment (e.g., steam generators, pressurizers, and reactor pressure vessels)

LLRW is typically stored on-site by licensees, either until it has decayed to levels that do not require regulated management so that it can be disposed of as ordinary trash or until amounts are large enough for shipment to an LLRW disposal site in containers approved by the U.S. Department of Transportation.

The regulations applicable to the management of LLRW and their respective scopes are summarized in Table 16.2.

### 16.6.1 Definition of LLRW

LLRW is variously defined by law and regulation in the United States. Under provisions of the Low-Level Radioactive Waste Policy Act (the Act) LLRW is defined as

The term “low-level radioactive waste” means radioactive material that—  
 (A) is not high-level radioactive waste, spent nuclear fuel, or byproduct material (as defined in section 2014(e)(2) of [USC Title 42]); and  
 (B) the Nuclear Regulatory Commission, consistent with existing law and in accordance with paragraph (A), classifies as low-level radioactive waste. (Public Law 99-240)

The NRC regulates LLRW under “Licensing Requirements for Land Disposal of Radioactive Waste” contained in Title 10, Part 61 of the Code of Federal Regulations (10 CFR 61). Under 10 CFR 61, LLRW is described as wastes

containing source, special nuclear, or byproduct material that is acceptable for disposal in a land disposal facility. For the purposes of this definition, low-level waste has the same meaning as in the Low-Level Waste Policy Act, that is, radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or byproduct material as defined in section 11e.(2) of the Atomic Energy Act (uranium or thorium tailings and waste). (10 CFR 61.2)

**TABLE 16.2**

Summary of Regulations Applicable to the Management of LLRW

Regulation	Content
10 CFR 20	“Standards for Protection against Radiation,” including requirements for shipping and disposing of LLRW
10 CFR 61	“Licensing Requirements for Land Disposal of Radioactive Waste,” including definitions of waste classes and requirements for waste form and characteristics
49 CFR 172	“Hazardous Materials Table, Special Provisions, Hazardous Materials Communications, Emergency Response Information, and Training Requirements”
49 CFR 173	“Shippers—General Requirements for Shipments and Packagings”

Thus, LLRW is defined more by what it is not than by what it is, that is, LLRW is defined by exclusion. LLRW is radioactive waste that is not HLRW, TRU waste, SNF, or byproduct material (commonly referred to as uranium mill tailings). All of these have separate legal definitions. Naturally occurring radioactive material and technologically enhanced NORM are also defined separately from LLRW but are not currently regulated under federal statute.

Some LLRW is subject to the hazardous waste management provisions of RCRA because it contains waste that is listed as hazardous waste or because it exhibits characteristics that make it a hazardous waste (refer to 40 CFR 261). If subject to hazardous waste regulations, such "mixed LLRW" must be managed to satisfy not only LLRW regulations but also hazardous waste regulations. Under certain restrictive conditions specified in 40 CFR 266, Subpart N, disposal of such mixed LLRW is allowed at licensed LLRW disposal facilities.

### 16.6.2 Classes of LLRW

The provisions of 10 CFR 61 define the four classes of LLRW that are characterized by the half-lives and concentrations of radionuclides contained in the waste. The requirements applicable to each class of LLRW are described in below.

- **Class A:** The physical form and characteristics of Class A waste must meet the minimum requirements summarized in Table 16.3 and be disposed of in disposal units separate from those for Class B and Class C LLRW at the disposal site unless it meets the 10 CFR 61 stability requirements summarized in Table 16.3. Class A LLRW contains types and quantities of radionuclides that will present an acceptable hazard to an inadvertent intruder (10 CFR 61.7(b)(2)).
- **Class B:** Class B LLRW must meet minimum requirements (Table 16.3) and the more rigorous physical form and characteristic requirements (Table 16.4). Class B waste must be stable (i.e., maintain its gross physical properties and identity) for 300 years after disposal (10 CFR 61.7(b)(2)). Stability can be provided (1) by the waste for itself, (2) by being disposed of in a container that is certified to be stable for 300 years, or (3) by being processed to render it into a stable waste form, according to guidance provided in NRC 1983 and NRC 1991 (10 CFR 61.56(b)(1)). Class B LLRW contains types and quantities of radionuclides that will decay during the 100 years after closure of the disposal facility, at which time it will present an acceptable hazard to an inadvertent intruder (10 CFR 61.7(b)(2)).
- **Class C:** Class C LLRW must meet minimum requirements (Table 16.3) and the more rigorous requirements on physical form (Table 16.4). Furthermore, Class C LLRW is waste that will not decay to levels which present an acceptable hazard to an inadvertent intruder within 100 years following closure of the disposal facility and must, therefore, be disposed of in a manner that protects against inadvertent intrusion (10 CFR 61.7(b)(3)).
- **Greater-than-Class-C:** Greater-than-Class-C (GTCC) LLRW is waste that is not generally acceptable for near-surface disposal. GTCC waste must have different forms and be disposed of using methods that are more stringent than those specified for Class C waste. Until otherwise provided through promulgation of regulatory provisions and requirements, such waste must be disposed of in a geologic repository as defined in Part 60 or 63 of 10 CFR. Disposal of GTCC waste is the responsibility of DOE.

### 16.6.3 Minimum Waste Requirements

All classes of LLRW disposed of at a facility licensed under 10 CFR 61 must satisfy the minimum waste form requirements stated in 10 CFR 61.56(a) and are summarized in Table 16.3.

### 16.6.4 Physical Form and Characteristic Requirements

The more rigorous physical form and characteristic requirements imposed on Class B and Class C LLRW are stated in 10 CFR 56(b) and are summarized in Table 16.4. Stability is intended to ensure that the waste does not structurally degrade and affect overall stability

**TABLE 16.3**

Basic Requirements That All Classes of LLRW Must Satisfy

Parameter	Requirement
Packaging	Waste must not be packaged for disposal in cardboard or fiberboard boxes.
Liquid waste	Liquid waste must be solidified or packaged in sufficient absorbent material to absorb twice the volume of the liquid.
Solid waste containing liquid	Solid waste containing liquid must contain as little free standing and noncorrosive liquid as is reasonably achievable, but in no case may the liquid exceed 1% of the volume.
Detonation, explosive decomposition, and reaction with water	Waste must not be readily capable of detonation or of explosive decomposition or reaction at normal pressures and temperatures, or of explosive reaction with water.
Toxic gases, vapors, and fumes	Waste must not contain, or be capable of generating, quantities of toxic gases, vapors, or fumes harmful to persons transporting, handling, or disposing of the waste, except as provided in for gaseous waste (10 CFR 61.56(a)(7)).
Pyrophoricity	Waste must not be pyrophoric. Pyrophoric materials contained in waste must be treated, prepared, and packaged to be nonflammable.
Gaseous waste	Waste in a gaseous form must be packaged at a pressure that does not exceed 1.5 atmospheres at 20°C. Total activity must not exceed 100 nCi per container.
Hazardous, biological, pathogenic, and infectious material	Waste containing hazardous, biological, pathogenic, or infectious material must be treated to reduce to the maximum extent practicable the potential hazard from the nonradiological materials.

Source: 10 CFR 61.56(a).

**TABLE 16.4**

Physical Form and Characteristic Requirements

Physical Form	Characteristic Requirements
Structural stability	Waste must be structurally stable. A structurally stable waste form must generally maintain its physical dimensions and its form under the expected disposal conditions (refer to guidance provided in NRC 1983 and NRC 1991). Structural stability can be provided by the waste form itself, processing the waste to a stable form, or placing the waste in a disposal container or structure that provides stability after disposal.
Liquid content	Liquid wastes or wastes containing liquid must be converted into a form that contains as little free standing and noncorrosive liquid as is reasonably achievable. In no case may the liquid exceed 1% of the volume of the waste when the waste is in a disposal container designed to ensure stability, or 0.5% of the volume of the waste for waste processed to a stable form.
Void space	Void spaces within the waste and between the waste and its package must be reduced to the extent practicable.

Source: 10 CFR 61.56(b).

of the disposal facility (that could lead to water infiltration) through slumping, collapse, or other failure of the disposal unit. Stability is also a factor in limiting exposure to an inadvertent intruder because it provides a recognizable and nondispersible waste.

### 16.6.5 Waste Classification

Classifying LLRW under 10 CFR 61.55 is a two-stage process that considers the concentrations of long-lived and short-lived radionuclides. Classification by long-lived radionuclides is controlled by Table 16.5 (taken from 10 CFR 61.55 Table 1). Where mixtures of long-lived radionuclides are involved, the “sum of fractions” rule applies. A fraction is calculated for each long-lived radionuclide as the quotient of its actual concentration in the LLRW and its respective concentration limit stated in Table 16.5. The sum of all such long-lived radionuclide fractions for a given container of LLRW determines its classification as follows:

- If the sum of fractions does not exceed 0.1, the container of LLRW is Class A.
- If the sum of fractions exceeds 0.1 but does not exceed 1.0, the container of LLRW is Class C.
- If the sum of fractions exceeds 1.0, the container of LLRW is not generally acceptable for near-surface disposal.

Classification by short-lived radionuclides is controlled by Table 16.6 (taken from 10 CFR 61.55 Table 2). Classification by short-lived radionuclides involving mixtures of radionuclides is similar to classification by long-lived radionuclides with mixtures of radionuclides: the “sum of fractions” rule applies by column. Three fractions are calculated for each short-lived radionuclide as quotients of its actual concentration in the LLRW and its respective concentration limits stated in columns 1–3 of Table 16.6. The sums of all short-lived radionuclide fractions for a given container of LLRW determine its classification as follows:

- If the sum of fractions based on the values in column 1 does not exceed 1.0, the container of LLRW is Class A.
- If the sum of fractions based on the values in column 2 does not exceed 1.0, the container of LLRW is Class B.

**TABLE 16.5**

Concentration Limits by Long-Lived Radionuclides

Radionuclide	Concentration (Ci/m <sup>3</sup> )
C-14	8
C-14 in activated metal	80
Ni-59 in activated metal	220
Nb-94 in activated metal	0.2
Tc-99	3
I-129	0.08
Alpha emitting transuranic nuclides with half-life >5 years	100 <sup>a</sup>
Pu-241	3,500 <sup>a</sup>
Cm-242	20,000 <sup>a</sup>

<sup>a</sup> Units are nCi/g.

**TABLE 16.6**

Concentration Limits by Short-Lived Radionuclides

Radionuclide	Concentration (Ci/m <sup>3</sup> )		
	Column 1	Column 2	Column 3
Total of all nuclides with less than 5 year half-life	700	( <sup>a</sup> )	( <sup>a</sup> )
H-3	40	( <sup>a</sup> )	( <sup>a</sup> )
Co-60	700	( <sup>a</sup> )	( <sup>a</sup> )
Ni-63	3.5	70	700
Ni-63 in activated metal	35	700	7000
Sr-90	0.04	150	7000
Cs-137	1	44	4600

<sup>a</sup> No limits are established for these radionuclides in Class B or C wastes. Practical considerations such as the effects of external radiation and internal heat generation on transportation, handling, and disposal will limit the concentrations for these wastes. These wastes shall be Class B unless the concentrations of other nuclides in Table 16.6 determine the waste to the Class C independent of these nuclides.

- If the sum of fractions based on the values in column 3 does not exceed 1.0, the container of LLRW is Class C.
- If the sum of fractions based on the values in column 3 exceeds 1.0, the container of LLRW is not generally acceptable for near-surface disposal.

Where LLRW contains a mixture of both long-lived and short-lived radionuclides (Table 16.5 and 16.6, respectively), the LLRW is classified as follows:

- If the sum of fractions for long-lived radionuclides does not exceed 0.1, the class is that determined considering short-lived radionuclides only.
- If the sum of fractions for long-lived radionuclide exceeds 0.1 but does not exceed 1.0, the waste is Class C, provided the sum of fractions for short-lived radionuclides based on column 3 of Table 16.6 does not exceed 1.0.

The concentration of a radionuclide may be determined by indirect methods such as those using scaling factors that relate the inferred concentration of one radionuclide to another that is easily measured, or radionuclide material accountability, if there is reasonable assurance that the indirect methods can be correlated with actual measurements. The concentration of a radionuclide may be averaged over the volume of a container of LLRW, or weight of a container of LLRW if the units are expressed as nCi/g.

### 16.6.6 Shipping LLRW

Requirements for shipping LLRW designated for disposal are stated in “Standards for Protection against Radiation” that is contained in Part 20 of the Code of Federal Regulations Title 10. Section 20.2006 and Appendix G of 10 CFR 20 present requirements for such transfers. Requirements for shipping LLRW are designed to control transfers of LLRW by any waste generator, waste collector, or waste processor licensee who ships LLRW (either directly or indirectly through a waste collector or waste processor) to a licensed LLRW disposal facility (as defined in 10 CFR 61). These requirements also establish a manifest tracking system (Appendix G of 10 CFR 20). LLRW shipping requirements are summarized in Table 16.7. One example of this manifest is included as Figure 16.7.

**TABLE 16.7**

Requirements for Transferring LLRW

Parameter	Requirement
Use of manifest	A licensee shipping LLRW ultimately intended for disposal at a licensed LLRW disposal facility must document the information required on NRC's Uniform Low-Level Radioactive Waste Manifest. The licensee must also transfer this recorded manifest information to the intended consignee in accordance with Appendix G to 10 CFR 20.
Generator's certification	Each shipment manifest must include a certification by the waste generator as specified in Section II to Appendix G of 10 CFR 20.
Persons involved in LLRW transfer and disposal	Each person involved in the transfer for disposal and disposal of waste, including the waste generator, waste collector, waste processor, and disposal facility operator, must comply with the requirements specified in Section III of Appendix G to 10 CFR Part 20.

Source: 10 CFR 20.

An authorized representative of the waste generator, processor, or collector must certify that the materials being transported are properly classified, described, packaged, marked, and labeled and are in proper condition for transportation according to the applicable regulations of the Department of Transportation and the NRC. The authorized representative does this by signing and dating the uniform LLRW manifest. Information that must be provided on the Uniform Low-Level Radioactive Waste Manifest includes that summarized in Table 16.8.

In certain circumstances, the manifesting requirements of 10 CFR 20.2006 need not be satisfied:

- When LLRW is shipped for processing, after which it will be returned to the shipper (i.e., for storage under their license) prior to disposal at a licensed LLRW disposal facility
- When LLRW is returned to the licensee who is the "waste generator" or "generator," as defined in 10 CFR 20
- When radioactively contaminated material that becomes the processor's "residual waste" is shipped to a "waste processor"

### 16.6.7 LLRW Disposal in the United States

Under provisions of the Act, each state was made responsible for disposal of its own LLRW and was encouraged to form compacts between states to manage LLRW on a regional basis. Initial efforts to form compacts resulted in more numerous compacts than envisioned in the passage of the Act, raising questions about the economic and practical viability of the smaller disposal facilities that would result.

After initial efforts to develop regional disposal facilities revealed the need for additional legislative specification, Congress passed the Low-Level Radioactive Waste Policy Amendments Act (the Amendments Act) that authorized a compact with an operating site to disallow the importation of LLRW generated in states outside the compact beginning in 1993 (Public Law 99-240).

Under provisions of the Act and the Amendments Act and subsequent congressional action, compacts of states have been organized for the purpose of controlling the disposal of LLRW generated within member states of each compact and to facilitate the



**TABLE 16.8**

## Information Required on Uniform LLRW Manifest

<b>Type of Information</b>	<b>Requirement</b>
General information	Name, facility address, and telephone number of the licensee shipping the waste Explicit declaration indicating whether the shipper is acting as a waste generator Name, address, and telephone number, or the name and EPA identification number for the carrier transporting the waste
Shipment information	Date of the waste shipment Total number of packages/disposal containers in the shipment Total disposal volume and disposal weight in the shipment Total radionuclide activity in the shipment Activity of each of the radionuclides H-3, C-14, Tc-99, and I-129 contained in the shipment Total masses of U-233, U-235, and plutonium in special nuclear material, and the total mass of uranium and thorium in source material
Disposal container and waste information	Alphabetic or numeric identification that uniquely identifies each disposal container in the shipment Physical description of the disposal container, including the manufacturer and model of any high-integrity container Volume displaced by the disposal container Gross weight of the disposal container, including the waste Maximum radiation level at the surface of each disposal container consigned to an LLRW disposal facility Physical and chemical description of the waste Total weight percentage of chelating agent for any waste containing more than 0.1% chelating agent by weight, plus the identity of the principal chelating agent Approximate volume of waste within a container Sorbing or solidification media, if any, and the identity of the solidification media vendor and brand name Identities and activities of individual radionuclides contained in each container, the masses of U-233, U-235, and plutonium in special nuclear material, and the masses of uranium and thorium in source material. For discrete waste types (i.e., activated materials, contaminated equipment, mechanical filters, sealed source/devices, and wastes in solidification/stabilization media), the identities and activities of individual radionuclides associated with or contained on these waste types within a disposal container shall be reported Total radioactivity within each container Classification of LLRW consigned to an LLRW disposal facility pursuant to 10 CFR 61.55. Waste not meeting the structural stability requirements of 10 CFR 61.56(b) must be identified
Uncontainerized waste information	Approximate volume and weight of the waste Physical and chemical description of the waste Total weight percentage of chelating agent if the chelating agent exceeds 0.1% by weight, plus the identity of the principal chelating agent Classification of the waste consigned to a disposal facility pursuant to 10 CFR 61.55. Waste not meeting the structural stability requirements of 10 CFR 61.56(b) must be identified Identities and activities of individual radionuclides contained in the waste, the masses of U-233, U-235, and plutonium in special nuclear material, and the masses of uranium and thorium in source material Maximum radiation levels at the surface of the wastes consigned to an LLRW disposal facility
Multigenerator disposal container information	Numerous requirements stated in Section I.E of 10 CFR 20, Appendix G
Control and tracking	Numerous requirements stated in Section II of 10 CFR 20, Appendix G

Source: 10 CFR 20, Appendix G.



development of regional LLRW disposal facilities. Of the several compacts that have been authorized, only a few are currently actively functioning to achieve the purpose of facilitating development of regional LLRW disposal facilities for which they were authorized.

Through the 1980s and 1990s, 10 compacts were formed and authorized, as listed in Table 16.9. Despite the expenditure of nearly \$600 million over an 18-year period under the compact system seeking to site and develop regional LLRW disposal facilities (GAO, 1999), only one new LLRW disposal facility came into operation during that period of time, and it happened completely outside the auspices of the compact system: the EnergySolutions facility located at Clive, Utah. In 2012, the only LLRW disposal facility sited and operated through the LLRW compact system was licensed. The Waste Control Specialists (WCS) facility in Andrews County, Texas, received a disposal license for the Compact Waste Facility (CWF) making it the only facility to be sited and licensed as a part of the compact system. The WCS CWF received the first shipment on April 27, 2012 (Waste Control Specialists, 2012).

Prior to the licensing and operation of the WCS CWF, access to facilities for the disposal of LLRW in the United States was limited and not generally available. Generators in all states have access to the EnergySolutions facility for disposal of Class A LLRW, provided authority to export LLRW from their respective compacts is granted by their respective compacts. Only generators in states that are members of the Atlantic, Northwest, and Rocky Mountain compacts have access to LLRW disposal facilities for disposal of Class A, Class B, and Class C LLRW, as shown in Table 16.10. Because most generators have no alternative disposal of Class B and Class C LLRW, they must find alternative waste management methods, cease its generation (not practical in most cases), ship it for off-site storage (of which very limited capacity presently exists), or store it in licensed on-site facilities.

As a result of the limited access to disposal for Class B and Class C waste, generators and waste processors developed alternative waste management strategies for high-activity

**TABLE 16.9**  
LLRW Disposal Compacts Authorized by Congress

Compact	Member States <sup>a</sup>
Appalachian	Delaware, Maryland, <u>Pennsylvania</u> , and West Virginia
Atlantic	Connecticut, New Jersey, and <u>South Carolina</u>
Central	Arkansas, Kansas, Louisiana, Nebraska, and Oklahoma
Central Midwest	Kentucky and <u>Illinois</u>
Midwest	Indiana, Iowa, <u>Michigan</u> <sup>b</sup> , Minnesota, Missouri, Ohio, and Wisconsin
Northwest Interstate Compact	Washington, Alaska, Hawaii, Idaho, Montana, Oregon, Utah, and <u>Wyoming</u>
Rocky Mountain Compact	<u>Nevada</u> , Colorado, and New Mexico
Southeast	Alabama, Florida, Georgia, Mississippi, <u>North Carolina</u> <sup>b</sup> , Tennessee, and Virginia
Southwest	Arizona, <u>California</u> , North Dakota, and South Dakota
Texas	Maine <sup>b</sup> , <u>Texas</u> , and Vermont
States not presently affiliated with a compact	District of Columbia, New York, Massachusetts, Michigan, Rhode Island, New Hampshire, and North Carolina

<sup>a</sup> State designated originally to host the first regional LLRW disposal facility underscored.

<sup>b</sup> Subsequently withdrew from compact.

**TABLE 16.10**

Operating and Proposed Commercial LLRW Disposal Facilities

Facility Location	Owner	States with Access	Licensed for Disposal of LLRW Classes
Andrews County, Texas <sup>a</sup>	Waste control specialists	Texas and Vermont	Class A, Class B, and Class C
Barnwell, South Carolina	EnergySolutions	Connecticut, New Jersey, and South Carolina	Class A, Class B, and Class C
Clive, Utah	EnergySolutions	All, provided export authority is granted by appropriate compact	Class A only
Richland, Washington	U.S. Ecology	Alaska Colorado, Hawaii, Idaho, Montana, New Mexico, Oregon, Nevada, Utah, Washington, and Wyoming	Class A, Class B, and Class C

<sup>a</sup> Proposed and in licensing process at the date of this writing.

LLRW waste streams. To address Class B and Class C waste disposal issues for the nuclear power industry, two new waste processing services have been implemented.

The first is a filter shredding service offered by WMG. In accordance with the 1995 NRC Low Level Waste Branch Technical Position on Concentration Averaging and Encapsulation, cartridge filters are typically considered individually for waste classification. Some of the cartridge filters have radionuclide concentrations above the Class A concentration. WMG receives containers of cartridge filters. The filters are processed through a shredder and mixed with a concrete grout. The output of the process is a disposal container with a homogenous solidified waste matrix that meets the Class A concentration limits.

The second process is a resin blending and processing approach through SempraSafe, a joint venture between waste management companies EnergySolutions and Studsvik. In this process, resin from nuclear power plants with radionuclide concentrations above the Class A limits are combined with resin containing lower concentrations of radioactivity and the resulting mixture is reformed at the Studsvik facility. The final product, reformed residue, meets the Class A concentration limits and may be disposed accordingly at the EnergySolutions Clive disposal facility.

For other high-activity LLRW waste streams (e.g., irradiated hardware) most generators have opted for on-site licensed storage for other than Class A LLRW, using the EnergySolutions facility in Clive, Utah, for disposal of Class A LLRW.

The failure of the LLRW compact system has led to the significant limitation on availability of LLRW disposal capacity described above. The need for additional federal action to address this limitation has been recognized but, to date, no serious initiative has resulted to resolve the issue (GAO, 2004).

Each operating disposal facility operates under a license that requires WAC. These WAC vary among the three operating facilities, and copies are available from each facility operator. With the exception of the EnergySolutions facility, all containers of LLRW are disposed of in steel-reinforced concrete canisters or overpacks. In the EnergySolutions case, bulk waste (uncontainerized) disposal is authorized for certain LLRW, and other LLRW is disposed of in pyramids formed by filling voids between waste containers with grout or controlled low-strength material.

Only EnergySolutions operates a facility licensed under 10 CFR 61 and permitted under 40 CFR 264 to receive, store, treat, and dispose of mixed LLRW.

The price of LLRW disposal at existing facilities is determined under rules that vary from state to state. Disposal prices at Barnwell are set for members of the Atlantic Compact by the South Carolina Budget and Control Board (SCBCB, 2008). Pricing for disposal of Class A LLRW at the Clive facilities is not publicly regulated and is negotiated with the generator by the facility owner. At Richland, disposal prices are regulated by the Washington Utilities and Transportation Commission (WUTC, 2008). Typical LLRW disposal prices range from a few hundred dollars to more than \$1000 per cubic foot of waste, depending upon characteristics such as container weight, radionuclide inventory, and radiation level.

### **16.6.8 Licensing Requirements for LLRW Disposal Facilities**

The NRC has developed and implemented in cooperation with agreement state a comprehensive system for licensing and regulating commercial LLRW management facilities (refer to 10 CFR 61). This system includes numerous requirements and features that provide confidence that the facilities licensed under its provisions will protect facility workers, members of the general public, and the environment as required. These requirements and features are divided among

- Performance objectives
- Waste characteristics requirements
- Siting requirements
- Design requirements
- Operating and closure requirements
- Environmental monitoring requirements

These requirements and features are summarized below.

#### **16.6.8.1 Performance Objectives (10 CFR 61, Subpart C)**

- Concentrations of radioactive material that may be released to the general environment in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 millirem (mrem) to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ of any member of the public.
- Reasonable efforts should be made to maintain releases of radioactivity in effluents to the general environment as low as is reasonably achievable (ALARA).
- Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in 10 CFR 20, except for release of radioactivity in effluents from the land disposal facility, which are governed as stated immediately above.
- Every reasonable effort should be made to maintain radiation exposures ALARA.
- Design, operation, and closure of the land disposal facility must ensure protection of any individuals inadvertently intruding into the disposal site and occupying the site or contacting the waste after active institutional controls over the disposal site are removed.
- The disposal facility must be sited, designed, used, operated, and closed to achieve long-term stability of the disposal site and to eliminate, to the extent practicable, the need for ongoing active maintenance of the disposal site following closure so that only surveillance, monitoring, or minor custodial care are required.

As a result of regulations promulgated by EPA, an additional general requirements supplements the performance objectives stated in 10 CFR 61 (40 CFR 141.66).

- A committed effective dose equivalent or total effective dose equivalent to any member of the public no greater than 4 mrem may result annually from exposure to and consumption of groundwater contaminated by releases from an LLRW disposal facility.

#### **16.6.8.2 Site Suitability Requirements (10 CFR 61.50)**

- The primary emphasis in disposal site suitability is given to isolating wastes and to disposal site features that ensure that the long-term performance objectives are met.
- The disposal site must be capable of being characterized, modeled, analyzed, and monitored.
- Within the region where the facility is to be located, a disposal site should be selected so that projected population growth and future developments are not likely to affect the ability of the disposal facility to meet the performance objectives of 10 CFR 61, Subpart C.
- Areas must be avoided having known natural resources which, if exploited, would result in failure to meet the performance objectives of 10 CFR 61, Subpart C.
- The disposal site must be generally well drained and free of areas of flooding or frequent ponding.
- Waste may not be disposed of in a 100-year flood plain, coastal high-hazard area, or wetland, as defined in Executive Order 11988, "Floodplain Management Guidelines."
- Upstream drainage areas must be minimized to decrease the amount of runoff that could erode or inundate waste disposal units.
- The disposal site must provide sufficient depth to the water table that ground water intrusion, perennial or otherwise, into the waste will not occur.
- The hydrogeologic unit used for disposal must not discharge ground water to the surface within the disposal site.
- Areas must be avoided where tectonic processes such as faulting, folding, seismic activity, vulcanism, or similar phenomena may occur with such frequency and extent to significantly affect the ability of the disposal site to meet the performance objectives of 10 CFR 61, Subpart C, or may preclude defensible modeling and prediction of long-term impacts.
- Areas must be avoided where surface geologic processes such as mass wasting, erosion, slumping, landsliding, or weathering occur with sufficient frequency and extent to significantly affect the ability of the disposal site to meet the performance objectives of 10 CFR 61, Subpart C, or may preclude defensible modeling and prediction of long-term impacts.
- The disposal site must not be located where nearby facilities or activities could adversely impact the ability of the site to meet the performance objectives of 10 CFR 61, Subpart C, or significantly mask the environmental monitoring program.

**16.6.8.3 Design Requirements (10 CFR 61.51)**

- Site design features must be directed toward long-term isolation and avoidance of the need for continuing active maintenance after site closure.
- The disposal site design and operation must be compatible with the disposal site closure and stabilization plan and lead to disposal site closure that provides reasonable assurance that the performance objectives of 10 CFR 61, Subpart C, will be met.
- The disposal site must be designed to complement and improve, where appropriate, the ability of the disposal site's natural characteristics to assure that the performance objectives of 10 CFR 61, Subpart C, will be met.
- Covers must be designed to minimize, to the extent practicable, water infiltration, to direct percolating or surface water away from the disposed waste, and to resist degradation by surface geologic processes and biotic activity.
- Surface features must direct surface water drainage away from disposal units at velocities and gradients that will not result in erosion that will require ongoing active maintenance in the future.
- The disposal site must be designed to minimize to the extent practicable the contact of water with waste during storage, the contact of standing water with waste during disposal, and the contact of percolating or standing water with wastes after disposal.

**16.6.8.4 Operating and Closure Requirements (10 CFR 61.52)**

- Wastes must be emplaced in a manner that maintains the package integrity during emplacement, minimizes the void spaces between packages, and allows the void spaces to be filled.
- Void spaces between waste packages must be filled with earth or other material to reduce future subsidence within the fill.
- Waste must be placed and covered in a manner that limits the radiation dose rate at the surface of the cover to levels that at a minimum will allow the licensee to comply with all standards against radiation protection at the time the facility is closed and stabilized.
- The boundaries and locations of disposal units must be accurately located and mapped by means of a land survey.
- Near-surface disposal units must be marked in such a way that the boundaries of the units can be easily defined. Three permanent survey marker control points, referenced to USGS or National Geodetic Survey control stations, must be established on the site to facilitate surveys.
- Horizontal and vertical controls must be provided by USGS or National Geodetic Survey control stations as checked against USGS or National Geodetic Survey record files.
- A buffer zone of land must be maintained between any buried waste and the disposal site boundary and beneath the disposed waste. The buffer zone must be of adequate dimensions to carry out environmental monitoring activities and take mitigative measures if needed.

- Closure and stabilization measures as set forth in the approved site closure plan must be carried out as the disposal units are filled and covered.
- Active waste disposal operations must not have an adverse effect on completed closure and stabilization measures.
- Only wastes containing or contaminated with radioactive material may be disposed of at the disposal site.

#### **16.6.8.5 Environmental Monitoring Requirements (10 CFR 61.53)**

- When a license application is first submitted, the applicant must have conducted a preoperational monitoring program to provide basic environmental data on the disposal site characteristics. The applicant must obtain information about the ecology, meteorology, climate, hydrology, geology, geochemistry, and seismology of the disposal site. For those characteristics that are subject to seasonal variation, data must cover at least a 12-month period.
- During the land disposal facility site construction and operation, the licensee must maintain an environmental monitoring program. Measurements and observations must be made and recorded to provide data to
  - Evaluate the potential health and environmental impacts during both the construction and the operation of the facility
  - Enable the evaluation of long-term effects and need for mitigative measures
  - Provide early warning of releases of waste from the disposal site before they leave the site boundary
- After the disposal site is closed, the licensee responsible for postoperational surveillance of the disposal site must maintain a monitoring system based on the operating history and the closure and stabilization of the disposal site. The postoperational monitoring system must also be capable of providing early warning of releases of waste from the disposal site before they leave the site boundary.
- The licensee must have plans for taking corrective measures if the environmental monitoring program detects migration of waste which would indicate that the performance objectives may not be met.

#### **16.6.8.6 Financial Assurance Requirements (10 CFR 61, Subpart E)**

Financial assurances must be provided to ensure that sufficient funds will be available to carry out disposal site closure and stabilization. The determination of the amount of financial assurances required must consider

- Decontamination or dismantlement of land disposal facility structures
- Closure and stabilization of the disposal site so that the need for ongoing active maintenance is eliminated to the extent practicable, and so that only minor custodial care, surveillance, and monitoring are required

Assurances required must be based on cost estimates that are approved by the regulatory agency and that reflect the plan for disposal site closure and stabilization approved by the

regulatory agency. These cost estimates must also take into account total capital costs that would be incurred if an independent contractor were hired to perform the closure and stabilization work.

The amount of financial assurance provided must account for changes in factors that affect the cost of future closure and stabilization. Factors affecting closure and stabilization cost estimates include

- Inflation
- Increases in the amount of disturbed land
- Changes in engineering plans
- Closure and stabilization that has already been accomplished
- Any other conditions affecting costs

Changes to estimated costs for closure and stabilization are typically reviewed, and adjustments to financial assurance provided are made annually to ensure that sufficient sureties exist at all times to cover the costs of closure of the disposal units.

#### **16.6.8.7 Other Requirements**

Each licensee must also implement and maintain Quality Assurance and Quality Control programs to provide documentary evidence that required activities are performed properly. Licensees maintain records of all activities that indicate and document the performance of their respective disposal facilities.

In addition to these universally applicable requirements, the regulatory agencies are authorized and empowered to impose license conditions that must also be met. The regulatory agencies maintain surveillance, monitor all activities related to the licensed facilities, and periodically perform inspections to ensure compliance with regulatory requirements and license conditions.

Facility licensees periodically prepare and submit environmental monitoring, operating, and other reports to the regulatory agency. The regulatory agency reviews and evaluates all such reports to assess whether the facility is being operated as required and as planned and to determine whether changes should be made to provide greater assurance that the facility will perform as required.

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# 17

## *Radioactive Materials Transportation*

**Kurt Colborn**

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## 17.1 Introduction

Planning and executing the safe shipment of radioactive materials is a complex process. The shipper must first have a thorough understanding of the material being shipped and must consider not only the isotopic makeup of the material but also the process that generated the material and the reason for the shipment. Physically identical materials may be regulated differently based solely on their source. The shipper's knowledge of the material will also figure heavily in the selection of a package for the material, and the shipper may find that packaging requirements for transportation do not allow shipment in a package that was deemed perfectly acceptable for storage. Further, packaging requirements sometimes vary based on the mode of transportation selected, be it truck, rail, or vessel.

The shipper may also face challenges to transportation not directly related to the act of shipping the material. The characterization data available to the shipper may have been assembled for purposes other than transportation, and may not satisfy the information requirements for safe shipment of the material (e.g., characterization of waste from a remediation project designed to satisfy site cleanup criteria, may not support proper classification for shipment). In addition, the shipper of radioactive waste is generally responsible for ensuring that the material and its packaging meet the waste acceptance criteria of the receiving facility. These can vary widely from facility to facility and often reflect issues of importance to disposal that may not have been considered in the waste generation and packaging process. Nonwaste shippers may also face limitations at the receiving facility, such as site-specific dose restrictions in the loading or storage area, or limits on the quantity that can be received.

The reader is cautioned that the following chapter must not be considered a "how to" guide for radioactive materials shipment. Each shipment of radioactive material must be carefully planned and evaluated to ensure the myriad and sometimes conflicting requirements of the shipper, receiver, transporter, and regulator are all addressed. In addition, specific training requirements apply to shippers, such as those prescribed in the United States Department of Transportation (US DOT) Regulations (49 CFR 172 Subpart H). Instead, this chapter provides an overview of the landscape of radioactive materials transportation requirements and is intended to introduce the basic concepts of transportation regulation, including the identification, classification, packaging, documentation, and other requirements that must be considered when planning a radioactive materials shipment. Inexperienced shippers may utilize the material in this chapter as a starting point in their shipment planning and perhaps use it as a basis for review of the basic elements of their shipment plan to ensure that no matter of critical importance is omitted in their preparations. Experienced shippers may find the chapter useful as a refresher, and for the

examples of shipment challenges and lessons learned that are woven into the text. At a minimum, the reader should take away one lesson from this chapter: that no radioactive materials shipment is “routine”; and each must be approached with due care and diligence.

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## 17.2 Regulating Authorities

### 17.2.1 International Atomic Energy Agency (IAEA)

The first set of international standards for the shipment of radioactive materials was assembled in the IAEA’s 1961 *Regulation for the Safe Transport of Radioactive Materials*. It was based largely on the experience in the United States. US DOT regulations largely mirror those of the IAEA, as published in the IAEA Safety Standards, Specific Safety Requirements, No. SSR-6. The IAEA standards for limiting package contents and external radiation doses have been adopted by other agencies as well, including the International Civil Aviation Organization (ICAO) who promulgates controls on radioactive materials shipments by air through the International Air Transport Association (IATA) regulations. Similarly, IAEA standards for control of radioactive material shipments have been adopted by the International Maritime Organization (IMO) and documented in the International Maritime Dangerous Goods (IMDG) Code. The IAEA also publishes a guidance document, *Advisory Materials for the IAEA Regulations for the Safe Transport of Radioactive Material, TS-G-1.1 (Rev. 1)*. The SSR-6 and the TS-G-1.1 are sometimes helpful to consider to help understand the intent or to clarify the language of US DOT regulations. This chapter addresses shipping requirements through reference to US DOT regulations. There are some “domestic shipment only” provisions in the US DOT regulations but, in general, the similarity of US DOT regulations to IAEA requirements supports fairly straightforward cross-referencing when needed. The IAEA and DOT both publish schedules of provisions of the regulations that map the regulations for different shipping situations (Boyle et al., 1999; IAEA, 2009).

### 17.2.2 US DOT

Radioactive Materials Transportation in the United States is regulated by the US DOT. The regulations for shipment are promulgated in the Hazardous Materials Regulations (HMR) in the Code of Federal Regulations (CFR), Parts 100–180, with shipper requirements for radioactive materials concentrated in (but not limited to) Parts 172 and 173. The US DOT regulations can appear complex, particularly because of their organization, but can also provide the careful shipper with a clear roadmap on how to compliantly execute a radioactive materials shipment. The regulations include requirements for characterization information; shipping documentation; marking, labeling, and placarding; packaging design, construction and selection; and dose and contamination control. The regulations also set out reporting requirements for incidents, as well as supporting requirements such as those for written quality assurance and training and security plans for packaging manufacturers, shippers, and carriers.

Regulatory oversight is provided within the US DOT by the Pipeline Hazardous Materials Administration (PHMSA). The PHMSA maintains a website, <http://www.phmsa.dot.gov>, with detailed information that can be extremely helpful to the novice and experienced shipper alike. The site features news of recently proposed and promulgated regulations,

links to the regulations themselves, interpretive guidance the DOT has provided to other shippers, and a library of special permits that have been issued to address unique situations (Willaford, 2006).

The US DOT is responsible for shipment regulation in general, and for packaging standards for low-risk materials. Packaging standards for high-risk materials are established by the U.S. Nuclear Regulatory Commission (US NRC).

### 17.2.3 US NRC

High-risk materials subject to US NRC packaging regulations include fissile materials, Type B quantities, and other radioactive materials (including low specific activity material and surface contaminated objects) (LSA and SCOs), having a dose rate  $>1$  roentgen equivalent man per hour at a distance of one meter from the unshielded material. The US NRC also has jurisdiction over other matters of importance to the shipper that are not directly related to transportation, such as the classification of radioactive waste in 10 CFR 61, as well as certification of Type B packages and registration of Type B package users, as reflected in 10 CFR 71.17. DOT regulations will often point to NRC transportation regulations when they apply, for example, the parallel registration requirement in 49 CFR 173.471.

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## 17.3 Identification and Classification

A casual conversation with a layman about radioactive materials transportation is likely to focus on packaging durable enough to protect the integrity of the shipment, and on transportation practices that ensure public safety and security. These are, of course, important considerations in a radioactive material shipment. However, these are not the first priorities when planning a compliant shipment.

Identification of the materials being shipped is the first, and arguably the most important, task in planning a radioactive materials shipment. Identification is a first priority because the planning process simply cannot begin until the detailed nature of the material being shipped is determined and documented. Identification is the most important task because it impacts the subsequent tasks of package selection and shipment documentation. Transportation regulations place a heavy emphasis on communicating the hazard of a radioactive materials shipment through proper identification, classification, and shipping documentation, not only as the basis for a compliant shipment but also to establish a reliable source of information for emergency responders in the event of a transportation incident.

### 17.3.1 Identification

The definition of a hazardous material is found in 49 CFR 171.8. The definition broadly includes hazardous substances, hazardous wastes, marine pollutants, and other materials designated as hazardous in the Hazardous materials table of 49 CFR 172.101, or meeting the defining criteria for hazard classes and divisions in 49 CFR 173. Hazardous substance is also defined in 49 CFR 171.8 to include any material listed in Appendix A to 49 CFR 172.101 that is present in a reportable quantity (RQ).

The shipper will have to read these definitions and referenced sections and evaluate the characterization information for the material being shipped to determine if it is a

hazardous substance. The determination will need to consider the reportable quantities for radionuclides found in 49 CFR 172.101, Appendix A, Table 2, and the mixture rule in Appendix A, paragraph 7.

It is possible that radionuclides may be identified and shipped as environmentally hazardous substances by this process, *even if the radionuclides are not present in sufficient quantity to be defined as radioactive material* in accordance with the definition of 49 CFR 173.403. It is a serious error to assume that radionuclides not meeting the definition of radioactive material are exempt from the hazardous materials regulations.

### 17.3.2 Radioactive Material Classification

Radioactive materials that meet the definition of a hazardous substance because of their radionuclide content must then be compared to the definition of radioactive material in 49 CFR 173.403. The definition requires that the following two criteria must both be exceeded for a material to be considered radioactive:

1. Activity concentration for exempt material
2. Activity limit for exempt consignment

This is determined by comparison with a table of values for each isotope in 49 CFR 173.436, or by comparison with values derived in accordance with the formulas in 49 CFR 173.433. Material is exempt from radioactive material designation (Class 7) if it fails either of the two criteria.

The activity concentration for exempt material is beneficial to shippers of high-volume, low-concentration material. The activity limit for consignments will be most beneficial to a low-volume shipment. The concentration limit will almost always drive the classification if the consignment is >100 kg.

Classification of the material is more complex if hazards other than radioactivity are present. While Class 7 radioactive material classification takes precedence over other hazards (i.e., the regulations consider “radioactive” to be the highest hazard class), the presence of secondary chemical hazards must still be considered in radioactive shipment preparation, documentation, and execution. Refer to 49 CFR 173.2 for information on other hazard classes.

SCOs are not regulated for shipment if contamination is below the definition of contamination in 49 CFR 173.403. SCOs are also exempt from classification as radioactive material if the total activity on the surface can be demonstrated to be less than the activity limit for an exempt consignment (see 49 CFR 173.401(b)(5) as of 2014, and DOT interpretations 05-0145 dated July 1, 2005, and 08-0012 dated May 7, 2008; available on the PHMSA website, <http://www.phmsa.dot.gov/hazmat/regs/interps>).

### 17.3.3 Radioactive Material Exemptions

One pitfall to avoid in preparing for a radioactive materials shipment is erroneously focusing on the detailed requirements of the shipping regulations without careful consideration of their applicability. For example, the definition of radioactive material is stated clearly in 49 CFR 173.403, and guidance for calculating activity for each isotope is plainly stated in 49 CFR 173.433 and 173.436. However, the shipper must also consider the scope of 49 CFR 173 Subpart I, found in 49 CFR 173.401.

The scope of Subpart I establishes the applicability of subsequent regulations in the subpart. Specifically, the scope of the regulations does not include:

1. Materials not in transportation (e.g., in storage, production, or use at a stationary facility)
2. Materials implanted into a person or live animal for diagnosis or treatment
3. Radioactive material that is an integral part of the means of transport (e.g., depleted uranium counterweights while serving their function on the aircraft)
4. Natural material and ores containing naturally occurring radionuclides not intended to be processed for the use of these radionuclides, provided the activity concentration does not exceed 10 times the values specified in 49 CFR 173.436

In the preamble to the regulation found in the Federal Register, the US DOT stated that the natural materials exemption was “intended to exempt ores and materials that contain naturally occurring radionuclides, but whose benefits lie in their nonradiological qualities (e.g., nonradioactive metals).”

When this regulation was first issued, shippers were uncertain if it could be applied to processed materials (e.g., tungsten or zirconium ores that had been processed for extraction of these nonfuel cycle materials). The regulation did not specifically mention waste products and did not clarify if processing affected the materials’ status as “natural.”

Resolution of this matter came in the form of an interpretation from the US DOT in July 2005 in an interpretation letter number 05-0145. The interpretation request and result provide a good example of how examination of the applicability of the regulations can be beneficial to the shipper, and how IAEA regulation and guidance can provide helpful clarification.

The exemption does not specifically mention waste products, but it seems reasonable to expect that the waste materials from beneficial extraction of nonradioactive metals would qualify for the exemption because the intent of the regulation (per the preamble in the *Federal Register*) is to permit their *continued use in commerce without making their use economically unfeasible*. Additional support of the applicability of the 10× concentration exemption was found in the IAEA Advisory Guide paragraph 107.4. This guidance explained that the IAEA regulations reflected in 49 CFR *do not apply to other (nonfuel cycle) ores that may contain naturally occurring radionuclides or processed materials... where the processing was not for the purpose of extracting radionuclides*.

The US DOT concurred with the shipper’s interpretation of the regulation for processed material, plans to ship the material as Class 7 were abandoned, and the approach to shipping several hundred tons of processed ore was greatly simplified.

A revision to 173.401(4) takes effect in 2014 that clarifies the requirement in this discussion:

Natural material and ores containing naturally occurring radionuclides, which are either in their natural state, or which have only been processed for purposes other than for extraction of the radionuclides, and which are not intended to be processed for the use of these radionuclides, provided the activity concentration of the material does not exceed 10 times the exempt material activity concentration values specified in Section 173.436, or determined in accordance with the requirements of Section 173.433.

The 2014 revision also adds a paragraph:

Non-radioactive solid objects with radioactive substances present on any surfaces in quantities not exceeding the threshold limits set forth in the definition of contamination in Section 173.403.



#### 17.3.4 Activity Determination in Preparation for Packaging Selection

The activity in a shipment must be determined to properly classify the material being shipped, as discussed above. In addition, the activity must be determined and compared with values the regulations define as a basis for packaging selection, specifically the  $A_1$  and  $A_2$  values for the material. The  $A_1$  and  $A_2$  values represent the maximum allowable activity that can be shipped in a Type A package. The regulatory basis is that the  $A_1$  and  $A_2$  values are the activity levels that can be shipped in standard Type A packaging, under normal conditions of transport, without undue risk to public health and safety. If the activity levels exceed  $A_1$  and  $A_2$  values, then the packaging must be more durable to address the greater risk of exposure to the public; this is addressed through the requirement to use a Type B package, designed to provide containment of its contents in accident conditions. Accordingly, quantities of materials exceeding the  $A_1$  and  $A_2$  values are also referred to as Type B quantities.

Definitions for  $A_1$  and  $A_2$  are found in 49 CFR 173.403. The  $A_2$  value is the most commonly used, as it refers to the maximum activity of Class 7 radioactive material (other than special form material) that can be shipped in a Type A package. The  $A_1$  value is the maximum activity of *special form* Class 7 material that can be shipped in a Type A package.

*Special form* is also defined in 49 CFR 173.403, as an indispersible solid radioactive material or a sealed capsule containing radioactive material satisfying several conditions for composition and testing prescribed in the definition.

The shipper must assess the radioactivity of the materials being shipped and in comparison with their  $A_1$  and  $A_2$  values in accordance with the regulations as one of the bases for packaging selection. In general, Type A (less than the applicable  $A_1$  and  $A_2$  values) quantities can be shipped in Type A packages; while Type B quantities (greater than  $A_1$  and  $A_2$  values) will require a Type B package. There are exceptions for the special classes of LSA material and SCOs that will be discussed later in this chapter.

The  $A_1$  and  $A_2$  values for individual isotopes are found in tabular format in 49 CFR 173.435. Formulae for determining the  $A_1$  and  $A_2$  values for mixtures of radionuclides are found in 49 CFR 173.433.

When using the tables anywhere in the regulation, take special care to consider information in the footnotes. For example, the footnotes in 49 CFR 173.435 include a clarification that the  $A_1/A_2$  values for some isotopes considers the activity of daughter products with half-lives less than 10 days. Longer lived daughters must be considered as radionuclides in a mixture, and the  $A_1/A_2$  value must be calculated in accordance with 49 CFR 173.433. Note that a similar, but not identical, provision exists in the footnotes for the exemption values in 49 CFR 173.436.

---

### 17.4 Types of Material and Packaging Selection

Armed with a description of the material being shipped, and knowledge of its activity and isotopic distribution to allow a determination that it is radioactive material, and its activity relative to  $A_1$  and  $A_2$  values, the shipper has the information needed to proceed to packaging selection based on the material type.

There are six main types of radioactive material, and the distinction between types correlates to the distinction between allowable packagings. These are

1. Limited quantity material
2. Type A material
3. Type B material
4. Fissile material
5. LSA material
6. SCOs

#### 17.4.1 Limited Quantity Material

Limited quantities of materials are defined in 49 CFR 173.403 and discussed in 49 CFR 173.425. The table in that section provides detailed guidance on what defines a limited quantity for solids in normal and special form, for tritiated liquids, and for gases. For example, a limited quantity normal form solid is a quantity of  $<10^{-3} A_2$  in total activity per package. The table also contains provisions for shipping instruments or articles containing small amounts of radioactivity as a limited quantity.

Limited quantity materials are sometimes referred to as “excepted quantity” because they are eligible for shipment in excepted packages as set forth in 49 CFR 173.421. Excepted packages are nonspecification (or general design) packages, meeting the general design requirements of 49 CFR 173.410.

At this point, it is appropriate to discuss the definitions of the terms “package” and “packaging.” These are defined in 49 CFR 173.403. The somewhat awkward term “packaging” is essentially the empty receptacle (and any shielding or related packing materials or devices) into which the radioactive material is placed. The term “package” means the packaging together with its radioactive contents. Moreover, “packaging” should not be construed to be the outermost layer or visible layer of a packaging with several elements. For example, the structural and containment requirements for a packaging might be independently met by different elements or layers of the packaging.

The definitions of the terms become particularly important when considering the dose requirements for various packages. For example, to qualify as limited quantity, the dose rate on the outside of the loaded package cannot exceed 0.005 mSv/h (see 49 CFR 173.421(a)(2)). Similarly, contamination limits set forth in 49 CFR 173.443(a) must not be exceeded in a limited quantity shipment. The excepted package requirements of 49 CFR 173.421 include additional requirements for marking and limitations on fissile material content.

Empty packagings that have previously contained radioactive materials are not strictly limited quantity shipments, but empty packaging shipments are regulated very much like limited quantity excepted package shipments. Empty packagings are excluded from most radioactive material shipping paper and marking requirements. Empty shipments must bear the UN identification number appropriate for an empty packaging (UN 2908) and must meet other requirements as set forth in 49 CFR 173.428.

#### 17.4.2 Type A Material

Recall from the definition of  $A_1$  and  $A_2$  values that a Type A quantity of radioactive material does not exceed  $A_1$  for special form radioactive materials or  $A_2$  for normal form radioactive materials. Special form radioactive materials must be determined to meet the definition in accordance with the testing requirements of 49 CFR 173.469.

A DOT Type A shielded cask is shown in Figure 17.1. The packaging requirements for solid Type A materials are listed below:

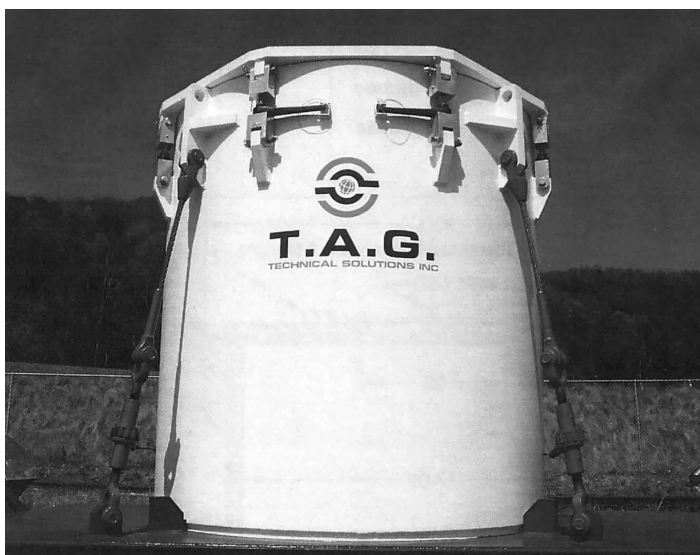
- Meet the general design requirements of 49 CFR 173.410 (including applicable referenced sections of 49 CFR 173.24, 173.24a, and 173.24b)
- Testing in accordance with 49 CFR 173.465, including water spray, free drop, stacking, and penetration tests
- Additional features and requirements in 49 CFR 173.412, including provisions for a tamper-indicting device or seal, a minimum outside dimension of 100 mm, and performance requirements related to temperature, closure, and securement
- Note that additional testing requirements apply for Type A packagings for liquids, as specified in 49 CFR 173.466.

### 17.4.3 Type B Material

Type B material is a radioactive material exceeding  $A_1$  for special form radioactive materials or  $A_2$  for normal form radioactive materials. Recall from the introductory section that Type B quantities are regulated by the NRC. Type B packagings are therefore certified by the NRC in accordance with the testing requirements of 10 CFR part 71 for their ability to withstand accident conditions in transportation. An example of a Type B package is the Waste Control Specialists' RT-100 cask, shown in Figure 17.2.

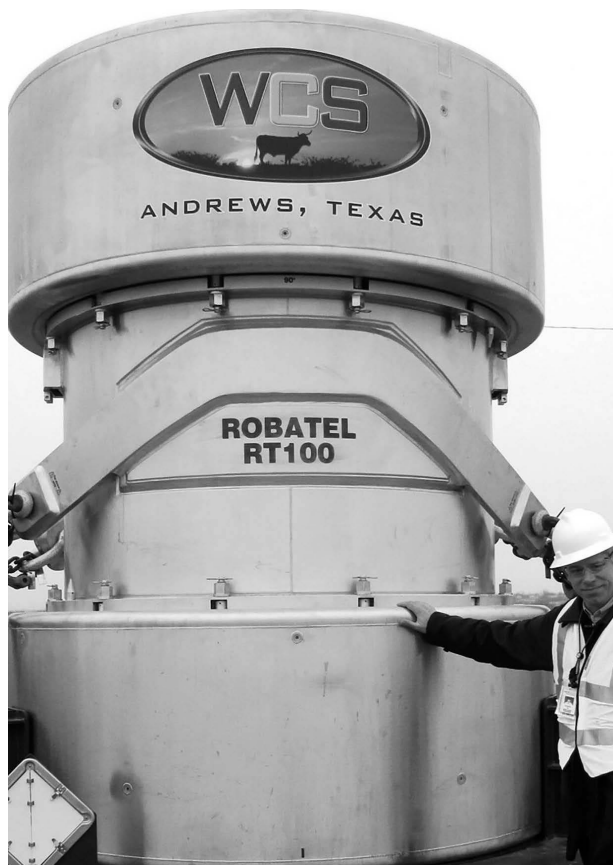
There is no upper limit on Type B activity as far as the material definition is concerned. However, Type B packagings are limited by their certification in the isotopes and quantities of material they can carry.

Suppliers of Type B packagings also provide detailed use instructions, or in some cases, personnel to oversee packaging use and closure. The US DOT references on the selection and use of Type B packagings are in 49 CFR 173.413 and 173.416. Type B



**FIGURE 17.1**

TAG 14-212 Type A shielded cask. (Courtesy of TAG Technical Solutions, LLC, <http://www.thetaggroup.co/>.)



**FIGURE 17.2**  
WCS RT-100 Type B cask. (Courtesy of Waste Control Specialists, LLC, <http://www.wcstexas.com/facilities/cask-services/>.)

package users in the United States must register with the NRC, a requirement also addressed by the DOT in 49 CFR 173.471.

The Department of Energy (DOE) maintains a website listing a wide variety of shipping containers and their ratings, including links to Type B packaging certifications. The website is <http://www.rampac.com>. The site includes a variety of other information of interest to shippers, including DOT special permits, available shipper training courses, and recent regulatory developments and interpretations.

#### 17.4.4 Fissile Material

Fissile material is defined in 49 CFR 173.403 as plutonium<sup>239</sup>, plutonium<sup>241</sup>, uranium<sup>233</sup>, uranium<sup>235</sup>, or any combination of these radionuclides. The term does not apply to material containing fissile nuclides, nonirradiated natural uranium, and nonirradiated depleted uranium, or to natural uranium or depleted uranium that has been irradiated in thermal reactors only.

Exceptions to fissile material designation exist in 49 CFR 173.453. These include exceptions for a total quantity of <2 g, exception of a total quantity of 15 g in a package that

includes at least 200 g of nonfissile material for every gram of fissile material, and an exception based on concentration of fissile material to nonfissile material not exceeding 1/2000.

Transportation requirements for fissile material are discussed in 49 CFR 173.457, and packaging options are discussed in 49 CFR 173.417. There are some options to ship fissile material in a Type A packaging, subject to limits on quantity and enrichment. For guidance, refer to 49 CFR 173.417(a)(1)(i) and 10 CFR 71.22, Table 71.1. Note that the drop test requirement of 49 CFR 173.465(c)(2) must be met in addition to standard Type A packaging requirements to use a Type A packaging for fissile material. Type B packagings may also be used in accordance with the packaging certificates (refer to RAMPAC to identify packagings and capabilities).

#### 17.4.5 Low Specific Activity Material

Much of the regulation of radioactive material shipment hinges on the  $A_1/A_2$  value, and the classification and packaging of the material may fall in line after this determination is made. However, a large portion of radioactive material shipments make use of provisions for LSA material that falls outside of the  $A_1/A_2$ , Type A/Type B regulatory structure.

LSA material is defined in 49 CFR 173.403 (2014) and falls into three groups.

##### LSA-I:

- (i) Uranium and thorium ores, concentrates of uranium and thorium ores, and other ores containing naturally occurring radionuclides that are intended to be processed for the use of these radionuclides; or
- (ii) Natural uranium, depleted uranium, natural thorium or their compounds or mixtures, provided they are unirradiated and in solid or liquid form; or
- (iii) Radioactive material for which the  $A_2$  value is unlimited; or
- (iv) Other radioactive material in which the activity is distributed throughout and the estimated average specific activity does not exceed 30 times the values for activity concentration specified in 173.436 or calculated in accordance with 173.433, or 30 times the default values listed in Table 8 of 173.433.

Recall from the discussion of radioactive material in this chapter that uranium and thorium ores, and other naturally radioactive ores that are *not* intended for processing to remove the radionuclides are exempt from the HMR. This is an example of how physically identical material, equal in isotopes and activity, can be regulated differently for shipment based on its administrative status, previous use, or intended use. When planning for compliant shipping, the history and planned use of the material can be as important as the isotopic analysis.

Also, note that the definition for LSA-I is isotope specific; radioactive material is LSA-I at activities up to 30 times the isotope values in 49 CFR 173.436. LSA status is not established at a fixed activity level, but at a level that varies by radionuclide.

##### LSA-II:

- (i) Water with tritium concentration up to 0.8 TBq/L (20.0 Ci/L); or
- (ii) Other radioactive material in which the activity is distributed throughout, and the average specific activity does not exceed  $10^{-4} A_2/g$  for solids and gases, and  $10^{-5} A_2/g$  for liquids

## LSA-III:

LSA-III is defined to include solids (e.g., consolidated wastes, activated materials but excluding powders) that meet the requirements of 49 CFR 173.468 and in which

- (i) The radioactive material is distributed throughout a solid or a collection of solid objects, or is essentially uniformly distributed in a solid compact binding agent (e.g., concrete, bitumen, ceramic)
- (ii) The radioactive material is relatively insoluble, or it is intrinsically contained in a relatively insoluble material, so that, even under loss of packaging, the loss of Class 7 (radioactive) material per package by leaching when placed in water for seven days would not exceed  $0.1 A_2$  and
- (iii) The estimated average specific activity of the solid, excluding any shielding material, does not exceed  $2 \times 10^{-3} A_2/g$

A key word in this definition is the “and” after the second bullet; all three conditions must be met to qualify as LSA-III.

Shipping materials that can be classified as LSA are often less restrictive than shipping the materials as Type A or Type B. In particular, shipping materials as LSA allows for the use of a wider variety of packagings that are often more versatile and less expensive than Type A or Type B packagings. These packaging options will be discussed after the discussion of SCOs because the packaging options for LSA and SCO are similar.

#### 17.4.6 SCOs

An SCO is a solid object that is not radioactive, but which has radioactive material distributed on its surface. An SCO exists in two phases, fixed and nonfixed, and in two different groups:

## SCO-I:

A solid object on which

- The nonfixed contamination on the accessible surface averaged  $>300 \text{ cm}^2$  (or the area of the surface if  $<300 \text{ cm}^2$ ) does not exceed  $4 \text{ Bq/cm}^2$  ( $10^{-4} \text{ mCi/cm}^2$ ) for beta and gamma and low toxicity alpha emitters, or  $0.4 \text{ Bq/cm}^2$  ( $10^{-5} \text{ mCi/cm}^2$ ) for all other alpha emitters.
- The fixed contamination on the accessible surface averaged  $>300 \text{ cm}^2$  (or the area of the surface if  $<300 \text{ cm}^2$ ) does not exceed  $4 \times 10^4 \text{ Bq/cm}^2$  ( $1.0 \text{ mCi/cm}^2$ ) for beta and gamma and low-toxicity alpha emitters, or  $4 \times 10^3 \text{ Bq/cm}^2$  ( $0.1 \text{ mCi/cm}^2$ ) for all other alpha emitters.
- The nonfixed contamination plus the fixed contamination on the inaccessible surface averaged  $>300 \text{ cm}^2$  (or the area of the surface if  $<300 \text{ cm}^2$ ) does not exceed  $4 \times 10^4 \text{ Bq/cm}^2$  ( $1 \text{ mCi/cm}^2$ ) for beta and gamma and low-toxicity alpha emitters, or  $4 \times 10^3 \text{ Bq/cm}^2$  ( $0.1 \text{ mCi/cm}^2$ ) for all other alpha emitters.

At this time, the contamination level above which an item should be designated as an SCO is found in the contamination definition of 49 CFR 173.403;  $0.4 \text{ Bq/cm}^2$  for beta and gamma emitters and low-toxicity alpha emitters or  $0.04 \text{ Bq/cm}^2$  for all other alpha emitters. This definition of contamination (or the lower limit on SCO) can be relied upon for international shipments to and from countries where this IAEA-based

standard is used both as a free release (decontamination) criterion and as a shipping criterion.

However, many entities in the United States routinely free release objects with surface contamination above the IAEA's minimum level. The effect is that the US DOT regulations may require radioactive material markings in transportation on items that have been free released in accordance with the terms of a governing radioactive materials license or DOE policy. The DOT clarified its position in the 2014 revision of 49 CFR 173.443, and in discussion in the Federal Register notice of changes to the HMR; 79 FR 40589–40618.

#### SCO-II:

A solid object on which the limits for SCO-I are exceeded and on which

- The nonfixed contamination on the accessible surface averaged  $>300 \text{ cm}^2$  (or the area of the surface if  $<300 \text{ cm}^2$ ) does not exceed  $400 \text{ Bq/cm}^2$  ( $10^{-2} \text{ mCi/cm}^2$ ) for beta and gamma and low-toxicity alpha emitters, or  $40 \text{ Bq/cm}^2$  ( $10^{-3} \text{ mCi/cm}^2$ ) for all other alpha emitters.
- The fixed contamination on the accessible surface averaged  $>300 \text{ cm}^2$  (or the area of the surface if  $<300 \text{ cm}^2$ ) does not exceed  $8 \times 10^5 \text{ Bq/cm}^2$  ( $20 \text{ mCi/cm}^2$ ) for beta and gamma and low-toxicity alpha emitters, or  $8 \times 10^4 \text{ Bq/cm}^2$  ( $2 \text{ mCi/cm}^2$ ) for all other alpha emitters.
- The nonfixed contamination plus the fixed contamination on the inaccessible surface averaged  $>300 \text{ cm}^2$  (or the area of the surface if  $<300 \text{ cm}^2$ ) does not exceed  $8 \times 10^5 \text{ Bq/cm}^2$  ( $20 \text{ mCi/cm}^2$ ) for beta and gamma and low-toxicity alpha emitters, or  $8 \times 10^4 \text{ Bq/cm}^2$  ( $2 \text{ mCi/cm}^2$ ) for all other alpha emitters.

Low-toxicity alpha emitters are natural uranium; depleted uranium; natural thorium; uranium<sup>235</sup> or uranium<sup>238</sup>; thorium<sup>232</sup>; thorium<sup>228</sup>, and thorium<sup>230</sup> when contained in ores or physical and chemical concentrates; and alpha emitters with a half-life of  $<10$  days.

#### 17.4.7 Packaging and Shipping LSA and SCO

There are several benefits to shipping as LSA or SCO, but selection of a compliant shipping method requires careful consideration of the material being shipped, and thorough considerations of the transport requirements for the material. Techniques for shipping as LSA and SCO will not be exhaustively discussed in this chapter, and the reader is encouraged to carefully review the requirements for shipping in the regulations, 49 CFR 173.427.

Note that 49 CFR 173.427 offers a defining criterion for LSA and SCO in addition to those in the definition of LSA and SCO in 173.403; the external dose rate from the unshielded material cannot exceed  $10 \text{ mSv/h}$  ( $1 \text{ rem/h}$ ) at 3 m. This limit on the dose rate of the contents may drive the shipper to use a Type B package even when the contents meet the definition of LSA in 173.403, and when the external package dose requirements for shipping (49 CFR 173.441) could be met using a Type A package.

The benefits of shipping as LSA or SCO start to become apparent with examination of the conveyance limits found in Table 5 of 49 CFR 173.427. By definition in 49 CFR 173.403, a highway or rail conveyance is *any transport vehicle or large freight container*. It's a common misconception that LSA and SCO can be packaged and shipped only in quantities up to the  $A_2$  value. A quick glance at Table 5 reveals that this is not the case. There is *no activity limit* for LSA noncombustible solids; and LSA-II and -III combustible solids and both classes of SCO can be shipped at up to 100 times the  $A_2$  value. Without the LSA provision, these

materials would have to be shipped in a Type B package, complete with NRC certification, and the shipment would be limited by extra bulk and limited capacity typical of Type B packagings. The LSA/SCO provision allows for the shipment of large volumes of materials of limited activity concentration with generous upper limits on the quantities that can be shipped in a single conveyance or large package.

The origin of the confusion about activity limits in LSA/SCO packages is in 49 CFR 173.427(b)(4). This paragraph allows for the transportation of LSA/SCO (up to and including LSA-III and SCO-II) in a general design package, provided that the total activity of the contents is limited to a single  $A_2$  value. The  $A_2$  value limit does not apply to other packaging types. This regulation is limited to exclusive use domestic shipments in the United States, and there is no equivalent provision in IAEA regulations. The benefit to the shipper is that a general design package (or an IP-1 package meeting the general design requirements in accordance with 49 CFR 173.411(b)) can be used. These packages require no specific testing or certification by an external authority. The regulation does not specifically mention 49 CFR 173.24b bulk package requirements for use in LSA/SCO shipments, but interpretations from the US DOT have clarified that bulk packages can qualify as general design or IP-1 packages (e.g., US DOT interpretation reference no. 05-0138 dated July 5, 2005). Assurance that the packaging meets these criteria is the responsibility of the shipper, often based on representations by the packaging manufacturer or supplier.

LSA-I and SCO-I materials are eligible for shipment "unpacked" in accordance with 49 CFR 173.427(c). They are not subject to any restriction on activity, so large volumes of material (e.g., gondola railcars) can be shipped without being limited by the  $A_2$  value. "Unpacked" is not a defined term, but the paragraph does require that unpackaged material *...shall be transported in such a manner that under normal conditions of transport there will be no escape of the radioactive contents from the conveyance...*

The use of the LSA/SCO provisions offer significant advantages over Type A or Type B shipping for eligible materials. The LSA/SCO provisions include many opportunities to make use of safe and cost-effective packagings whose physical capacities exceed most common Type A and Type B packagings. These opportunities exist in the text of the regulation (as discussed above) or in 49 CFR 173.427 Table 6, which sets out the requirements of use of industrial packagings (IPs).

The NRC and DOT have published helpful joint guidance on Categorizing and Transporting LSA and SCO in NUREG-1608/RAMPAC-003.

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## 17.5 IP and Additional Packaging Guidance

### 17.5.1 Type A and Type B General Guidance

Approved packaging for Type A and Type B is discussed at some length in the previous section of this chapter. Briefly, unless the shipper is prepared to engage in an independent design, test, and certification program, a packaging supplier should be contacted to provide the required packaging. Once the shipper identifies the defining characteristics of the shipment (volume, activity, material type), that information can be provided to a vendor for delivery of an appropriate and certified packaging. A search of the Internet in general, of the RAMPAC site in particular, can help guide the shipper to the necessary suppliers and information.



### 17.5.2 IP-1

IPs, specifically IP-1 packagings, can be self-certified by the shipper. No specific testing requirements apply to IP-1 packagings; test documentation is required only for IP-2 and IP-3 packagings. Packagings can be purchased with an IP-1 rating, but shippers are also able to self-certify commercially available packagings meeting the IP-1 requirements of 49 CFR 173.411(b). Vendors can also supply packagings complete with certificates of compliance to applicable standards, and with required marking already placed on the container.

The IP-1 packaging (formerly referred to as “strong tight”) must conform to the requirements of 49 CFR 173.411(b), including the referenced standard design requirements of 49 CFR 173.410, 173.24, 173.24a, and 173.24b. These include performance-based requirements (not test-based requirements) for

- Ease of handling
- Strength of lifting attachments or features (must be designed for three times the container capacity)
- Ability to be cleaned/decontaminated
- Containment of radioactive materials during normal conditions of transport

IP-1 Intermodal Bulk Packaging is shown in Figure 17.3.

Non-rigid packagings for radioactive materials have benefitted from increasing acceptance in recent years. Sometimes referred to as “bulk bags” or “super sacks,” these packages can be demonstrated to meet IP-1 or IP-2 requirements. Generally, these are constructed of one or two layers of a heavy woven plastic, and sometimes feature an internal bladder as an additional barrier to the release of contamination. They are top-lifted by several straps that typically form a basket sling around the bag. These are typically single-use packagings but benefit from a much lower cost to purchase and to mobilize to a remote site than large rigid packagings. IP-1 flexible packaging is shown in Figure 17.4.



**FIGURE 17.3**

ICE Services Group IP-1 intermodal container. (Courtesy of ICE Service Group, Inc., [http://iceservicegroup.com/dt\\_portfolio/intermodal-containers-ip-1-strong-tight/](http://iceservicegroup.com/dt_portfolio/intermodal-containers-ip-1-strong-tight/))



**FIGURE 17.4**

Strategic packaging systems 9 yd3 IP-1/IP-2 flexible package. (Courtesy of Strategic Packaging Systems, LLC, <http://www.spsonline.biz/packages/loading-frames/>.)

Shipper self-certification of packagings not certified by the manufacturer should be approached with caution. For example, some shippers consider a standard cargo container capable of meeting the IP-1 design and requirements. However common this certification may be, it is nonetheless incorrect. Standard cargo containers must not be certified as general use IP-1 packagings. There are two main reasons why these containers do not qualify as IP-1 packagings:

1. The wood slat floors and single gasket doors of a standard cargo container do not provide single-barrier assurance that the packaging will not release hazardous materials to the environment during conditions normally incident to transportation (as required by 49 CFR 173.24(b)(1)).
2. The fork pockets on a standard cargo container are rated to lift the loaded package at the origin or destination site, but they are not rated with the required three times yield strength for use in transportation activities regulated by the DOT (reference 49 CFR 173.410(b)).

A standard cargo container can be made to meet IP-1 and certified by the shipper to meet that standard. Recall that regulations define the “package” as all of the elements of the shipment, including the radioactive contents, packing, shielding, and all packaging elements. Markings and labels must be applied to the outside of the finished package, but the packaging (including all the features required to meet the package performance requirements) is not defined by the outer layer of the packagings.

Hence, a shipper may use a standard cargo container as an IP-1 package, so long as the shipper provides internal packaging sufficient to ensure the containment of radioactive materials during conditions normally incident to transport. Properly secured and IP-1 rated drums inside a cargo container would allow the package (defined as a cargo container made sift-proof by the use of internal containers) to be rated IP-1. The use of a liner on the floor of the container,

and/or caulking and sealing of the wood slat floor is also highly recommended to further ensure that there is no loss of contents during shipment. In addition, the fork pockets must be blocked or otherwise made “inoperable” during shipment (per 49 CFR 173.410(b). “Inoperable” is not defined in the regulations, but an administrative control (a container marking and loading instruction prohibiting fork pocket use) may be sufficient to meet this requirement.

Smaller IP-1 rated containers may include metal boxes, drums, and pails. Smaller containers may not have “lifting attachments.” The lack of a feature mentioned in 49 CFR 173.410 does not prevent a packaging from being rated IP-1. Section 173.410 merely establishes the performance requirements of these features if they are present; the section does not require that a package actually include all the listed features. For example, the US DOT has concurred that a lined railcar can meet the IP-1 standard for packages (reference US DOT interpretation reference no. 05-0138 dated July 5, 2005), even though neither the gondola railcar nor the liner is designed to be lifted. Because there are no lifting attachments, the three times capacity lifting requirement is not, and need not be, met for the packaging to be rated IP-1.

### 17.5.3 IP-2

IP-2 packaging design requirements are similar to those of IP-1 packagings, with the addition of demonstrated performance of the drop and stacking tests of 49 CFR 173.465(c) and (d), respectively. The drop and stacking tests are two of the tests required of a Type A packaging, so the pedigree of an IP-2 approaches, but does not necessarily equal, that of a Type A.

### 17.5.4 IP-3

IP-3 packagings are required to meet all the applicable requirements and test criteria in 49 CFR 173.412 and, by reference, 49 CFR 173.465. These are the requirements established for Type A packagings. IP-3 and Type A are essentially equivalent packaging standards. Figure 17.5 shows a Type A 7A, IP-3 Cargo Container (37 cubic yard capacity).



**FIGURE 17.5**

MHF services Type A cargo container. (Courtesy of MHF Services, [http://www.mhfservices.com/Files/Admin/PDFs/37\\_CYDA\\_7A\\_Cargo\\_Container.pdf](http://www.mhfservices.com/Files/Admin/PDFs/37_CYDA_7A_Cargo_Container.pdf).)

### 17.5.5 SCO Wraps and Coatings

An SCO wrap can be used to ship SCOs unpackaged, or if an IP-1 rated packaging is required. The SCO wrap is especially useful for large, surface contaminated pieces of equipment when rigid packaging is not available or not feasible. For example, SCO wraps have been used for compliant shipments of nuclear power plant turbine rotors, steam generators, pressurizers, and other large components. The wrap consists of a durable plastic fabric covering, tightly secured around the piece, and sealed to prevent the release of contents. Lifting the piece can be accommodated by sealed penetrations through the wrap to uncontaminated lifting eyes or loops. Similarly, securement on the conveyance can be accommodated through the wrap. Figure 17.6 shows a large contaminated component wrapped for shipment.

Durable coatings can also be applied to the outer surface of an SCO and serve to fix otherwise loose contamination for shipment in compliance with 49 CFR 173.427(b) or (c). A fixative coating may not qualify as a packaging in and of itself but can be used to address loose contamination issues on large objects in conjunction with other measures. Figure 17.7 shows four large power plant components that were coated for shipment.

### 17.5.6 Freight Containers used as IPs

In a 2004 revision of 49 CFR 173.411(b)(6), the US DOT inserted new language to allow freight containers to be used as types IP-2 or IP-3, subject to specific conditions, specifically, freight containers can be used as IPs provided that



**FIGURE 17.6**

ICE Services Group, contaminated component in an SCO wrap. (Courtesy of ICE Service Group, Inc., <http://iceservicegroup.com/>)

**FIGURE 17.7**

Waste control specialists disposal of power plant components with contamination fixative coating.

- (i) The radioactive contents are restricted to solid materials
- (ii) They satisfy the requirements for Type IP-1 specified in paragraph (b)(1)
- (iii) They are designed to conform to the standards prescribed in the International Organization for Standardization document ISO 1496-1: Series 1 Freight Containers—Specifications and Testing—Part 1: General Cargo Containers; excluding dimensions and ratings (IBR, see Section 171.7 of this subchapter). They shall be designed such that if subjected to the tests prescribed in that document ... they would prevent ... loss or dispersal of the radioactive contents ...

Some shippers are embracing the new provision as a broad license to use standard freight containers as IP-1, IP-2, and IP-3 packagings. However, closer examination of the container specifications and the requirements of this regulation suggest that the practice is not compliant with the regulations. The solid materials limitation is straightforward, but considering the next two requirements in turn:

- (i) It meets the requirements for Type IP-1 specified in paragraph (b)(1).

The requirements of an IP-1 packaging are not especially restrictive and do not require testing or independent certification. However, the requirements of 173.410, and the requirements of 173 Subpart A and B apply to the container design, manufacture, and performance. While many container suppliers will certify standard freight containers to the IP-1 standard, shippers responsible for compliant package selection and shipment certification are advised to require the packaging supplier provide evidence that the design and manufacture of a freight container supports an IP-1 certification. The greatest challenge to a standard freight container is the requirement of 173.410 (b), that

Each lifting attachment that is a structural part of the package must be designed with a minimum safety factor of three against yielding when used to lift the package in the intended manner, and it must be designed so that failure of any lifting attachment under excessive load would not impair the ability of the package to meet other requirements of this subpart. Any other structural part of the package which could be used to lift the package must be capable of being rendered inoperable for lifting the package during transport or must be designed with strength equivalent to that required for lifting attachments.

Freight containers typically bear only one certification, the International Convention for Safe Containers (CSC) plate. For a container to qualify for CSC certification, its fork pockets must be tested to 1.25 times the rated capacity of the container. A CSC plate alone provides no assurance that the container fork pockets meet the three times capacity requirement of the 173.410(b). Indeed, an engineering analysis commissioned by MHF Logistical Solutions indicates that the fork pockets of a standard freight container do not meet the safety factor of 3 required by the regulations (assuming a wood-floored container tested to 1.25R). Shippers are advised to request supporting evidence of fork pocket capacity before accepting a standard freight container for use as an IP-1 packaging.

- (ii) It meets the standards prescribed in the International Organization for Standardization document ISO 1496-1: "Series 1 Freight Containers—Specifications and Testing—Part 1: General Cargo Containers...

Again, the only certification typically found on a freight container is the CSC plate. The CSC plate certification alone does not assure that the container has been constructed to the ISO-1496 standard. Investigation of a container certification may reveal that it meets both standards. However, the only way the shipper can assure that a container meets the ISO 1496-1 standard is to obtain certification of the design from the manufacturer. A link from the CSC plate to the ISO 1496 design may be possible from the approval reference number on the CSC plate. Freight containers are typically of foreign manufacture, and tracing the CSC plate approval number to documentation of construction to the ISO 1496-1 standard can be difficult, if not impossible.

Shippers seeking to assure that strict compliance are advised; CSC plate certification and ISO 1496-1 construction certification are not equivalent, because the standards are not equivalent. For example, ISO requires fork testing to 1.6 times capacity, versus the CSC's 1.25 times requirement.

- (iii) ... They shall be designed such that if subjected to the tests prescribed in that document...they would prevent ...loss or dispersal of the radioactive contents...

This language is a significant deviation from the language of IP-1 certification. IP-1 packages are required to prevent the release of their contents in conditions normally incident to transportation. There are no tests or rigorous documentation requirements to meet this IP-1 standard; it is generally considered to be a performance-based standard. However, use of a freight container as an IP-2 or IP-3 industrial package does require testing in accordance with this requirement. Neither ISO nor CSC establishes any requirements to test or certify containers against the release of their contents. Therefore, determination that the containment

requirement is met is left to the shipper. Presumably, compliance may be determined by any of the methods prescribed in 173.461 (full-scale testing, models).

The tests prescribed in ISO 1496 include transverse and longitudinal load testing for which the pass criterion is no permanent deformation. Temporary deflection of container walls of up to 60 mm (sufficient to temporarily unseat doors, lids, or other sealing surfaces) is entirely acceptable during the load test. A container passes the test if it experiences no permanent deflection.

Because temporary deflection can compromise container integrity, freight containers meeting ISO-1496 design and test requirements *will not necessarily prevent loss or dispersal of radioactive contents if used to ship loose bulk materials when subjected to the tests of that standard*. Therefore, it is not sufficient for a container to meet IP-1 and ISO-1496 requirements for it to be used as an IP-2 or IP-3 package. Such use (in accordance with 173.411(b)(6)) requires that the container perform as an IP-1, preventing loss or dispersal of its contents *while subjected* to the test conditions of ISO-1496.

The US DOT has stated the need for testing freight containers for their ability to prevent leakage of their contents in interpretation numbers 06-0063, dated June 16, 2006; and 07-0115 dated February 12, 2008. In addition, the UK Department for Transport has issued a *DfT Guide to the Approval of Freight Containers as Type IP-2 and Type IP-3 Packages*, expressing similar reservations about loosely interpreting the requirement of this new provision in the regulations (Webster, 2005).

### 17.5.7 Fissile Materials Packaging

Allowable fissile materials packaging is described in 49 CFR 173.417. A fissile-rated Type A or Type B packaging is required for fissile material. Recall that fissile materials cannot be shipped as LSA unless they are fissile excepted per 49 CFR 173.453, so industrial packages do not qualify (with the possible exception of IP-3 packages shown to comply with Type A and fissile requirements). Limitations on the use of Type A packagings for fissile shipments are set forth in 10 CFR 71.

### 17.5.8 Pyrophoric Class 7 Materials

Shippers must be particularly cautious and aware of pyrophoric shipping requirements as promulgated in 49 CFR 173.418. Some shippers erroneously assume that restrictions on the shipment of pyrophoric uranium no longer apply because the material-specific proper shipping name was removed from the hazardous materials table (49 CFR 172.101) in 2004. Pyrophoricity represents a dangerous additional hazard of some radioactive materials that did not go away when the proper shipping name was removed from the table. Shippers must consider the pyrophoricity of radioactive materials in the shipment planning process, and limit the contents of specified packages in accordance with the regulations. Note that 49 CFR 173.418(a) is an outright prohibition on the shipment of radioactive material that is both fissile and pyrophoric; none of the subsequent treatment or stabilization steps listed in this section overcome this prohibition unless the contents are fissile excepted in accordance with 49 CFR 173.453. Other property-specific packaging requirements (for oxidizing materials and for uranium hexafluoride) are found in 49 CFR 173.419 and 173.420.

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## 17.6 Transport Methods and Specific Requirements

Selection of a means to transport radioactive material will depend on the answers to conventional questions such as the size and weight of the shipment, cost, and speed of delivery. However, the shipper of radioactive materials must be aware of other considerations as well.

### 17.6.1 Exclusive Use

The term “exclusive use” appears in the hazardous materials regulations for transportation, including those governing LSA shipments. 49 CFR 173.403 defines exclusive use as follows:

Exclusive use means sole use by a single consignor of a conveyance for which all initial, intermediate, and final loading and unloading are carried out in accordance with the direction of the consignor or consignee. The consignor and the carrier must ensure that any loading or unloading is performed by personnel having radiological training and resources appropriate for safe handling of the consignment. The consignor must provide to the initial carrier-specific written instructions for maintenance of exclusive use shipment controls, including the vehicle survey requirement of Section 173.443(c) as applicable, and include these instructions with the shipping paper information provided to the carrier.

Exclusive use status can only be claimed by the shipper if the above requirements are met, and if specific written handling instructions have been provided to the carrier. In contrast, nonexclusive use carriers (e.g., commercial air and ground freight companies) do not receive specific handling instructions from the shipper, are not expected to have specific radiological training, and are not expected to survey packages or vehicles for compliance with dose or contamination requirements.

One aspect of the exclusive use definition is that it specifically allows intermediate handling, loading, and unloading, so long as it is performed under the direction (via the written instructions) of the shipper. Direct, in-person physical oversight of intermediate handling by the shipper is not required. Hence, it is perfectly acceptable for an exclusive use shipment of a cargo container to be transferred between different modes of transportation, as long as written instructions controlling that transportation are provided to the carrier. Exclusive use can include transfer of the container from truck, to rail, and back to truck again for delivery provided that the transfer activity is anticipated, ordered, and controlled by the shipper. US DOT interpretations offering clarifying guidance on the implementation of exclusive for radioactive material shipments include 03-0061 dated August 28 2003 and 04-0072 dated July 22, 2004.

### 17.6.2 Type B Notification Requirements

The NRC requires licensees shipping waste in Type B packages in excess of certain limits notify each state on the route in advance of the shipment, as described in 10 CFR 71.97. The Conference of Radiation Control Program Directors (CRCPD) publishes a helpful list of these and other State and Federal Agencies Involved with the Transportation of Radioactive Material a document available at no cost on the organization’s website: <http://www.crcpd.org>

Notification to the NRC and the states is also required prior to the transport of Category 1 radioactive material quantities of concern, as described in 10 CFR 37.77. The NRC’s NUREG-2155 provides helpful guidance on the implementation of security requirements.



### 17.6.3 Transloading

“Transloading” refers to the transfer of packages or materials from one mode of transport to another, or from one packaging to another. Transloading is defined in 49 CFR 171.8 as follows:

Transloading means the transfer of a hazardous material by any person from one bulk packaging to another bulk packaging, from a bulk packaging to a nonbulk packaging, or from a nonbulk packaging to a bulk packaging for the purpose of continuing the movement of the hazardous material in commerce.

Related definitions include loading incidental to movement and unloading incidental to movement. Transloading allows for multiple bulk packages to be consolidated into a single shipment (e.g., the consolidation of bulk bags of radioactive material into a single shipment in a gondola railcar). The definition is of particular importance because it establishes that packages remain in transportation when they are being transloaded from one conveyance to another.

An important concept to remember is that multiple modes of transportation can be used in a single shipment, and the routing between modes must be reflected in the shipping documentation. Failing to properly communicate the planned transloading in shipping documentation may be in violation of exclusive use requirements and invites enforcement actions on other aspects of the shipment such as having to resurvey packages for each mode of transportation. The preamble to the regulation as published in the Federal Register, dated April 15, 2005, pages 20018–20034 provides helpful instruction and clarification.

Transloading may be necessary because of logistics considerations (e.g., the desired mode of transportation not being directly accessible at the origin site). However, transloading can also offer significant cost savings for shippers of bulk quantities of radioactive materials. Railroad car capacity is several times the capacity of a standard truck, but long-haul railroad car shipping costs are only incrementally higher than truck transportation costs over a similar route. Hence, direct rail or transloaded shipments are generally considerably more cost-effective than straight truck transport of large quantities of material. Also, rail shipment is inherently safer than over-the-road shipment by truck.

### 17.6.4 Carrier Qualifications

The shipper should ensure that the carrier selected for the radioactive material shipment is professionally qualified and is in compliance with federal registration requirements. Some of the basic requirements to check include

- Carrier hazardous materials registration in accordance with 49 CFR 107.601
- Existence of a hazardous materials security plan as required by 49 CFR 172.800
- Satisfaction of training requirements in 49 CFR 172.700, including specific training in radioactive materials
- Hazardous materials endorsements on individual driver’s licenses
- Existence of a hazardous materials emergency response plan

Carriers are not all equally qualified to transport radioactive materials. It’s wise to verify that the carrier is fully qualified well in advance of the shipment date, and to share as much information as possible with the carrier to obtain the carrier’s concurrence that they are qualified to complete the shipment.

### 17.6.5 Contamination and Radiation Control

In addition to the dose limits that apply to limited quantity and LSA shipments, there are radiation level limitations set forth in 49 CFR 173.441. Allowable radiation levels are generally limited to 2 mSv/h on the surface of the package, and 0.1 mSv/h at 1 m. This section also contains specific provisions for shipping packages reading up to 10 mSv/h on the surface of the package.

Loose (nonfixed) surface contamination limits are established in 49 CFR 173.443. These are levels established for allowable surface contamination on packages of radioactive materials shipments subject to Class 7 controls. The limits apply to contamination on the surface of the package, which is assumed to be 10 times greater than the amount of contamination measured on the wiping material used to collect loose contamination from the surface. This is in sharp contrast to surface contamination measurement techniques employed elsewhere (e.g., in some NRC and DOE guidance), in which the regulated contamination level is the level detected on the wiping material.

### 17.6.6 Special Permits

Shippers will sometimes encounter a shipment for which there is no reasonable and compliant shipping approach. The shipment may require a variance or waiver in the regulations to allow for the use of alternative packaging or test procedures. Special permits can be obtained for such shipments, and can be used repetitively. The permits and the application process are described in 49 CFR 107 Subpart B, beginning in 49 CFR 107.101. Shippers can obtain their own unique permit or can request party status to utilize an existing permit (see 49 CFR 107.107). A library of existing special permits is maintained on the RAMPAC and PHMSA websites.

### 17.6.7 Specific Requirements

While the hazardous materials regulations establish general transportation requirements, there are also modal specific requirements that must be considered in shipment planning (U.S. Department of Transportation, 2008). These include

- Carriage by rail: 49 CFR 174
- Carriage by aircraft: 49 CFR 175 (see also 49 CFR 171.11)
- Carriage by vessel: 49 CFR 176
- Carriage by public highway: 49 CFR 177

Each of the modal specific parts of the hazardous materials regulations contains subparts or paragraphs specific to the carriage of radioactive materials. When transporting by rail, the shipper should pay special attention to methods used to secure the radioactive cargo in the railcar; the forces normally incident to rail transportation can be significantly greater than those generated in truck shipments. The American Association of Railroads standards can be used to establish compliance with the requirements of 49 CFR 174.700(g)(2). In addition to the cited sections of the US DOT regulations, training in, and compliance with, ICAO regulations and IATA Dangerous Goods Regulations is required.

### 17.6.8 Incident Reporting

Certain types of incidents must be reported to the National Response Center within 12 hours after occurrence. Technically, this is a carrier responsibility, but the shipper should be aware of the requirement and be prepared to assist the carrier with the preparation of the incident report as necessary. A reportable incident is one in which, as a direct result of the hazardous materials being shipped, any one or more of the following occurs:

- A person is killed.
- A person receives an injury requiring admittance to a hospital.
- The general public is evacuated for one hour or more.
- A major transportation artery or facility is closed or shut down for one hour or more.
- Fire, breakage, spillage, or suspected radioactive contamination occurs involving a radioactive material.

Additional requirements for reporting and detailed hazardous materials reports apply; refer to 49 CFR 171.15 and 171.16 for detailed requirements and reporting instruction.

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## 17.7 Communication Requirements

Many of the requirements of the hazardous materials regulations focus on the effective communication of the nature of the materials being shipped. Of course, basic shipping documents must communicate the point of origin and the intended destination. In addition, the nature of the hazard the material represents is communicated to the carrier, the receiver, and to emergency responders who may come into contact with the material in the event of an incident. There are four basic communication tools shippers are required to use in accordance with the HMR:

- Shipping documentation
- Markings
- Labels
- Placards

### 17.7.1 Shipping Documentation

The basic requirements for shipping papers are described in 49 CFR 172 Subpart C, beginning at 49 CFR 172.200. The US DOT does not prescribe use of a particular form, but the shipper may be driven to the use of specific forms by the NRC that prescribes the format for shipping manifests used for licensed, low-level radioactive waste (10 CFR 20, Appendix G). In addition, radioactive materials shipped together with chemically hazardous waste are subject to the hazardous waste manifesting requirements of the U.S. Environmental Protection Agency (US EPA), in 49 CFR 259 and 261.

Some basic requirements for the preparation and retention of shipping papers are found in 49 CFR 172.201. The section discusses

- Properly describing shipments that include both hazardous and nonhazardous materials
- Provisions for supplying additional nonrequired information, and for extending the shipping paper to more than one page
- Requirement to list an emergency response telephone number
- Standards for record retention

The emergency response number must be answered 24 hours a day by a person qualified to discuss the hazards in the shipment. Contracted services can be used to meet this requirement. Records must be retained for 2 years.

Detailed requirements for the description of the hazardous material on shipping papers are set forth in 49 CFR 172.202. The requirements establish a strict standard for the format and order in which a hazardous materials description is presented on shipping papers. The requirements for a proper shipping description include all of the following (when applicable):

- Proper shipping name determined for the material in accordance with column 2 of the table in 49 CFR 172.101
- Material hazard class (Class 7 for radioactive material)
- UN identification number
- Packing group
- Total quantity of material being shipped
- Total number of packages being shipped

Additional requirements may apply to the shipping description when applicable, as described in 49 CFR 173.203, including

- Identification on the shipping paper of any special permits used or issued in connection with the shipment
- Specific inclusion of the words "Limited Quantity" in the description of materials so shipped
- Designation of reportable quantities as "RQ" on the shipping document

For radioactive materials, additional specific requirements may be applicable, including

- List of isotopes comprising any radioactive material shipment
- Description of the physical and chemical form of the shipment
- Specifying the activity contained in each package
- Listing the type of label applied (e.g., "RADIOACTIVE WHITE-I")
- Transport index when required
- Specified description of any fissile material hazard present
- Notice of any DOE or NRC approval required for fissile packaging

- Fissile international shipments that require documentation of packaging competent authority approval
- Identification of the shipment as “exclusive use” when applicable, and inclusion of exclusive use handling instructions
- Additional requirements for empty packagings and modal specific instructions
- Further requirements for specific other hazards (e.g., marine pollutant, poisonous materials, elevated temperature materials)

Finally, the shipper is required to include on the shipping paperwork a statement accepting personal responsibility for the correct preparation of the shipment and shipping documentation. The requirement is set forth in 49 CFR 172.204 and includes exact language for the shipper certification statement.

While the shipper is required to express acceptance of responsibility on the shipping paperwork, the US DOT does not limit responsibility to the shipper. Indeed, the US DOT holds any and all “offerors” of the shipment to be jointly responsible for assuring compliance with the regulations. Transportation brokers may be viewed as offerors depending on their physical and contracted role in any given shipment. Refer to 49 CFR 171.2 (b) and (f) for further discussion. The definition for “offeror” is under “Person who offers” in 49 CFR 171.8. Every person involved in offering the shipment for carriage may be held responsible for the accuracy of the shipping documentation and the compliant preparation of the shipment. Additional detailed discussion of the responsibilities of offerors can be found in the Federal Register, dated July 28, 2005, pages 43638–43644.

### 17.7.2 Markings

Marking requirements for nonbulk packages are addressed in 49 CFR 172.301, and bulk package marking requirements are found in 49 CFR 172.302. Additional requirements for radioactive materials are found in 49 CFR 172.310.

Generally, information required to be communicated through package markings includes

- UN identification number of the material being shipped
- Proper shipping name and technical name of the material
- Communication of any special permit packaging being used
- Shipper’s name and address

For radioactive material, additional marking requirements apply

- Design of the packaging (IP, Type A, or Type B)
- Country of origin of the design

Additional markings may apply (and exceptions may be found) in the specific shipping regulations for the material. For example, LSA and SCO are exempt from many marking requirements in 49 CFR 173.427, but the additional marking “RADIOACTIVE-LSA” or “RADIOACTIVE-SCO” is required.

### 17.7.3 Labeling

The labeling requirements for radioactive material are set forth in 49 CFR 172.403. There are, however, many exceptions to labeling requirements in the specific provisions for limited quantity shipments, LSA, and SCO, as discussed in 49 CFR 173.421 through 173.427, and the requirements of these sections should always be reviewed first before considering the standard labeling requirements.

Standard labeling requirements for radioactive material shipments are based on the radiation level at the surface of the package and the transport index (the radiation level at 1 m, as defined in 49 CFR 173.403). Standard labels are as follows, in order of increasing dose rate:

- White-I
- Yellow-I
- Yellow-II
- Yellow-III

The required label is determined based on dose rates in accordance with the table in 49 CFR 172.403.

This section of the regulations includes discussion of other required labels, including those for

- Empty packages
- Fissile packages
- Radioactive LSA/SCO packages

### 17.7.4 Placards

General placarding requirements are described in 49 CFR 172.504. Placards are required at each end and on each side of a transport vehicle, bulk package, or freight container for hazardous materials in accordance with Table 1 of this section. Radioactive materials require Class 7 placards if package dose rates merit a Yellow-III label, and for exclusive use LSA and SCO shipments. Note that placards for a highway route controlled quantity are of a different design than standard Class 7 placards.

The requirement to placard all four sides of a vehicle can be met with placards placed affixed to packages on the vehicle. However, there is a common error made when shipping bulk packages and freight containers. Placards are commonly perceived to be a *vehicle* requirement, and some shippers believe that distributing four placards over multiple freight containers on a shipment will fulfill the placarding requirement. This is incorrect. The permission to use placards on the load to meet the vehicle requirement does not negate the requirement to placard each freight container on all four sides. It is not sufficient to have four placards on a shipment containing more than one bulk package or freight container. If there are two freight containers on a truck, each freight container must have four placards. Similarly, a railcar carrying 4 freight containers must have a total of 16 placards, 4 on each container. Vehicle placards are normally required, but the vehicle requirement

can be considered fulfilled by the placards on the freight containers. Refer to US DOT interpretive guidance letter 05-0311, dated May 3, 2006, for additional clarification.

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## 17.8 Highway Route-Controlled Quantities (HRCQ) and Spent Nuclear Fuel

Higher curie content materials are subject to additional shipping controls as HRCQ. The term is defined in 49 CFR 173.403 as a quantity within a single package that exceeds

1. 3000 times the  $A_1$  value of the radionuclides as specified in Section 173.435 for special form Class 7 (radioactive) material;
2. 3000 times the  $A_2$  value of the radionuclides as specified in Section 173.435 for normal form Class 7 (radioactive) material; or
3. 1000 TBq (27,000 Ci), whichever is least.

HRCQ shipments have special communication requirements including special placards (49 CFR 172.507) and the exclusive use-like instructions of 49 173.22(c). Highway routes are limited to preferred routes, and pre- and postshipment carrier reporting requirements apply as described in 49 CFR 397.101.

### 17.8.1 Spent Nuclear Fuel

The shipment of spent nuclear fuel has myriad requirements that will require careful assessment, including notification, escort, and fee requirements imposed by individual states along the route. The NRC requires licensees shipping spent fuel or waste in Type B packages notify each state on the route in advance of the shipment, as described in 10 CFR 71.97 and 10 CFR 73.72. The CRCPD publishes a helpful list of these and other state and federal agencies involved with the transportation of radioactive material a document available at no cost on the organization's website, [www.crcpd.org](http://www.crcpd.org). The NRC also has regulations for the physical protection of irradiated reactor fuel in transit, set forth in 10 CFR 73 Appendix D. Other security requirements relevant to shipments are found elsewhere in 10 CFR 73, including security for international shipments in Appendix E.

The notification requirements of 10 CFR 37.77 for radioactive material quantities of concern can also be expected to apply to spent nuclear fuel shipments. The NRC's NUREG-2155 provides helpful guidance on the implementation of security requirements.

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## 17.9 International Shipments

U.S. regulations governing import and export shipments are found in 49 CFR 171.12. For air shipments, 49 CFR 171.11 allows the shipper to use ICAO technical instructions to prepare the shipment. For shipments by vessel, the US DOT authorizes shipment preparation

in accordance with the IMO's IMDG Code, subject to conditions and limitations in 49 CFR 171.12. For Type B and other packages that are approved by the NRC (or for imports, similar foreign agencies), a U.S. IAEA competent authority approval certificate must be obtained from the US DOT.

Corresponding IAEA competent authority validation (acceptance) must be obtained from each transit and destination country. These IAEA competent authority validations need to be included in the shipping paperwork (reference 49 CFR 173.471, 173.472, and 173.473). While a formal competent authority approval is not required for IPs, the shipper may be required to have approval for the package in the country of origin and destination. This needs to be verified prior to making any international shipment.

Packaging requirements for international shipments will be consistent with US DOT and IAEA regulations (except where the US DOT allows for domestic shipment exceptions). The basic requirements to describe the radioactive material on the shipping paperwork (isotopes, activity, proper shipping name, etc.) are also the same for international shipments. In addition, the IMDG communicates the nature of the hazardous material using the shipper's Declaration of Dangerous Goods (DGD). International shipping will generally require the use of a broker and freight forwarder to coordinate with the shipping line and authorities in ports of call on the shipment route. The broker and freight forwarder can provide assistance with the DGD handling, coordination with the carrier, and customs and tariff issues (Brown and Woods, 2004).

Depending on the materials being shipped, additional documentation may be required to prepare for the shipment, or as part of the shipping documentation, including

- Valid export license for nuclear material
- Valid import license for nuclear material
- Shipper's Export Declaration Commerce Form 7525-V
- Commercial Invoice
- Packaging List
- NRC 741 form for accountability control of special nuclear material
- Physical protection agreement
- Sealed source and leak test certificates
- Emergency response guide
- Locally required transportation documents

Confirmatory information the shipper should require from the broker or freight forwarder for marine shipments include:

- Dock receipt
- Vessel and booking number
- Voyage number
- Last received port of export
- Vessel departure from port of export



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# 18

## *Decontamination and Decommissioning*

Cidney B. Voth

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### **18.1 Background of Decontamination and Decommissioning (D&D)**

“D&D” is the act of decontamination and decommissioning a facility for further action reuse, demolition, or another purpose. Decontamination stabilizes the area by taking a material and removing it from an unwanted place and placing in a more controlled situation. Decommissioning is the act of transitioning an area, facility, or site out of its current or previous operations or mission and placing it into a new state. The term D&D can not

only apply to the nuclear industry but to remediation of areas, chemical industries, construction industries, military actions, environmental actions, and large or small business actions as well. The definition of decontamination does not necessarily denote treatment or the stabilization of the material. One may presume that the material is unstable prior to decontamination but that is not normally the case. The act of decontamination generally poses the greatest hazard to the worker and the environment. The methods, controls, and actions taken during the movement of contaminated material must be made in a planned, controlled, and protected manner.

It does not matter whether you are performing D&D; DD&D (decontamination, decommissioning, and demolition); DDD&D (deactivation [involving modifications, shutdown or operations, or reuse], decontamination, decommissioning, and demolition); DDDD&D (discovery, deactivation, decontamination, decommissioning, and demolition); TDDDD&D (transition, discovery, deactivation, decontamination, decommissioning, and demolition); or even maintenance during the operation of the plant, the process and strategy for physical work remains the same. In the United Kingdom, one of the "D's used is "Deplanting." Deplanting includes materials inventory removal, equipment removal, infrastructure stripping, and soft stripping. The key to a successful project is to know what the initial conditions and hazards of the site are, what the purpose of the project is, what hazards are created by the actions to be taken, and what the end objective is for the project. This can be accomplished by the proper data gathering, planning, project management, procedures, and safety management systems. D&D is difficult because of the dynamic nature of the changing conditions that exist, not only from a physical prospective but from a regulatory, environmental, and worker protection aspect.

With these dynamic conditions, the art of D&D is to balance the characterization activities with remediation to create "design-flexible" conditions to guarantee worker safety and environment protection while new discoveries are made. Furthermore, project preplanning should empower policy makers to make optimal decisions that create a successful environment for the D&D project. There is no one project management and engineering system used for all the activities identified above. However, the following aspects of a tailored or graded approach allows D&D to be performed in an effective, efficient, safe, and controlled manner for the least cost. Many sites have been successful in their D&D efforts. Figure 18.1 shows the "before and after" of the D&D that occurred on the Fernald site near Cincinnati, Ohio. This site was transformed from a uranium processing plant to a wildlife refuge.

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## 18.2 Phases in the Life of a Site to Which D&D Applies

D&D applies to most phases in the life of a site. During operations, the site performs maintenance, equipment removals, spill cleanup, environmental remediation, and upgrades that require decontamination efforts in isolated areas of the site. These actions are essentially the same D&D actions required of decommissioning of the site. Prior to performing these activities, recognition of hazards and conditions must be taken into account to protect the worker and environment. Procedures and work control documents must be written. Proper tools, equipment, and materials must be organized. Proper training must be given. During a surveillance and maintenance (S&M) mode on a site, if proper waste containers can be obtained, there is little difference between counting and monitoring a thousand items; or putting the items in boxes during inventory monitoring so you only have to count 300 boxes.



**FIGURE 18.1**  
Aerial view of the Fernald, Ohio, site before cleanup (1987) and after cleanup (2006).

During transition or shutdown, common sense actions should be taken that can significantly reduce D&D costs in the future. If a closed system is properly shut down, hazardous waste is segregated, radiological and chemical conditions identified, and documentation collected, such as maintenance, chemical, equipment, and utility information; D&D could proceed much easier and sooner. If the plant is shutdown to a documented, stable condition, the surprises due to unknown energy sources, contamination, and changing conditions during D&D are minimized. If one looks at these simple actions during shutdown as safety requirements of the future, the initial conditions of the plant would solidify the D&D approach during planning. Shutdown and D&D are best planned together. Shutdown procedures are different for a facility to be re-used than one that will be demolished.

Discovery occurs in all phases of activities. MSDS sheets are supposed to provide some information of the chemicals on site. Routine radiological surveys provide the changing radiological conditions. Equipment, preventative maintenance, and maintenance records provide equipment status. Spill, accident, and operations data provide clues to contamination locations. Even when constructing radiological facilities, the initial design should take into account ways to minimize D&D. The ductwork and glove box systems can be modularized to fit waste containers. The floors can be curbed and covered with epoxy or solid linoleum versus linoleum squares, thus preventing the spread of potential contamination during operational cleaning. Systems can be designed with surge volumes and closed-loop characteristics to contain hazardous product. If the random locations included all areas and were precise enough during routine monitoring, the characterization of a facility could occur in the S&M phase, thus minimizing this D&D activity. This methodology could lead to simply having to monitor the buildings as sections are taken down versus having to work in a contaminated environment fully characterizing the area. Figure 18.2



**FIGURE 18.2**

Workers performing D&D, contaminated work dressed out above and typical construction work below.

depicts the containment of unwanted material and contamination while protecting the worker and the environment, depending on the hazard.

Deactivation is the cleanup activities prior to D&D. This concerted effort stabilizes and sets up the area for D&D. Again, the funding may be different but the actions are the same. Field activities must be controlled through solid project management systems. Licenses, regulations, standards, and orders drive compliant management plans, which drive the development of safety management systems and organizations to assure compliance. The management structure establishes the training and procedures of operation compliant with the regulations and orders and ultimately the scope of work, through work packages, that the worker uses to know what work is desired and how it must be performed.

Decontamination and decommissioning is sometimes identified as decommissioning and decontamination. The explanation for this is simple. Decommissioning is transferring something out of commission or operation. Some people think “cradle to grave” operations start with preconceptual design (a mission needs statement) and ends after all activities have been completed, including decontamination, demolition, waste disposal, and decommissioning. Others think of decommissioning as the act of transfer at the end of its original operations and the start of cleanup (deactivation, decontamination, and demolition) that follows. Still others think of the process of decontaminating various facilities, performing all remediation activities, both radiological and environmental, prior to decommissioning the site. When “transfer” occurs, the “life” of a site varies. At the Mound facility in Miamisburg, Ohio, shown in Figure 18.3, the decommissioning of the site was earmarked by a series of property transfers to the city for its eventual use as an industrial park.

Again, contamination is a material in an unwanted place. Decontamination is the act of removing this material to a more stable, permanent condition. A simple example is when flour is spilled on a counter. The contamination is the flour and decontamination is the act of cleaning it up. The flour could be put back in a container or thrown away but the dirtiest (most hazardous) part of the job is moving the flour residue from where it is to where it belongs. In decontamination, one must change the condition or location of the contamination, often creating more hazardous conditions in the interim, before the situation is stabilized. This is why it is so important that the worker knows the characteristics of the contamination, environment, work objectives, and procedures.

Demolition is the act of tearing something down. It is the industrial activity when it relates to commercial operations. The challenges to performing demolition as part of D&D concern are the safety to the environment and the worker. The structure must be set up so that when the structure is torn apart and/or shattered, the radioactivity and hazardous components are not dispersed to the environment or exposed to the worker. Extensive characterization, stabilization, and monitoring must occur prior to “ripping and tearing.” Safety zones must be established and support systems such as dust suppression systems, remote cameras and lighting, environmental monitors, and security must be set up.

Disposal of waste in the radioactive world is a unique process in itself. Remember, D&D is the act of making contaminated material more unstable prior to placing it in a wanted place (disposal site) in a final stable condition (the act of disposal). Disposal of waste involves packaging it into a condition that will keep it stable for hundreds, even thousands of years. Needless to say, the disposal sites have placed stringent waste acceptance criteria on how, what kind, and how much waste is disposed of on their site. Because this is the final resting place of the waste in a static condition, environmental regulations and monitoring plays an important part in assuring the waste poses no threat to the environment or the public for many years to come. Figure 18.4 shows stabilized waste awaiting shipment to the waste site for disposal.



**FIGURE 18.3**

Aerial of the effects of D&D at Mound Facility in Miamisburg, Ohio, when the site was returned to the city and converted into an industrial park.

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### 18.3 Establishment of Initial Conditions, Approach, and End State

To know the initial conditions of the site, facility, or area, upfront planning is essential. If this is a nuclear decontamination effort, initial conditions must not only include the contamination and radiological conditions present, but must also take into account the conditions that could be encountered by the chemical hazards present, the industrial hazards of the site, the environmental conditions present, and the physical condition of the facility. These factors interact to create the unique environmental condition that must be recognized in the characterization and planning effort.

First and foremost, one must understand the scope of work and environmental requirements controlling this work. One should ask: What are the controlling factors for this work and how much priority does each factor have? Obvious conditions needed are the end





**FIGURE 18.4**  
Depiction of a typical temporary waste storage facility.

point, schedule, and return on investment to obtain the proper funding level. The desired radiological control, the desired waste form for disposal costs, the required utilities, and the capability of the staff to use the chosen technology play an important part in selecting the D&D approach. The potential hazards to personnel and the environment are important in what activities will be performed. For instance, plasma/torch cutting is a fairly common practice in some plants, but there are plants that do not allow this practice due to the perceived fire hazard and safety concerns. Perceptions of operational control by different organizations within the plant play an important part in the selection of a technology. The technology must generally be able to complete the scope of work at the lowest cost, in the most rapid and most controlled manner using the available resources (equipment, utilities, services, etc.) of the plant while accommodating modulating environmental conditions and requirements. Only by understanding the full scope of the work and how it relates to other activities can one select the best technology.

Another important factor to know is what technologies are available to perform this work. Sources of this information include the D&D Handbook, various EM-24 websites, experience of personnel that have performed similar scopes of work, technology exchange between individuals and sites; and specific focal points for technology such as the National Energy Technology Laboratory, the Remedial Action Public Information Center, Large

Scale Decommissioning Demonstration Projects, innovative technology status reports, vendor solicitations, and Internet searches. Technology acquisition can be one of the hardest processes to accomplish, but may have significantly impact the project, because experience plays such an important role in the selection process.

It is somewhat outdated, but the Department of Energy (DOE) described the D&D process in their DOE/EM-246 Decommissioning Resource Manual, August 1995. Some of the organization and DOE orders, manuals, and guides have been replaced, but it is a good reference for the philosophy and practices that could make D&D effective. Another reference is DOE/EM-0142P Decommissioning Handbook, dated March 1994, and the ANS/ASME/DOE funded Decommissioning Handbook, dated 2004. These documents describe different types of field equipment available for use, broken down into mechanical, chemical, and demolition devices. There are also new and improved technologies found on the Internet that can provide a basis for a design-flexible D&D arsenal. Design-flexible equipment and devices that can perform under a variety of changing conditions are most valuable in D&D due to changing conditions and new discoveries, both chemical and radiological.

More recently, the DOE documented the planning function through DOE Order 413.3A, guidance from the Office of Environmental and Construction Management. This establishes a series of approvals that control the D&D process through funding obligations. Private industry and the nuclear industry have their own methods for planning and budget formulation, prior to design, which is far less rigorous. DOE Order 413.3A describes a methodology of how a D&D project gets money for design and contract acquisition up to and including construction. Construction is driven by an accurate activity-driven, fully resource-loaded, scheduled baseline whose work breakdown structure (WBS) matches up with how the project costs are reported (invoiced). The DOE O 413.3A document with its corresponding manual can be obtained over the Internet in the DOE Orders website.

In DOE Order 413.3A, the first activity is the generation of a Mission Needs Statement (Critical Decision (CD)-0). This step allows the department to determine if the project is required to meet the mission. For construction projects, Congress generally establishes and approves these line items individually. For D&D, the money for the project generally comes out of the government's operating budget, in which money is applied to various projects on an annual basis. There are generally more projects requested than there is money to perform these cleanups. The Mission Needs Statement is generated to make the compelling case that this particular D&D project be given priority. During this phase, a key activity is to identify alternatives of how project objectives can be accomplished most effectively. The approval of CD-0 data demonstrates that this project has government backing and will satisfy the state environmental requirements and agreements when completed. Commercially, the need for D&D is determined through return rates, profitabilities, changing regulatory requirements, pay back periods, and so on.

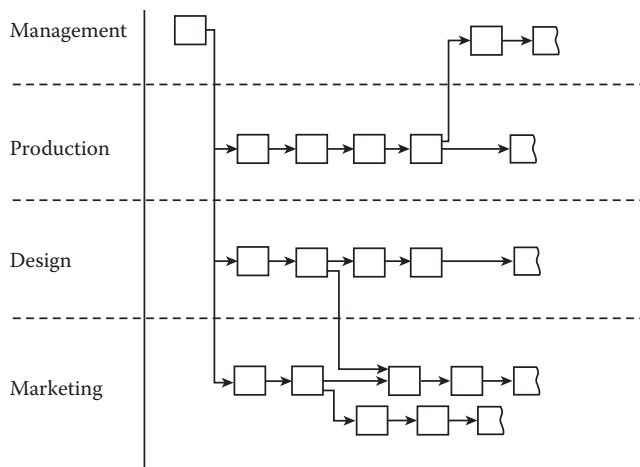
Following the Missions Need Statement (CD-0), the preliminary design, the conceptual design report (CDR), a range of estimated cost, and associated documentation are generated for CD-1 approval. The CDR provides key information to determine if the government has the funding and resources necessary to perform, in this case, the D&D project on a lifecycle basis. The CDR includes a description of the D&D process, the establishment of the recommended alternative from an alternatives analysis, a range of estimated cost, risk assessments, a preliminary project execution plan (PEP), a hazard and security analysis and an acquisition strategy. The scope of work must be established in the preliminary design.

The CD-1 preparation is really the formulation of the project parameters and scope, with major constraints and assumptions identified. The regulatory process (RCRA, CERCLA, TSCA) that controls the D&D process and the permits, release levels, control levels, and points of emission required are identified. How the project is to be contracted is decided. Major project activity interactions are documented so that the project can demonstrate a cradle-to-grave concept. The format of the baseline and a high-level WBS structure is created. From an operational standpoint, the D&D site begins to “take shape and form” and the general conditions of the site pertaining to the upcoming D&D begins to be recognized. The site’s required S&M is coordinated with the D&D effort. This data is vital to progress the design in the CD-2 process, where the specific activities for D&D above are solidified and quantified.

Following CD-1 approval, the CD-2 process involves the detailed design of the project and establishment of a usable, activity-driven, fully resource-loaded, scheduled baseline whose WBS matches up with how the project costs are reported (invoiced). A typical baseline shown includes a logic network that should be developed by this time. Commercially, the information is ready to bid as a construction contract. Approval of CD-2 should show field work is ready to begin, demonstrated by an Operational Readiness Review (ORR). Procurement and acquisition of long lead items should be considered during the design phase (CD-2) to enhance the construction phase (CD-3). A typical baseline layout is shown in Figure 18.5.

Other support documentation generated during the CD-2 preparation phase include finalizing the PEP, updating safety and security documentation, completion of the NEPA and environmental documentation, development of a Construction Safety and Health plan, establishment of the Authorization Basis and Safety Management Systems for the project, validation of the baseline through an independent external review, and approval of design reviews and readiness assessments.

Following CD-2 approval, the CD-3 process involves the construction phase of the project. The CD-4 process is the transfer of the area for re-use and closeout activities. Control of these phases in a D&D project will be discussed under baseline control and project management.

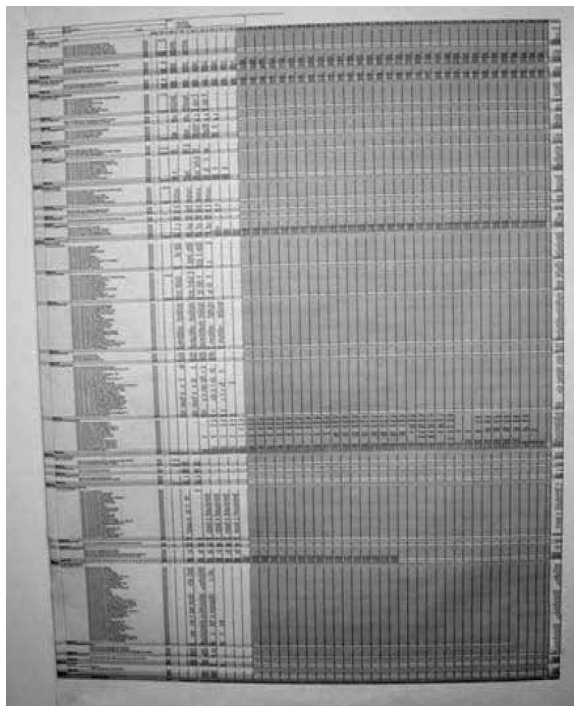


**FIGURE 18.5**  
Typical baseline layout.

### 18.4 True Cost of Performing D&D Work in the United States

Part of creating an effective D&D process is to examine where and how the money can be spent. Several areas of operations (what should be support functions to the project) can greatly affect the ability to perform the project objective (clean the site, take down and properly dispose of the facilities, dig up the contaminated soils, and turn the completed project site back over to the owner). The physical labor and material (physically performing the D&D activity) controls only 20% of the funding, including the health physics, waste management, and laborer support. Overhead costs, general and administrative costs, and profit can consume up to 50% of the total funding. Then there are “hotel” costs such as site infrastructure and utility costs that consume up to 20% of the total funding. The field support operations such as supervision, project management, oversight, maintaining documentation, and reporting are another 10% of the total. A typical funding profile, shown in Figure 18.6, should match up with the lower baseline activities within the higher project WBS levels creating the Validated Site-Wide Life Cycle Baseline. Although this funding profile is not readable, it shows the volume of baseline activities that need to be generated with the funding profile of a typical site. Each of these funding profile activities must be matched up with the scope of work in contractual vehicles.

Most of the funding is controlled by the operational managers and administrative requirements outside the project manager’s control of the project. The contractor’s effectiveness when performing activities will be determined by how he controls his administrative and site costs. Some upfront questions needing to be addressed for a successful



**FIGURE 18.6**  
Typical funding profile for a site’s baseline.

project are: What is the true objective of the funds being expended? Is the objective of the project to perform physical activities, relieving the customer of the liabilities they have on site, and returning the site back to society as efficiently and quickly as possible? Or is the objective to study and identify all the methods available to each activity while minimizing safety risk; to reach the other objectives as effectively as possible? Does everybody have the same project objectives in mind?

The integrating contractor sets the direction and tone of how the funding will be spent. Direction is set through procedures and work packages to empower the worker to do the work. The field worker performs the activity and meets the objective of the project. If the customer, contractual vehicles, integrating contractor, oversight personnel, inspectors, or regulators have different objectives than allowing the field worker to perform safe, compliant closure of the site at minimum risk to the worker and the public, then money will be captured for other activities and operations. Operations should support the project rather than the project supporting ongoing operations. Employ only the resources required to perform the contract objective at that time, that is, supply the project with the right mix of expertise and equipment only when those resources are needed and do not burden the project with ineffective operations.

There are two defining areas that will help determine the financial success of the project. One key area is the determination of the requirements and the pathway to compliance (DOE, Nuclear Regulatory Commission (NRC), NEPA, CERCLA, or RCRA). The second key area is the assignment of who controls the project coordination activities; how the project is performed and met. This requires one to define job performance, interpret the requirements, and establish the integrated objectives before the job starts.

The project manager needs to know who makes the interpretive decisions and how conservatively. CFRs, OSHA, EPA, and NRC regulations as well as QA, safety systems, DOE Orders and Procedures properly set limits on the way a project will be performed and on how it is reported or documented in terms of the project progress, performance, and status. However, the limits and wording pertaining to these extensive regulations and orders can be interpreted in a variety of ways, while they must be complied with. ALARA is as "low as reasonably achievable." What is the definition of low, reasonable, or achievable? What is the definition of compliant and how are the limits interpreted?

Furthermore, the client may set more conservative cleanup criteria than the regulations. Operations may demand more conservative methodologies than are required in the regulations. Interpretations into "What is clean?" can cause confusion in how many dollars need to be spent. Is a free release criteria defined? What status is a building in, that is, a nuclear facility, a radiological facility, or a slightly contaminated facility? How will the waste be disposed of, regardless of the D&D effort involved? Many activities require a competent person for inspections, observation, and interpretation. Who is competent? What documentation and experience are required? How much credit can be taken for engineering controls? Who can perform what analysis and write what procedures? And who has control of the project coordination activities to perform the project objective which way? Only the person that controls the answers to these questions can fund projects efficiently. A good project manager must influence and control project costs and establish the project organization early.

In the endeavor to control standards in the areas of quality, safety, conduct of operations, instrument accuracy, and so on; the balance of risk and performance of the mission and project performance can get skewed. Risk for doing work through work packages, procedures, and instructions under a certain set of standards can always change to higher expectations, causing additional funding and effort than estimated to meet the objectives

of the project. The expected standards of quality and safety must be determined and integrated into the original approach of effectively performing the work. How the work is to be performed must reflect compliance with the interpreted standards and yet complete the work effectively with the expertise and resources available. The standards must be built into the day-to-day process to add value to the mission, not be separate or cumbersome to the mission. For instance, it makes no sense to encumber low risk routine and repetitive activities, or off-the-shelf product procurements with NQA-1 quality requirements and reviews. Activities should be balanced by meeting appropriate standards, minimizing risk, and utilizing a tailored or graded approach.

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## 18.5 True Cost of Performing D&D Work Internationally

The physical process of performing D&D internationally isn't significantly different than in the United States. Internationally, however, the approaches to characterization, planning, risk assessment, and execution processes are prioritized differently, particularly in the control and standards being used. Also, the definition of when D&D starts, how it proceeds, and when it ends may be different. In the recent Fukushima Daiichi NPP accident in Japan, from a US perspective, initial conditions of D&D were defined as starting after the emergency management actions were taken to stabilize the incident, but those same emergency actions could have been considered to be part of D&D (stabilization). Rather than recreate all the conditions that the different countries perform in the "Act of D&D," this section will focus on the significant differences in strategy and standards internationally that three countries (France, Japan, and the United Kingdom) use in D&D. The data and documentation for this section are derived from experts of interviews of D&D Project Management personnel that have experienced both the United States and international D&D processes and from a recent lessons learned document put out by the Organization for Economic Cooperation and Development and the Nuclear Energy Agency.

### 18.5.1 France

D&D in France has taken a much more remote approach than the United States. As the labor conditions in France are much different than the United States, the cost and "act of D&D" are simplified by using more remote-handled machines for hazardous operations. The use of machines provides a much more stable process, regardless of changing radiological and biological conditions. There is less safety risk as the job progresses and encounters changing conditions. The technology and the material cost upfront are much larger but the machine usage can be a constant from job to job. The use of minimal labor and remote handling allows minimal process changes regardless of the conditions. Machines don't have unions, can't ask for wage increases, are simply stored when not in use (no severance or worries about labor effects of a region), and are available for indefinite periods with operators that only need to be trained once with continuous proficiency training. Machines may actually have a longer lifetime and are cheaper to utilize than a massive labor force.

The French have a much easier time relating their field work to different jobs with changing conditions because their approaches center on the technology available and it

is simply the remote machines at risk versus their people. With the acceleration of remote technology improvements that the world has experienced in the last few years, the French approach can be cost-effective while utilizing limited resources.

### **18.5.2 Japan**

Japan's Fukushima Daiichi NPP accident is having a profound effect on the changing international regulatory and emergency preparedness environment. The simultaneous earthquake and tsunami events caused beyond-design-basis events to occur. Specifically, the plant experienced major adverse conditions through a complete and prolonged loss of electric (both grid power and emergency power) and the loss of the plant's ultimate heat sink, which is required to stabilize the nuclear conditions and protect the equipment and auxiliary systems. The result of this event caused an international assessment of the defense-in-depth the nuclear community applies to minimize and/or mitigate accident scenarios of this nature within the authorization or design basis. It also revealed a need for facilities to be more robust and for improved collaboration, communications, and transparency among the nuclear international community. Within Japan's defense-in-depth, as well as in other countries, the simultaneous upset conditions experienced had not been completely considered credible and/or realized. Therefore, many countries are considering beyond-design-basis accidents within their regulatory framework.

If one considers part of D&D to be the stabilization of dynamically changing conditions, then the emergency actions taken after the incidents could provide lessons learned for both characterization and the act of D&D. Possible mitigations could include the use of more remote equipment available, larger water storage, ponds, quick connections for water and other services, filtered containment venting, hydrogen management, robust electrical systems with connections for portables, change out-of-use equipment as it wears out, maintenance, and aging.

Post accident recovery is an inherently complex situation depending on the conditions that have been stabilized. As such, Japan is taking a cautious, systematic, slow, and methodical approach to D&D because conditions and characterization are not well known. They do not want to unduly upset stabilized conditions achieved through emergency actions to perform D&D. Furthermore, from a radiological aspect, time allows the short-lived isotopes to decay, which lets nature continue the stabilization process. Therefore, the comprehensive D&D approach is still being developed as Japan addresses infrastructure and containment issues. This allows them to continue the characterization and assessment of conditions while improving the overall site situation.

### **18.5.3 United Kingdom (UK)**

The observation of the UK D&D is mainly derived from the Sellafield nuclear reprocessing site D&D. In the United Kingdom, the Nuclear Decommissioning Authority (NDA) is a nondepartmental public body created under the UK Energy Act (2004) to ensure that the historic civil public sector nuclear sites are decommissioned and cleaned up safely and efficiently. The NDA has a very clear remit that is limited to decommissioning and clean up plus running existing designated operational facilities efficiently and effectively until operational activity ceases.

The UK's historic nuclear legacy consists of nuclear sites and facilities operated by the United Kingdom Atomic Energy Authority (UKAEA) and British Nuclear Fuels plc (BNFL); the wastes, materials, and spent fuels they produced between the 1940s and 1960s; and the

Magnox fleet of nuclear power stations designed and built in the 1950s, 1960s, and 1970s and operated on the Government's behalf by BNFL; plant and facilities at Sellafield used for the reprocessing of Magnox fuel; and all associated wastes and materials. The NDA and its Site License Company contractors must comply with extensive UK and European Union legislation and regulations which, compared to the United States, often create a more restrictive D&D working environment. Minimization of risk to the worker, public, and protection of the environment (keeping the conditions stable) is a higher priority than how fast or efficiently the work is performed. That does not mean that work cannot be performed across time, but how the work is performed is much more restrictive. As such, the United Kingdom is much more deliberate in determining the best practicable method and the best practicable environmental option to accomplish the task while creating the least change in conditions.

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## **18.6 S&M Phase and the Activities Performed to Enhance D&D**

In generating the design for D&D, several principles and assumptions should be applied. To use the most effective methods and activities for D&D, one must have a fairly good characterization of the site. Then you should use design-flexible methods for remediation, anticipating changing conditions. A machine that accommodates higher levels of contamination removal than presently exists, or a machine that can accommodate chemical as well as radiological hazards, is much more effective in the long run than machines that meet only known requirements. Invariably, during the D&D remediation process, the regulatory environment will change and new hazards will be discovered. These conditions need to be anticipated and taken into account to assure a safe and efficient operation. The D&D project design should focus on the end state, a cradle-to-grave approach, and a flexible, systems engineering design to accommodate changing site conditions and unexpected discovery. Considerable savings can be realized by using S&M operations to enhance conditions of the site prior to the D&D process.

For example, Battelle's West Jefferson North site was originally funded in a traditional S&M mode. While awaiting D&D funding, the site was being maintained through inspection, monitoring, repair, and surveys. Critical systems and equipment required to maintain stable radiological conditions were being kept in an operational condition. The radioactive and hazardous materials on site were identified, segregated, stabilized, and controlled with a staff of four health physics technicians, two decontamination workers, two operations technicians, and one project manager.

Because waste management and project management systems were available from time-to-time based on ongoing D&D operations at a nearby Battelle site, the West Jefferson site had the ability to perform special projects and waste disposal activities at the discretion of management. The ongoing D&D at the nearby site and the S&M activities at Battelle West Jefferson were funded by the DOE.

The facilities, however, were privately owned and the decommissioning process was being controlled and regulated through Battelle's Decommissioning Plan and the provisions granted by the NRC through Battelle's SNM-7 and Byproduct Materials license. Battelle's field actions were controlled through NRC compliance along with the Code of Federal Regulations as set forth by the EPA, OSHA, NESHAPS, DOT, and other federal



regulatory agencies. A limited number of DOE orders, mostly in the areas of reporting, records control, cost control, quality assurance, and waste management, were invoked.

Major achievements for the success of this project was the establishment of the project's free release criteria and the ability to effectively sort, segregate, and ship waste to multiple waste disposal centers. Without this waste program, the field accomplishments would have been less efficient. The free release criteria also established the accepted level of residual radioactivity allowable at the end of the project. The basic decommissioning process, that is, assessment, radiological control, stabilization, separation of contamination from the facility, monitoring, waste minimization, isolation of the contamination, volume reduction, disposal, restoration of the facility, and closure was used.

With these conditions in mind, Battelle took advantage of the situation with an aggressive approach in their use of S&M funding to allow their present inspection and repair actions to enhance planned D&D activities. This did not require significant additional dollars to be spent upfront, but did require establishment of the D&D objective and a conceptual approach to the D&D of the West Jefferson area. Battelle used this upfront planning through its NRC Decommissioning Plan.

Specifically, Battelle assessed the activities being performed for S&M at the West Jefferson North site. They found that, with some coordination, not only could the systems be checked operationally but the radioactive and hazardous materials on site could be identified, segregated, stabilized, and controlled by the S&M staff.

Also, because waste management and project management systems were available from ongoing D&D operations, Battelle packaged unwanted material during S&M per present waste criteria into appropriate containers for shipment off site. In FY 1995, the West Jefferson North site disposed of 29,800 cubic feet of waste with only three extra decontamination personnel and one extra health physics technician. Battelle could have performed this function without additional personnel, but it would have significantly lowered the total volume achieved. Most of this waste was equipment items just setting around that had passed their useful life. Of the 29,800 cubic feet, 12,000 cubic feet were sent to SEG for smelting, 2000 cubic feet were sent to SEG for compaction, 15,500 cubic feet were sent to Envirocare for bulk disposal, and 300 cubic feet were sent to Hanford for disposal. In addition, numerous items were free released.

The material removal campaign had several long-term benefits. The S&M personnel were better coordinated and more effective with this activity. Because there was always something to pick up, there was always something to do. This essentially eliminated downtime for the worker. The waste costs were significantly less in this time period than in the future, even when inflation was taken into account. With the piles of equipment and material eliminated, there were fewer radiological concerns and less material to perform surveillance on. Then, with the buildings cleaned up of trash, the real task of radioactive D&D could be focused on more productive tasks.

Another benefit of this material removal campaign was more business related. DOE was impressed with the area being put in order and because the total volume to D&D on the site was being reduced, and the schedule and cost of D&D were favorably affected, DOE provided a higher budget for accelerated completion. Furthermore, the radiological risk of cross contamination or an unexplained release was reduced. It also caused most of the contaminated material remaining on site to be placed in a more controlled condition and reduced radiological exposures.

Opportunities for savings continued during S&M by utilizing free release criteria to lower the D&D footprint by upfront monitoring. The action not only lowered the total

D&D scope of work, but drove the lay down areas and D&D support areas onto land that would eventually have to be decontaminated anyway, minimizing cross contamination. Because this release activity could occur in the S&M phase, the systematic planning for the D&D of the site became much more effective.

Other considerations for project improvement utilized the required random surveys and smears in radiological buildings to enhance the characterization of the site prior to D&D. There are computer programs on the market that can identify the location and time that random surveys and smears are taken as part of the S&M program. If materials and areas are cleaned up as part of maintenance or stabilization, this data can also be located. Because the computer can identify the location of this health physics data, it would be able to be used as part of the characterization of the site. Because the radiological conditions of the site during S&M do not change significantly across time, this data will significantly reduce the characterization required.

Although some of these activities may require a slightly higher funding level, mixing some deactivation activities with the S&M phase of the project can significantly reduce the overall cost of the D&D program. If and when D&D funding does become available, the remediation can be better planned with more accurate characterization data. Many actions would have already been implemented in procedures, such as characterization/free release of areas, material handling activities, waste management activities, and continuing inspection activities.

From an operational standpoint, having the right support systems also contribute to an effective D&D operation. In nuclear power plants, personal dosimetry is only required in areas that exceed 100 mrem/year. By using area dosimeters to monitor buildings and rooms, the amount of plant personnel requiring dosimetry can be significantly reduced. Many nuclear plants, such as the one shown in Figure 18.7, have taken advantage of this approach.

Pre-D&D resources can best be used for D&D goals if the objective and end state of the D&D effort is defined upfront. Again, an effective D&D design needs to be balanced between characterization and work definition. An effective design must be tailored or graded into a systems engineering process to allow changing conditions to be effectively handled.



**FIGURE 18.7**  
The Tennessee Valley Authority's Sequoyah nuclear power plant.

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## 18.7 Balancing Changing Conditions through D&D

The art of D&D is to control the changing external and internal conditions while focusing on the objective and goals of the mission. It takes a keen sense to detail, continuous assessment of the hazards created and mitigated by the D&D process, and an understanding of the conditions that must be maintained to accomplish the work in a safe and satisfying manner. One can relate the relationship of D&D and construction to driving a motorcycle and a car.

Both vehicles get you where you want to go, but a motorcycle is much more dynamic and susceptible to the elements. A motorcycle is unstable sitting still (you have to have your feet down). The ride is exhilarating but, without a keen sense of changing conditions, can be very dangerous. It takes the right training, the right vehicle, the right protective equipment, and the right techniques to ride a motorcycle. The mindset must be one of monitoring the machine, the process, the hazards, the changing traffic conditions, and the external circumstances such as the weather. After a hot spell and a brief heavy rain, one must realize that the oils on the pavement may make the road very slick. A car, with its four wheels, sees much less effect of this condition than a motorcycle. If something breaks down on a car, you stop along side of the road. If something breaks down on a motorcycle, you can slide to the side of the road. If you bump something with a car, the damage is to the body of the car. If you bump something with a motorcycle, the damage is to your body. Riding a motorcycle can be a safe and rewarding experience, but only if you keep all your senses about you and anticipate changing conditions. Radioactive work requires the same thought processes in terms of personal protective equipment and attention to detail, as shown in Figure 18.8.

Once it starts, the process requires the activity to be stable. If the process is shut down, it requires some action to stabilize the area prior to walking away. It takes the right equipment, the right training, the right protection, the right techniques, and a keen eye for the changing conditions. You should not only monitor what conditions the process is changing and the new hazards that may come into play, but monitor changing “external” conditions such as training, labor stability, morale, security, site access, materials availability, budget, deterioration of the facilities and structures, stored energy, waste disposal, work priorities, changing regulations, and changing project direction or end state. Performing D&D safely requires a well-coordinated team with a common goal to be observant of changing conditions at all times. This is not always easily achieved when the site is generally undergoing a change in culture from an operating mentality (continuous, repetitive activities that produce product with no apparent end of activities) to a project mentality (starting specific activities that change across time to reach an end state and completion of the work).

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## 18.8 Balancing Changing Cultures

One of the major challenges to transitioning a site to D&D is changing the management and labor culture from an operational business to a directed project. A project not only has a start and an end, but a project also ends the operational culture of the plant. A D&D project has a start, activities that support specific project goals versus ongoing production goals, and a completion that ends the project and the plant organization. D&D is not a modification that enhances and improves plant operations. In many cases, the management of



**FIGURE 18.8**  
D&D work in front of equipment with PPE.

D&D should be similar to Architect Engineering Firms and Labor Unions that move from place to place as the jobs continually finish. The problem with developing a D&D culture is that the operational manager and worker must essentially work himself out of a job, in the hope of going somewhere else.

Although the cultural change is fairly extreme, the process is not as dramatic as it sounds. After all, when the completion of a D&D project is 20 years' off, loss of employment is not of immediate concern. However, this just makes the cultural change harder to achieve. All operations and activities on the site should be focused on the end state or conditions of the site, rather than operations dictating what modifications and activities should occur. From an operational standpoint, having projects drive what operations exist is backward thinking.

The project focuses on worker attitude and the populous thinking. A major problem implementing D&D is the change that management thinking must go through. The operational support activities that management performs are exactly the same, but the reason why these activities are performed is very different. There is no longer a need for financial services, support services, and maintenance on the site to be focused on the ongoing profit and well-being of the plant. Instead, all support activities should be focused on making the plant go away in a safe and cost-effective manner. Because management runs the plant, this change in philosophy is very difficult for them to accept.

Normally, when a commercial business goes away, the plant operations, employment, and culture immediately dies with the layoffs at the site. The physical site is turned over to other organizations (construction and destruction organizations) to eliminate the previous environment liabilities and tear down or reuse the plant. The production culture is physically dislocated from the site and a project culture, one that generates a start, schedule, and finish, is superimposed on only the physical assets of the site.

When a government business leaves a site, they often request that the very people that operated the site perform the D&D of the site. Although these people have the best knowledge, history, and workings of the site, they must convert their mindset from repetitive operations without an end to specific activities that close the site and terminate their employment. Although environmental liabilities can be high, the need for technical personnel to handle unique hazards is great, and D&D takes a long time, the employee must work to a completion objective. The objective is no longer the journey of operations, but the act of finishing or completing the project.

There are still valid reasons for organizations to continue to exist as support, as resource pools, and as knowledge bases, but their size should not be constant across the project. Projects are generally run within an operational organization and environment. Compliance is much easier if the regulations are applied by standard operating procedures and training. Then the tasks to be accomplished can be performed in a safe and consistent manner. Making safety, hazards recognition, and quality a part of how tasks are performed (an operational aspect of a project) can significantly reduce the risk of injury or noncompliance.

Monitoring and characterization organizations are needed only to determine the level of contaminate present. Finance, payroll, administration organizations are only needed for the size of the project. The level of project management is needed only for the standards that the project is being held to. The problem is that these conditions are interpretive and may vary between controlling factions. As mentioned before, the level that support organizations are "required" on a project can constitute up to two-thirds the cost of performing the project.

This is the difference between an operation and a contractual project. An operation receives funding internally from the product it produces and the product's value (in terms of quality, need). A contractual project receives funding externally for the accomplishments it makes and how well it performs these tasks. Performance can be very subjective. Is success getting the field work done or is success measured as having the contractor follow all the requirements and rules per verbatim while getting the field work done? What is the level of risk or liability acceptable to the owner in getting the field work done? These are all important questions in every action that is taken to accomplish the work. As shown in Figure 18.9, even the decision to use one piece of equipment versus another can have a lasting effect on the outcome of a project.

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## 18.9 Balancing Planning and Engineering Processes

Besides balancing the organization and cultures as described above, the success of a project is dependent on balancing a variety of other things. Unlike a steady-state operation, D&D work better resembles construction activities with large staffing and technical expertise changes across the life of the project. There is a significant technical staff associated



**FIGURE 18.9**  
Project manager pondering questions.

with planning and acquisition in the early stages of a project. This is followed by a very significant engineering effort to design the project, define the scope of work, and establish the methodologies used to meet the D&D objectives. As engineering becomes complete, the project management function carries out the engineering along with construction management by bringing on the labor and material necessary to complete the activities. Based on the contractual arrangements, support services for the project are often provided for the project in different ways. Balancing how the staffing is met and transitioned is very important because it represents significant project costs.

The planning effort should be balanced with the execution effort within a project. Most projects use a graded or tailored approach for each of the major parts of the project. Just as the relationship of scope, cost, and schedule must be balanced within a baseline, the amount of planning must be balanced with the amount of return it provides in executing the project. Most of these graded considerations boil down to the amount of detail that is required to complete the project in the most effective and efficient manner. Many times regulations and orders dictate the minimum level of detail required, but interpretation of these same regulations can drive a project to success or failure.

The design engineering effort should also be tailored or graded. In other words, it should be written to account for how the activities should be completed but not so restrictive that changing conditions shut down the project until new designs are generated. When generating the Project Management Plan, considerations should be given as to the levels of resources used. In many cases, the work may be able to be performed through operational procedures versus engineering design. The level of design can affect the requirements of the work package. When required documents are produced and decisions made, they should reflect the most effective way to perform the project based on the data available and its complexity. There are times when work must be stopped and an engineered “work-around” must be produced due to new discovery and a significant change in condition. However, in general, a design should be flexible enough to accommodate conditions that are not exactly as imagined. It is important to recognize when there are workable field solutions and when a redesign is warranted.

The retrieval of operations information, records recovery, information of discovery and historical records are important in establishing initial conditions of the facilities and equipment on sight. However, this information has diminishing returns as the project design, baseline, scope, and project is defined. This information may help the project manager anticipate changing conditions in the D&D process but, once the transition to D&D takes place, proper monitoring should be performed in an iterative manner, allowing the field information to drive each successive activity more than historical data. There is a curve of greatest effectiveness between the effort used to obtain characterization data and the effort of D&D. Depending on the dynamics of changing conditions, these two activities should be balanced. If massive amounts of money are spent for detailed characterization when D&D immediately changes these conditions, labor and resources are essentially wasted. On the other hand, depending on how design-flexible the D&D is, starting D&D without knowing the initial conditions is another way to waste labor and resources, and place the entire project at risk.

The project manager needs to control the balance of effort to create the most effective and efficient project. The project manager balances resources, effort, energy, cost, and schedule. The project manager establishes the graded environment that maintains compliance and meets the objectives of the project, but also balances the requirements with the cost and schedule constraints.

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## 18.10 Tailored (or Graded) Systems Engineering Approach

By the tailored or graded approach, DOE encourages the management of decommissioning programs commensurate with their size, complexity, and level of effort. They encourage the management process to concur within the regulations and orders using the graded approach. At the same time, the management process will have enhanced the ability to perform changing field conditions safely and effectively.

The decommissioning framework in the DOE's Decommissioning Resource Manual of 1994 once documented a desire to account for differences in the complexity, risks, and objectives of decommissioning projects by the application of the tailored or graded approach. This means that the level of analysis, documentation, and action must be commensurate with

1. The relative importance to safety, safeguards, and security
2. The magnitude of any hazard involved
3. The lifecycle stage of the facility
4. The programmatic mission of the facility
5. The particular characteristics of the facility
6. Or any other relevant factor

The graded approach should be considered from the very start of the project. The start point of a project is very important because it establishes not only the initial conditions, but identifies what level of data is needed that must be obtained to perform D&D. The ownership of the facility, when it is available, what is occurring around the area, which support services are available, who is controlling the D&D actions, and who controls the money to perform D&D are just a few of the questions to be answered that establish the



**FIGURE 18.10**  
Typical large equipment demo.

initial conditions of the project. You might notice that none of these questions concerns the technical scope of work but are just as important for a successful project as the contamination levels present, access to the facility, how the facility is constructed, what operating history the facility has, energy requirements, and space available.

One of the most important aspects of a project is to know the project's end point. Establishment of the project's objectives has a significant influence on which contract, engineering, and implementation systems are used and how the activities are set up. Is the project deactivation, decontamination, decommissioning, demolition, or all the above? If the facility is to be re-used, the activities performed and their sequence are very different than if one demolishes the facility.

In the D&D of facilities, it has been found that the process involves a series of radiological and physical considerations demanding a multitude of changes and adjustments in the field actions across the course of the project. In performing decontamination, field conditions change across time because, in most cases, the location and extent of the contamination is only partially known. Even the act of decontamination involves changing the work environment. First, the contamination must be classified, located, identified as to its environmental stability. Next, the environmental stability must change so that the contamination can become mobile and be placed in the desired long-term storage condition. From discovery to the final disposition, field controls and considerations will change. As such, the traditional engineering design/build approach is not as effective as a graded systems engineering approach, which will be described below. In Figure 18.10, note the different protective clothing required and the different conditions that are changing with the work.

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### 18.11 Balancing the Graded Systems Engineering Design Approach

Management processes have been developed to perform changing field conditions safely and effectively. The use of a traditional building construction approach to D&D has a hard time accommodating the lack of site information or the changing radiological and physical



conditions encountered. A traditional construction approach to D&D would utilize design agencies to specify the specific materials, labor, procedures, and activities to be used for the effort. For straightforward construction, where most of the construction conditions are created by the construction activities, this method is most cost-effective. However, under D&D, a detailed design can be ineffective because most of the condition changes are created by the environment and not controlled by the design.

In traditional construction, interfaces with existing conditions are limited, such as soil conditions, area to work in, weather, surrounding activities, and building codes. The design of the project controls quality of the labor and material, schedule, and construction techniques. Design can go so far as to procure long lead items, which tie the construction manager into one way of performing the work. Construction design is generally a straightforward process of material procurement and labor activities that are repetitive from project to project. The design function can easily accommodate what could be called the interface with the environment. These include the size of the footers, amount of strength for the weather, the design life of the facility, the insulation for the winter, the doors and windows for the operation, and occupancy.

In traditional contracting, the terms and conditions and different types of contracting are also much more straightforward. Contracting and design is necessary to control the commitments that must be made with any agreement. However, in D&D contracts, the terms and conditions need to be blended between what is being specified, the objectives of the project, what is being committed to, and the conditions that might be encountered. Unfortunately, in D&D, vital contract information is not always available. All the isotopes might not be known, the contamination is not homogeneous in concentration or in distribution, and the utility services, structural integrity, and hazards of the facility changes as the decommissioning proceeds.

A systems engineering approach to the field activities and the regulatory requirements of the project can overcome these dynamic conditions. If a systems engineering approach is incorporated into a project's management system through the project management plans, it can drive procedures, work packages, instructions, and training. Project Management plans set up the Safety Management programs needed for integration throughout the D&D organization. The plans are interpreted and implemented into a series of procedures that describe *how* the worker performs different aspects of the work. These documents include administrative procedures, operational (health physics) procedures, training procedures, quality procedures, waste management procedures, and D&D procedures. This allows the project to train and qualify personnel in the required actions they will perform and allows the universal hazards in the project to be identified and mitigated.

To describe *what* field actions will occur in a particular area, and to identify unique hazards of the activity, work instructions are generated, taking into account the radiological and industrial conditions present and how those conditions are anticipated to change. Not only are the specific procedures identified that apply but the work instruction provides real-time "checks and balances" approvals for the proposed actions. A part of the work instruction is the safety checklist, which assesses the safety condition of the work area on a daily basis at a planning meeting with the workers. From the work instruction information, a radiation work permit is generated to properly assess, control, and communicate the radiological conditions that the work will be performed in. Because the work instructions and radiation work permits are generated fairly close to the execution of the work, valuable information concerning changing radiological and industrial conditions can be effectively incorporated, documented, and communicated to the worker. Moreover, the approval signatures allow a variety of disciplines to evaluate the detailed work.

The systems engineering approach is a procedural/work instruction integration demanding a high level of communication enhanced by an effective change control process. It serves as the interface and integration of operations into the project direction. Each of the segments in the Management System is designed to enhance the implementation of the work, as well as provide a framework to accommodate changing conditions. This systems engineering approach helps to assure that the regulations and orders are followed. It also provides real-time information about the work environment, thus minimizing extensive use of the change control process. With the proper training and documentation programs, the systems engineering approach is extremely effective and responsive to the changing conditions encountered in the decommissioning process. Transmitting information can only be effective if the skill level of the worker can use it. Otherwise, more specific instructions must be used. Communication and a common understanding of the work are vital in a dynamic environment. If policies and definitions are continuously interpretive, chaos is generated.

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### 18.12 Choosing Design-Flexible Technologies

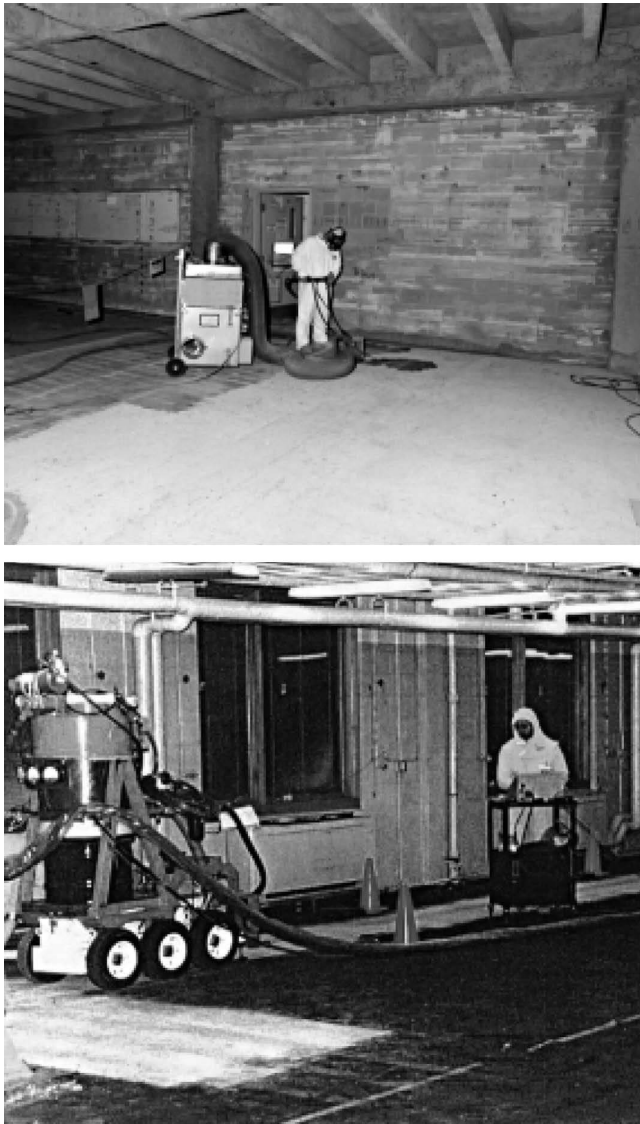
There are many factors to be considered when determining the right technology and methodology to be used for a particular scope of work at a particular site. When a scope of work is contracted out, the overriding factors tend to be what capabilities the contractor has and how these capabilities match with the customer's needs. However, when screening technologies for completing D&D work, there are many more options, factors, and approaches to consider. In developing the best approach, one must recognize the specific risks, issues, constraints, and objectives of each alternative. The best technology/methodology must satisfy a variety of conditions and be design-flexible to comply with a variety of operational requirements and unknowns. It must meet the requirements of compliance, cost, and schedule while enhancing the overall baseline. A good technology for a particular activity can be detrimental if it creates other project problems, such as waste disposal.

In selecting the best technology, several requirements should be considered including radiological control, waste management, the plant's mode of operation, plant capabilities, plant resources, environmental conditions, project risks, site hazards, client requirements, return on investment, project cost, project schedule, contractual arrangements, training requirements, utilities/services available, and how often this technology will be required in the future. An innovative technology screening process should be developed to allow the project to proceed in an organized manner interactively.

A design-flexible approach anticipates changing conditions by the selection of activities and equipment to minimize work stoppage or the need for a new design, new procedures, new equipment, new training, and new activities that would incur additional costs and affect schedule. For instance, the most aggressive or flexible equipment should be identified when specifying a piece of expensive equipment to remove contamination from a wall. If decontaminating a wall was the objective, a "flexible design" thought process could proceed as follows. If washing the wall meets the initial decontamination requirement and condition, a common sprayer is adequate for this particular activity. Due to the potential of changing contamination across the wall, a sprayer that can supply a variety of pressures should be considered. Not only could it be used if the contamination was deeper than originally planned, but if the area needed to be cut out, many high-pressure sprayers could be used for that purpose also. The disadvantage of this equipment is that the area cleaned is reduced

by the square as the needed pressure goes up. Therefore, the use of a vacuum blaster with various blasting media might be considered a better alternative because it could manage the same purpose (except for cutting the area out) without restricting the area cleaned the deeper you have to go to decontaminate the wall. Different attachments and blasting media also provide flexibility in the process. In Figure 18.11, the wall unit is also used as a floor unit.

With a design-flexible approach, different methodologies can be employed even though the same type of equipment is being used. When evaluating methods to perform D&D, it is wise to consider cradle to grave or lifecycle costs versus activity costs. For example, there are two types of vacuum blasters: a carbon dioxide blaster and a grit blaster. The carbon dioxide blaster cuts the surfaces only half as deep as the lightest grit blaster. However, one does



**FIGURE 18.11**  
Workers scabbling floors.

not have the grit to dispose of with the contamination, thus minimizing disposal weight and volume. The carbon dioxide vendors will emphasize the lower product waste volume. However, if one or more grit passes must occur, carbon dioxide grit blasting will take two times as long (incurring greater lifecycle labor costs) and much more PPE (incurring larger overall waste volumes) than grit blasting. Costs would actually be higher for carbon dioxide blasting because waste disposal is by cubic feet and not generally by weight. Both pieces of equipment have their application, it just depends on how flexible the equipment has to be.

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### **18.13 Control of Field Work through Design**

Although a lot of emphasis in the D&D process is given to the preliminary documentation required by the EPA, DOE, and NRC prior to work, detailed design and construction remain the key factors in controlling risks during the remedial actions and D&D process. That is not to say that the preliminary documentation is not required. Detailed design cannot occur without the establishment of the objectives (release criteria, restoration guidelines) and a clear definition of the lifecycle scope. Construction management cannot be performed without the established reporting systems and documentation policies as ways to communicate D&D's changing field environment. To have a controlled D&D construction process, there must be agreement, communications, and understanding among all the participants of what is to be done, where it is to be done, who will be involved, when each activity will occur in relationship to one another, and how much these activities will cost. Along with this understanding, there must be an established mechanism or system to report changes to the process as progression in the field occurs.

DOE, NRC, and the EPA all require this preliminary information be provided, but in different forms. DOE requires a project charter, project plan, and a detailed baseline in the form of a WBS. NRC requires this information in the form of a decommissioning plan according to Regulatory Guide 3.65. To communicate EPA compliance and initial planning, the CERCLA process requires a remedial investigation/feasibility study (RI/FS) and record of decision (ROD), which contain this information. All three agencies require a detailed characterization to help quantify the scope of work. Only a statistically sound characterization can minimize the total project cost by quantifying the existing condition and, thus, maximizing the design effort. A well thought out sampling plan is required for an effective characterization due to the cost of obtaining hazardous constituent information.

Unlike a normal construction process, detailed design for D&D is unique because it cannot describe exact actions that need to take place at any point in time. Even if the conditions within a project site are well established, the D&D work will alter that condition as it decontaminates the site. Structural conditions become less stable, unsuspected contamination is found, fixed contamination becomes removable, health physics conditions change, cost and schedule requires process improvements for higher efficiencies and lower risks, and regulations change. This demands a flexible approach to the construction management process so that changes can be made effectively and the project can continue on schedule. The art of construction management in the D&D process is to select the most effective alternative for the conditions that exist at that point in time in the project. This can be accomplished as long as a flexible detailed design approach is utilized.

The detailed design (design after CD-2 approval, which has been known as Title II) should concentrate on the objectives and establish the scope of work for the project. It should provide

alternatives to handle different situations and yet be able to accommodate the changes needed to allow the remediation actions to proceed. It must also integrate the design of all the supporting functions into the operations performing the decontamination and decommissioning work. This is called “integrated logistic support” and includes the integration of D&D processing facilities, operating procedures, health physics requirements, equipment (enclosures, HVAC), technology development, and design changes. Configuration management and systems engineering should also be integrated into this process.

Detailed design is a complete design of the work, including preparation of all preliminary and final working drawings, specifications, procedures, sketches, and estimates. Definitive design normally includes

- Restudy and redesign work resulting from changes that may be required from the preliminary design
- Development of final (working) drawings and specifications for procurement and construction
- Development of construction, labor, equipment, and material quantity estimates
- Development of detailed estimates of the cost of construction, procurement, and construction schedules, methods of performance, and identification of specific work
- Preparation of analysis of health, safety, environmental, and other project aspects
- Identification of test plans and permit requirements, preparation of a procurement plan, and determination of utility service requirements in coordination with the existing site services
- Other work as required

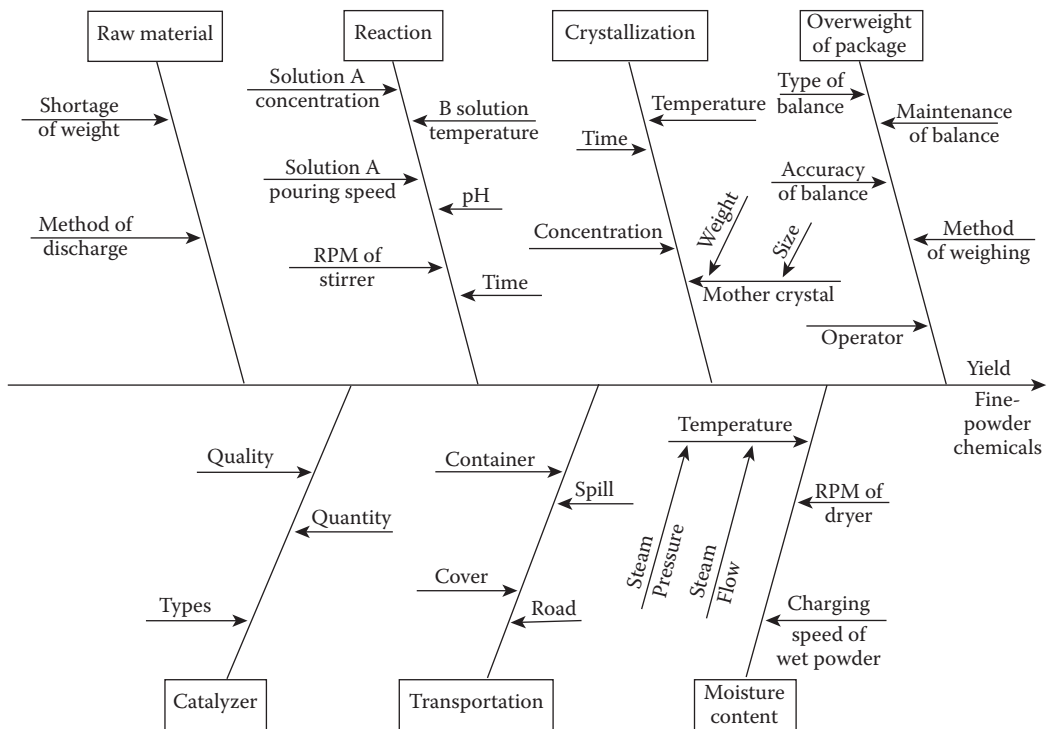
The mechanism to convey what is to be done in the field is commonly referred to as a “work package.” Work packages will be sufficiently complete (job site conditions, hazards present, sketches, drawings, specifications, job descriptions, orders, material lists, a list of required permits, health physics requirements, coordination) to provide the information needed for doing the work in a safe manner. The engineering/design deliverables in the work packages, which are approved for D&D projects, include the following:

- A sequenced scope of work with the necessary procedures to accomplish the specific D&D activities in question:
  - Estimates of labor, equipment, material, and waste quantities
  - Detailed cost estimates
  - Resource-loaded D&D schedules
  - Environmental, safety, and health analysis
  - Temporary (utility) service requirements
  - R&D/Technology development needs
  - Permitting requirements
  - Special instructions (e.g., methods of D&D task performance)
  - Procurement requirements and schedules
  - Final drawings and specifications
  - Detailed procedures

The interface between detailed design and the start of the D&D field activities is the approval of CD-3, and generally involves a major readiness review, or "ORR." The review provides consensus, commitment, and common direction within the entire D&D matrix team. It verifies that the procedures, personnel, and equipment are in a condition to begin the field remediation work (construction). A review at the end of definitive design is specifically performed to

- Ensure that project development and design are proceeding in an orderly manner.
- Ensure that the project will satisfy program and operating objectives.
- Review performance, schedules, and costs.
- Identify potential and real problem areas.
- Initiate action for timely solutions and corrective measures.

A successful review will result in the conclusion that each of the items above has been completed to the extent required to start the physical work on the project. Usually, however, the review generates an "open items list" that must be completed before certain portions of the work can be started. Generally, a management oversight risk tree (MORT), shown in Figure 18.12, or risk analysis is generated for each review and the reviews are "externally" audited for completeness. The acceptance of a graded approach and risk plays a major part in deeming the project is ready to move forward.



**FIGURE 18.12**  
Picture of a MORT or "fish" diagram.

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### 18.14 Establishment of Project Management Systems

D&D project management systems are identical to many construction projects. The Project Management Institute (PMI), as well as off-the-shelf books, can teach, explain, calculate, and provide an understanding of the baseline mechanics used in today's construction and D&D projects. The DOE has DOE Order 413.3A describing the planning process used in the government. There used to be DOE Order 4700 that basically explained what PMI teaches and certifies. The project has different requirements depending on whether it works under RCRA, NEPA, TSCA, or CERCLA, or a combination of these. In all cases, baselines involve the scope of work to be done (Scope), the timeframe the work is to be done in (Schedule), and the cost to complete the work (Cost). It is generally a living document that maintains the status of a project and the contractor's performance through formal documentation when compared to the initial contractual requirements and baseline.

Baselines can be used as planning tools, design definition, oversight tools, common communication of the project, project control tools, reporting tools, performance measurement tools, and so on. It is important to know what the baseline is being used for, and who controls it, because it can be manipulated in different ways across the life of a project. This is no more apparent than when one looks at DOE Order 413.3A and the old DOE Order 4700. DOE Order 413.3A emphasizes actions prior to design in its approval process of CD-0 and CD-2, reflecting the "range of estimate" baseline generated. The completion of this conceptual baseline provides for a general approach and recognition of the initial risks in the project. DOE Order 4700 emphasized its ability to enhance oversight of the construction activities (now CD-3 and 4). This baseline, following expansion and design of the previous baseline, has enough information in it to validate the project being done. In general, baselines were generated and controlled by technical personnel charged with accomplishing the work. Presently, a baseline may be controlled by the actions of the contract.

Baselines are used by different organizations for different purposes, and yet they remain a common document from which each organization's information can be obtained. Some organizations that use baselines are upper management for reporting, marketing personnel, proposal personnel, human resources personnel, budget personnel, procurement personnel, oversight personnel, management approval, technical verification personnel, audit personnel, technical contractors and subcontractors, project management personnel, engineers, construction management personnel, work planning personnel, and the worker. All of these organizations are linked to the baseline and use it as a communication tool with the other organizations in different ways. The baseline (Figure 18.5) is the common project communication tool and yet it must react to other organizations' control, such as in a contractual environment.

The next section will attempt to describe what a baseline should be and how it should be used from a project manager's/engineer's perspective. Project managers think of the baseline as a planning, design, construction, and reporting tool. Most importantly, however, they think of the baseline as a day-to-day reflection of the project from which reporting data, performance data, and cost/budget control data are derived. It not only identifies what needs to be done, when it needs to be done, and how much it will cost, but the estimate itself is a project commitment to accomplish this work, in this way, within this schedule and cost. The baseline itself, with its depiction of a project's activities and logic, establishes project scope, schedule and cost using dollars as a common denominator (not simply as a budget number). It describes the activities, labor, and material needed to perform the project in the most effective and efficient manner under known and unknown

constraints. It should establish the commitment and bond between the construction (D&D) contractor and management/oversight on how the project will be performed. A common phrase is "Plan the Work and Work the Plan."

From a project management perspective, it is extremely important that a fully developed baseline and its activities involving scope, schedule, and cost be generated from three distinct "passes" or analysis of the activities. This focuses the scope, schedule, and cost into one baseline performance and objective, taking into account how these factors are affected under various conditions. The balance and understanding of cost, schedule, and scope required in a fully developed baseline are essential for the baseline to be effective in the field.

The first pass, Scope/Schedule/Cost, generates the "scope baseline" concentrating on "what" needs to be done or "the scope of the work." This phase is really a design engineering function that documents what activities need to be performed, the labor and material required to perform each activity, how the activities are assumed to be carried out with what equipment, the strategy of when the activities should occur, establishment of the physical constraints that effect the project's execution, who in the construction organization is going to carry these activities out, and so on. It also quantifies the level-of-effort (LOE) support functions and organizations required for success such as training, general administrative, overhead, records, reporting, documentation, safety and health, security, emergency management, infrastructure, utilities, equipment management, records management, procurement, and the negotiation of fee and profit.

This "scope" baseline can be quite specific but is generally generated from "the top down." That is to say, the engineer looks at what is to be accomplished and tries to include all those activities required to accomplish that task. Much of the design of the baseline is tailored to what exists on hand and how the companies are organized and contracted. This baseline is sometimes also called a strategic baseline because it establishes risks, risk mitigations, and how the project will generally be executed. The generation of this baseline is also a "design engineering" level document. The engineer uses common activities, historical costs and resources, and even standard labor and material costs (root means data) to determine what it will cost and how long it will take to perform a specific amount of work.

At the end of this first process, all parties should have a clear understanding of *what* needs to be performed to accomplish the project in terms of resources and activities. All parties should also have an understanding of *how* the project will be performed under which identified constraints (Regulations). The "scope" baseline should define the project in general.

However, there are some limiting characteristics the scope baseline has, including a lack of definition for specific activities (activities are described at a higher level than field activities), a lack of construction commitment to perform the activities as documented, a lack of defined logic that substantiates the completion date, a lack of definition in how the support organizations need to change across the project, the lack of a critical path, the lack of understanding of field constraints, and an inability to apply the baseline activities to field construction work. However, the scope baseline is important because it establishes resources and materials needed; allows a starting point to develop procedures, training requirements, work packages, and instructions; and gives construction management a starting point to plan the job.

The second pass, Cost/Schedule/Scope, derives a "schedule" baseline from the scope baseline and establishes an optimum schedule for the project. This employs a systems engineering approach that documents the logic of when the activities can be performed, the labor and material that can be applied to each activity, how the activities are assumed



to be carried out with what equipment, the strategy of when the activities should occur, establishment of the physical constraints that effect the project's execution, and who in the construction organization is going to carry these activities out. Establishment of the project's logic as it relates to float, sequencing, and activity interrelationships as well as the establishment of the critical path is the goal of the schedule baseline.

The schedule baseline is also quite specific and is generated from "the bottom up." That is to say, the systems engineer uses the scope baseline and refines the activity to establish the starting conditions in the field for the most efficient construction activities. This exercise takes the tailored design approach and customizes it to only what is needed to accomplish that task. It establishes the changing LOE needed by the support organizations across the life of the project. For instance, if health physics and security are support functions, their activities should change as D&D reduces the hazards in these areas. Overhead and administrative activities change as the population, contractual subcontracting, or streamlining functions change.

The schedule baseline represents a construction engineering level document, and coordination or knowledge of construction activities within this baseline is essential. The construction engineer uses existing site conditions to determine when each activity can occur. The logic of the baseline must not only take into account the effect the activities have on one another but must also display how the sequence of activities recognize hazards, mitigate risks, accommodate weather, and recognize changing conditions, regulations, and objectives. At the same time, the schedule baseline should demonstrate a logic that takes advantage of opportunities to lower cost and reduce schedule by utilizing day-to-day activities.

At the end of this second process, all parties should have a clear understanding of *when* and *how* the project will be performed by the construction contractor. The construction contractor should have a complete grasp of the work at hand, commitment to the project's logic, and an understanding of the flexibility within the day-to-day activities in the project. Upon completion of the schedule baseline, there should be a full understanding of the project's critical path. By the completion of this second pass, all contractual arrangements exist and the ownership to perform the work at hand will lie with the construction manager. The definition of the work is complete at the lowest level and communicating with the worker can be made through training, work packages, instructions, daily meetings, and supervision.

Although activities management is in construction, the project manager has major tasks to perform, including baseline management. The characteristics the schedule baseline may have to overcome include balancing the baseline with the changing yearly budget and limiting the schedule due to other current labor, material, or cost constraints. This leads the project manager to subject the schedule baseline to a third and final pass, Scope/Cost/Schedule, prior to validation to generate a "cost" or "finished" baseline in line with physical and external constraints.

This phase is a nontechnical approach that adjusts the baseline to head count, material acquisition, or budget constraints. It allows management to make the final approval decisions as to what the labor effort will be and what money is going to be spent when. It also allows the project manager to fully understand how the baseline data and the performance of the project are going to be judged. Baselines usually report through the Earned Value Management System. Again, the mechanics of this system can be understood through various academic sources. However, how this data is interpreted and judged tends to vary from project to project. The generation of the "final" baseline is an attempt for the baseline data to accurately display how the project will be performed. That way, changes and corrections can be made prior to it becoming a significant problem. Because management

judges its health on staffing, budget control, profit, money flow, and other business factors, the final baseline must bring the work at hand into prospective with these staff and support functions. Configuration management, quality control, authorization basis, health and safety all must be balanced and in concert with the project at all times. The procedures and support processes are many times applied and adapted to how the project is going to be worked in this last adjustment to the project baseline. Safety and how it is implemented in the field is always an area of interpretation. The ultimate goal is the accomplishment of work without injury.

In conclusion, baseline management is a valuable communication and control tool to establish and monitor the status of a project. Without lecturing on the mechanics of baseline management, this section has tried to describe the process of generating a functional project management system for D&D. The importance of a baseline is to document how the work was and is being done; status of the project in terms of cost, scope, and schedule; and demonstrate the most effective and efficient method to complete the project. This can only be done with definition, structure, consistent interpretations, and communication. Many times “project controls” dictate the project’s performance baseline control and reporting techniques. This may let the project be viewed as a success but loses the connection with how the project is meeting commitments in the field.

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### 18.15 Control Provided by Project Management

Because detailed design establishes the scope of work as described and construction management performs that work as described, one might question the role of project management. Project management provides an overview of the role of communications and reporting, which allows people, agencies, and management a central focal point for interfacing. The project management function provides, among other things, the interface between the field activities and the requirements of management. Areas of integration include providing progress, technical changes, cost and schedule, operations, quality control, and project control. A prime factor in project management is to maintain baseline integrity throughout the project; that is, to complete the project successfully on time and within budget. The use of the Earned Value Management System as a reporting tool for baseline management is an important and recognized project management tool in reporting. Baseline management is a means of controlling a project, from design to completion, and ensuring close adherence to the plans (technical, cost, and schedule) by accommodating scope changes while providing control on the project. Proper baseline management concentrates on the project management disciplines listed below.

- Systems engineering, testing, and evaluation, which covers the engineering procedures of turning technical requirements into completed projects.
- Configuration management and change control, which establishes the procedures for handling technical changes and identifies, records, and reports the technical interface.
- Contract support and management, which strives to have the contract reflect the baseline while reflecting the work going on. This brings the areas of budget status, scope status, contractor direction, and negotiated commitments into play.

- Quality assurance, which emphasizes the procedural aspects of quality from design through execution.
- Construction management, which details the management aspects of performing the D&D activities.
- Reporting and control performance, which allows those executing the project to record and report the project's progress.
- Project termination, which deals with the procedure for terminating a project before its planned conclusion.

Baseline management, through project management teams and brainstorming, can be very effective. One important aspect is evaluation of selected technologies to compare, prioritize, weigh, or score an established set of criteria so that they can be compared to one another. How the technology meets the project baseline's objectives and end points are extremely important. Other indicators include acceptance of the operation, return on investment, which technologies provide the greatest cost savings, and which technology adapts to multiple scopes of work. The technology screening process is generally performed informally because it is more important to move forward with a good technology than "study" the screening process till "the best" technology is determined.

Risk is an important aspect of baseline management. The decision to use one technology or technologies and designs must be well researched and tested. However, these decisions can have a dramatic effect on both the schedule and the cost of the project. Risk can also establish the priorities of performing one task over another, thus controlling the work environment.

The managerial evaluation process performed by project managers has been given many names over the years but represents a Kepner–Tregoe technique of project management, along with other tools, including Monte Carlo. The technique is a systematic process that prioritizes criteria used to judge technology/methodology approaches (criteria phase), find the technologies that meet the criteria (brainstorming phase), and compare the technologies against one another to determine their relative effectiveness (analysis phase). It takes into account what has happened in the past, what could happen in the future, and the probability that assures the best path forward. It helps develop the implementation of the right system for the right project.

Other project management systems include planning for end points of a project with passive versus active solutions to minimize large legacy costs. One example would be to plant trees in strategic locations to reduce HVAC requirements for temperature controlled buildings. Another example specific to D&D is to install a series of shallow ponds on a property to slow runoff for monitoring purposes in case a cleaned site has future contamination issues or to slow and/or concentrate the movement of contamination should it occur.

In conclusion, the project management systems can create a flexible but effective means of integrating changing conditions in D&D with field activities. D&D, in particular, requires a flexible system because the conditions of the work being performed continually change. However, the contract must legally reflect stability. An effective change control system is also required for the success of a D&D system. Only then can detailed design, construction management, and project management interface effectively to achieve the D&D project objectives through the legal commitment of the contractor.

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## 18.16 Control of Field Work: Integrated Safety Management System (ISMS) and Other Tools

The ISMS is an excellent model to establish a safe approach to D&D. With this system, management can control the environment the worker works in and the worker can control the work that occurs. ISMS can create the environment to achieve safe, compliant closure of the project for the worker and the public. The following elements of the system are integrated into one process:

1. Communicate the objective and the program philosophy.
2. Define the task and its objective (including the waste form).
3. Establish requirements for compliance in the performance of the task.
4. Find the right worker for the task.
5. Train the worker so that the worker is qualified for the task at hand.
6. Recognize the environment the worker is working in.
7. Recognize the radiological hazards within the environment.
8. Recognize the chemical hazards within the environment.
9. Recognize the industrial hazards within the environment.
10. Provide the documentation and instructions that take the above into account.
11. Empower the worker to communicate concerns, experience, and knowledge.
12. Document the communication and consensus that has occurred with the worker.
13. Provide the right personal protective equipment (radiological and industrial) for the job.
14. Provide the right equipment to perform the task.
15. Provide the tools (preferably the most effective) for the task.
16. Empower the worker to perform the task.
17. Receive feedback for future work to improve performance.

For example, assume that a contract can control the integration of its own activities such that the accomplishment of the project can be performed in a safe compliant way in the best interests of the DOE financially. This implies that all CFR, OSHA, EPA, DOE, NRC, and WAC requirements are performed in compliance with the criteria above, but with a fully knowledgeable, integrated, and graded approach. For instance, safety vests should be worn at an appropriate time, not by every person on the project. Work involving a ladder should be minimized, not forbidden. Those entering radiation areas should require a dosimeter, not everyone on site. The existence of some radioactive contamination should not necessarily make an entire building structure slated for radioactive disposal. These examples, as well as many others, can cause many dollars to be saved. If this is a cost-plus contract award, this way of doing business would not necessarily matter to the contractor unless contractual incentives are in place. However, it should be of concern to the customer. One must be able to employ a graded approach and perform the work based on the physical and environmental conditions that exist. Otherwise, lots of activities occur, but few activities complete the project objective or end state.

In the best interests to the customer, certain actions should take place to isolate the project site and place it in a safe condition so that the project can most effectively accomplish the work at hand. These tasks include

- Reroute and isolate facilities from the project site
  - Electrical
  - Domestic water
  - Natural gas
- Establish control of the project site and its lay down areas
  - Minimize the radiological controlled areas
- Isolate the utilities and services in facilities to be remediated
- Remove personnel and materials out of areas to be decontaminated and demolished
- Establish the true radiological conditions of the facilities so that the right methods of demolition can be applied

In the middle of all these systems and methodologies, communication at all levels of the organization is vital. Without effective communication to the worker, as shown in Figure 18.13, none of these systems work. The worker is really the only one that actually performs the field work. His environment and his ability to perform the work safely have a big effect on the progress of the work.

Besides ISMS, there are other tools controlling field activities. There is an organization that concentrates on best management practices ([www.BMPCOE.org](http://www.BMPCOE.org)). It sets standards of estimating to improve consistency and accuracy in estimating projects. Along with its estimating capabilities, it also establishes methods or principles to sequence activities that make the estimates most effective. It does not concentrate on the activities themselves, but



**FIGURE 18.13**

Communication with project manager, construction foreman, and worker.

on the principles the systems engineering process should use to make a project safe and successful. These principles include ways to find the right level of characterization activities to remediation activities depending on the homogeneousness of the conditions, establishing what conditions determine whether cleanup proceeds from least contaminated to most contaminated or *vice versa*, and how to report the project to produce the most accurate picture of the project's status.

Another useful tool in the quest to produce a successful sequence of activities is the Lessons Learned programs. Lessons Learned can be extremely useful when performing projects that correspond to past projects. The previous project may identify the best methods and tools to consider, unforeseen assumptions to consider, an insight into possible changes of conditions your project may experience, or the major concerns the regulators or the public may have when performing your project. The disadvantage of Lessons Learned is that they only provide what should be considered but should not be depended on to necessarily identify how the project should be performed. Often a project assumes that the same type of project requires the same actions as previous projects. This leads to problems because every project exists in different environments, different starting conditions, and different operational status. Because the new project is in a unique condition that differs from previous projects, Lessons Learned are only partially applicable after evaluation.

This is exceedingly true of the environmental management (EM) projects being performed by the DOE. Some projects were systematically shut down and transitioned into D&D while others were simply walked away from and mothballed for 25 years. These actions make a large difference in the deterioration of the facilities, the hazards left in the facilities, the actions that need to be taken during the remediation, and the way the project should be approached.

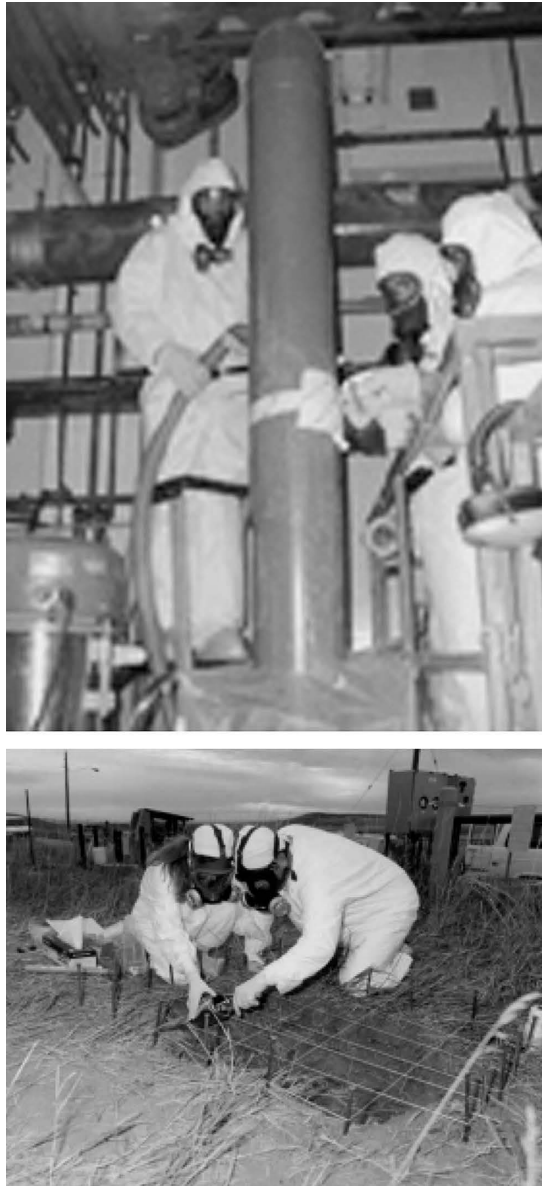
The use of an effective risk management plan and risk analysis throughout the project is useful to controlling the scope of a D&D project. Risk changes across time as the project is performed and more information is obtained. In general, risk should go down as the project is completed but changes in conditions that are discovered during execution can significantly increase risk for a period of time. DOE has attributed risk, in terms of dollars, to the management reserve and contingency on a project. A good, well-maintained risk management plan is invaluable in anticipating changing conditions and effectively mitigating risk.

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### 18.17 Balancing Worker Selection

Determining the skill level of the worker being used will affect the quality, schedule, and cost of a project. The worker's skill level is directly related to who the worker is. Skilled craft requires less training because there is some dependence on "skill of the craft." Skilled craft can generally perform the task quicker using a wider variety of tools and methods. Utilizing skilled craft creates a safer environment because they recognize changed conditions and hazards more readily and have methods to mitigate those conditions within their skill set. However, skilled craft are also more expensive per hour than common laborers. On the other hand, laborers require more training, supervision, and may take longer to complete the tasks even though they also cost less per hour. Depending on the complexity of the work package, how long the activity takes for

each skill level to complete, and the amount of training and the oversight required, the skilled craft could be cheaper for certain activities than using laborers. These decisions are generally made in the design, strategy, and planning functions of the project and will affect how the required quality and regulatory commitments are accomplished. Figure 18.14 illustrates coordination between skilled craft and other labor support in a D&D setting.



**FIGURE 18.14**  
Coordination of skilled craft and other labor support.

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### 18.18 Balancing the Execution Approach

There are several factors influencing the execution approach to D&D. The amount of the project being performed by subcontractors and the amount being performed by in-house services should be decided as early as possible. This is often synonymous with how many of the activities are operations based and how many of the activities are project based. Depending on the activity, each approach has its advantages and disadvantages. Compliance of regulations, safety methodology, meeting quality, having proper configuration management, and complying with other standards are examples where a good procedural process allows the worker to stay in the right operational mode. However, there can be many support organizations embedded in the operational mode that a specific project does not need during the entire project schedule. One should strive to perform specific activities to accomplish an end purpose while maintaining safe conditions in a project type mentality. The two operational and project philosophies should be synchronized at the work package level. The two philosophies could cost considerably different on a lifecycle basis if the activities and support are not focused on the project mission rather than sustaining plant functions.

Three areas where the operations and project philosophies must be integrated early in the program include the waste processing, health physics, and security areas. Some organizations actually call D&D projects waste management programs because the requirements for the waste disposal and disposal site waste acceptance criteria (WAC) drive the endpoint of the D&D activities. Others consider D&D projects health physics and authorization basis programs the driver because the radiological manual requirements consistently cause the project to perform D&D in a safety, quality, and configured manner but continue to change across the project as the conditions change. These drivers influence the performance of work is how the regulatory requirements are met but can be frustrating if the changing conditions are not recognized within the regulatory framework. During the course of D&D, the project will have to interface and recognize such drivers as permits, points of emission, notifications, authorization basis, and environmental impact. The project has different requirements if worked under RCRA, NEPA, TSCA, or CERCLA, or a combination thereof. Security also places similar demands on the project, depending on the changing vulnerability of the site. The D&D project manager must focus on the organization's requirements and not on the organization's "way of meeting the requirements." The art of integrating activities to meet the project's mission at lowest cost and schedule is not the same as allowing one organization to dictate how activities are performed to their specific advantage or to requirements that are not in line with the conditions.

Project performance has a lot to do with a variety of factors such as organizational interfaces, being in compliance (not necessarily being perfect) in operations, balancing requirements (sometimes having to change procedures), and focusing on the project's completion within the constraints identified (be they orders, regulations, directives, process requirements, safety, money, schedule, configuration, or quality). Every decision, to perform something one way rather than another, represents some level of risk. Identification of risk, risk management, and risk mitigation are important processes to assure the project can be performed safely in an effective manner. In each phase of a project, risk should be periodically assessed and mitigated to make sure the project is not inadvertently accepting too much risk. As the project nears its completion, the remaining activities become more defined, the workers are more experienced, and the hazards are lowered. The risk, management reserve, and contingency required should become less, both in the amount and percentage of money remaining.



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### **18.19 Planning Conclusion: The Effect of the D&D Planning and Design Approach**

By balancing the previous approaches, as well as others, in the design and planning phases of a project, project management can have a significant effect on the project's success. Many people are involved in setting up the project for success. The work really involves controlling external influences and establishing the compliance, configuration, quality, and control of a project in the processes being used.

Performance of the project objectives will be in a continuum from "not met" to "fully met" interpreted by a variety of people. This is why the establishment of the compliance standards, minimum standards, and graded approaches must be agreed upon to drive the actions of the project. Workers vary in their training, abilities, skills, recognition of hazards, and willingness to comply. Interpretations of orders, directives, standards, and regulations, if not taken into account, can stop a project completely. Not recognizing the working conditions or not having a recommended path forward, direction, or strategy is fatal to a project.

A project manager's job is to compensate for the existing conditions; provide continuity, direction, and strategy; recognize and accommodate changing conditions; encourage communications in the planning phase and in all phases thereafter; and maintain an environment that allows the project to move forward. One of a project manager's major functions is to establish and maintain baselines and to provide leadership to hold all the changing conditions of a project in perspective so that the project workers can progress in a well-coordinated manner.

The technical execution of the project is straightforward once the baseline activities are properly identified, scheduled, resource loaded, and leveled. If planning the work is completed in detail, the only thing left is to work the plan and report on the progress in a consistent manner. Normal reporting utilizes an earned value management system identified under the project management section.

The design and baseline effort of a project including changing cultures, establishing indirect costs, accommodating changing conditions, performing design, and setting up workable systems and technologies is over 60–70% of the project cost. The remaining 30% effort involves executing the project. The following explores how the project execution can save money and schedule as the activities are accomplished.

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### **18.20 Control of Field Work through Construction Management**

Construction management takes the detailed design procedures and manages the field activities to accomplish the D&D design objectives. It entails all the management aspects of performing the D&D field activities, including (1) design and procedural control (inspection), (2) change control, (3) coordination, and (4) reporting.

Design and procedural control (inspection) monitors the day-to-day activities in the field and makes sure the project is performed according to the approved design and procedures being used. For each activity, inspection observes the safe operation from an industrial and health physics standpoint utilizing the quality of materials and workmanship consistent with the project requirements. Because hazards emphasis is focused on the radiological aspects of the work through the Health Physics department, and focused on the chemical aspects of the job through the EPA and environmental constraints,

a major pitfall in D&D projects could be allowing the industrial hazards of a project to be over-shadowed and forgotten. Development of structural problems should be of particular concern in D&D because structural material is generally being removed during decontamination.

Control of field actions should also be influenced by the day-to-day actions the construction manager is trying to promote. One of the manager's objectives during field actions is to increase efficiency and effectiveness of the worker's activities by continuous improvement. An objective of the project should be to finish the end points of the project with passive versus active solutions to minimize future maintenance or legacy management. Operational improvements and coordination can influence how quickly the activities can be achieved and, consequently, lower the costs required to complete the work. This is illustrated in Figure 18.15, depicting facilities and soils remediation.

For example, during a soils remediation project, the amount of soil removed depended on real time analytical monitoring and the coordination of excavation equipment. By



**FIGURE 18.15**

Typical coordination for both facilities and soils remediation.

coordinating the movement of waste boxes in the transfer zone (increasing the zone slightly) and changing the break times and shift times of the monitoring crew, twice the amount of soils could be removed at a cost of two hours' overtime by four people on the monitoring crew. Just having the right level of personnel at the right time performing in concert made significant progress in achieving the work. One project was completed a year ahead of schedule and \$3 million under budget by utilizing this methodology. This, again, is an excellent example of a systems engineering approach.

Changes in field actions should be documented in the baseline control system. Change control is the process of documenting the changes that need to occur in the project. It involves an established change control system in which changes to the project are submitted through Change Requests for approval by the D&D project team. The goal of change control is to have the construction contract, the baseline, and the field activities all reflect the same scope of work. Construction Management submits change requests if they meet at least one of the following reasons:

*Functional Necessity.* The change is necessary for the decommissioning task function to be performed.

*Safety.* The change is needed to meet changing safety conditions and requirements.

*Marked Improvement.* The change is a significantly better way to meet planned decommissioning objectives.

The project must be flexible to allow different levels of changes to occur by different organizations. If there are only minor ways of accomplishing the work, or if there is no cost, schedule, or safety changes on the conditions, the field should be able to make field changes. If outside design is required, the procedure and work package changes should dictate, and if outside influences (major scope, schedule, costs changes, budget changes, regulatory changes, and requirements, etc.) dictate changes, the contract needs to dictate the change in the operation. The detailed steps to be taken in processing and approving change requests should follow a graded approach and be documented in the change control procedures.

Control of field actions requires coordination with the different organizations. Coordination is the interfacing of the design requirements with the operations available to perform the design objectives. It is the process of using the right tools to accomplish the design objectives most efficiently. In the interrelationship between the construction management function and the detailed design function, care must be taken so that each design requirement is performed with minimal risk to personnel or the environment. Construction management must continually assure that the activity is worked in a stable environment, that the process of decontamination is in a controlled manner and contained, that the process is effective, and that the waste is processed in a controlled and acceptable mode of operation to be accepted for shipment.

The day-to-day coordination of a project is where the act of configuration management is most evident. The objectives of configuration management include

- Achieving, at the lowest cost, the required system performance and operational efficiency
- Promoting the maximum degree of design and development latitude, yet providing the appropriate degree and depth of configuration control
- Obtaining maximum efficiency in the management configuration changes with respect to their necessity, benefit, cost, timing, and implementation

- Providing a systematic review of all changes to ensure that all primary and secondary effects of the proposed changes are identified and their costs/benefits are weighed in making a decision to incorporate the change in Lessons Learned
- Providing an audit trail of the project from conception to closeout including status, changes, and Lessons Learned

Again, we find that configuration management and change control are required for an effective construction management function.

One major requirement discussed concerning construction management is the effective communication of the project status to management through established reporting systems, such as the earned value management system. Without these systems, the overall control and coordination of the project can be in jeopardy. Modifications, progress, control, and achievement of the project objectives cannot occur without effective communications through these reporting systems. Many feel concentration on the field work is what construction management should concern itself with but effective reporting is crucial for the overall success of the program. Without it, the project may be accomplished, but funding for any more work will never come around.

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### 18.21 Balancing Field Execution Processes

The key to improving the efficiency of field day-to-day activities is concentrating on meeting the requirements versus performing the work with one specific procedure or another. During the execution of the project, there are techniques that can be considered to lower the labor and material effort and still meet the requirements of the functions. The trick is to concentrate the action on the requirements while minimizing the support costs to perform the effort. Even the effort required in introducing people into and out of the work area must be performed in a coordinated and effective manner to minimize effort and use working time wisely. Dosimetry is an area where innovative thinking may reduce the effort normally required with the traditional methods and thinking. Traditionally, the dosimetry organization placed TLDs on all plant personnel because there were specific areas on site that required TLDs. However, utility plants found that (1) a considerable part of the population never entered these areas and (2) the areas might not be accurately identified. The dosimetry effort was reduced by >50% by defining who needed dosimetry as well as where and when the dosimetry was required. Here is how it worked. The public can encounter up to 100 mrem a year of industrial radiological exposure without monitoring. As areas in the plant were posted with hanging dosimeters, it was found that the vast majority of the plant received <100 mrem and the exposure was limited to the individual's working hours. If maximum working hours were taken into account with where people presided, there were very few employees that actually required dosimetry and access into limited areas. For those workers that required dosimetry, the plant setup goals for these workers that simply added 100 mrem of exposure to their readings so that a conservative approach for exposure could be taken. This significantly minimized the amount of dosimetry required on site.

If this concept was applied to respiratory services, the number of jobs and employees that had to use respirators would also go down significantly. However, this application has its downfalls because employees may demand dosimetry and PPE for perceived exposure when

the requirements say it is not required. It is hard to change the radiological culture in a previously operating plant where previously higher levels of dosimetry and PPE were required than that necessary by regulation. Another trade-off that should be analyzed, depending on the job requirements, is whether to use disposable (Tyvek) PPE or laundry services. Each type of PPE has its advantages and disadvantages but, when making the cost analysis between the two materials, remember to take into account the cradle to grave costs (i.e., collection and maintenance for laundry services, packaging and disposal costs of Tyvek).

An important process to be performed in D&D activities for work to be performed efficiently and safely is the recognition of hazards. With the conditions of D&D changing so drastically, the worker and the field organization must be trained to continuously look for the changing hazards and determine acceptable ways to mitigate those hazards. Every day, before work begins, the worker should assess the area he is working in and identify potential hazards or any changes in the area because he was there last.

Another good approach is to work from the “outside-in,” or from “least to most” contaminated. The worker can become more proficient in less hazards areas. The contamination can be approached in the most effective manner, and the response infrastructure can become more mature across time. The disadvantage is that the source is not contained as soon and the worker risk remains high longer. In Figure 18.16, by removing noncontaminated asbestos first, the worker gets used to PPE and safe work practices and the amount of radioactive disposal is minimized. The disadvantage is that other containment may have to be erected.

Some conditions always change. Weather, the organization of hoses and wires, the workers’ physical and emotional fitness, time of the day, and who is working together are factors that are external and “not in the worker’s control.” Some conditions should always be checked. Underground and overhead lines, obstructions of the equipment and tools, training and proficiency to perform the job, PPE being used, understanding of the step-by-step procedures and instructions, and the workers actions are examples of conditions the worker can control. The worker should also anticipate conditions. The worker must project his actions forward to anticipate what changing conditions and hazards he may be creating. It is one thing to take down a wall. It is a mistake to have people on both sides of the wall not talking to each other.



**FIGURE 18.16**

Decontamination workers fully dressed removing asbestos materials.

Even the smallest incidences can show the lack of recognition. I was once giving a worker safety orientation when a bee flew into the room through an open window. It landed and buzzed on the window pane next to the opening trying to get out. I asked everybody to move out of the way but one guy said "I'm going to kill it" and took it upon himself to swat at it with his hand, he broke the window, and cut himself on the glass. The bee flew out the window. His intentions were good, but he displayed a complete lack of hazards recognition in his actions. The bee was not being aggressive. He did not assess the situation properly. He was not trained. He did not follow directions. He did not recognize the hazards. He did not use the right equipment. He did not reach a safe mitigation of the situation, and he got hurt by not thinking before he took action. Generally, initial constraint is better than rash action.

With changing conditions, it is often field activities that can best determine when cleaning has reached adequate levels to stabilize the area for demolition. The decision to stabilize, demolish, and dispose of material versus continuing to remove contaminate is an important one and should be approached based on safety, project cost, and environmental concerns. Cleaning is often labor intensive and costly. The main reason it is being performed is to relieve environmental concerns. At the same time, the level of contaminate dictates the type of packaging, transportation, waste management, and disposal. In some cases, cleaning can concentrate contamination to the extent that it creates a more expensive disposal pathway (low level waste during cleaning could become TRU waste if concentrated enough). As with all project scenarios, the cleaning/stabilization action must be thought out from a holistic (cradle to grave) approach and determination of when to convert efforts from decontamination to stabilization should be established and planned.

Another example of planning is in the deactivation of a facility. The removal of contamination can reduce worker hazard and change the worker's requirements as the facility becomes cleaner. Early deactivation of a facility helps define the D&D required. However, continued deactivation has diminishing returns in "step-wise" functions. Once removal and cleaning levels are reached that allow D&D to be performed effectively, further deactivation will cause higher costs in the project for no value added. These actions can actually complicate D&D in the areas of mobile equipment needs, security issues, lack of structural integrity, and so on. Allowing a facility to go "cold and dark" too early causes accelerated deterioration or shuts down support facilities, which could complicate the D&D process. Temporary facilities, as shown in the background of Figure 18.17, can be a solution to providing facilities and allowing the D&D facilities to go cold and dark.

There is an inherent safety advantage in introducing temporary services, facilities, and utilities versus using the facility systems at hand. Generally, the existing systems are not well documented, somewhat deteriorated, significantly oversized, and inefficient for the D&D actions required. The facility systems have usually gone through modifications not shown on the original building drawings and the status of these older systems are not well known. The chances for these old existing systems to fail or for D&D actions to put the worker or the environment at risk are significant. The use of mobile facilities and utilities allow the project to know where all the active energy sources are and to properly protect the worker from inadvertent exposure to hazards.

In the D&D planning process, there are scenarios that encourage the D&D project to work the D&D activities from "dirty to clean" or from "clean to dirty." Dirty to clean allows the project to remove the most hazardous conditions first but clean to dirty allows for training and experience before the most hazardous work. Many times a combination of the two are best: clean to dirty allows the teams to become proficient, dirty to clean to remove the source most expeditiously and minimize tracking; clean to dirty to systematically minimize the footprint of the contamination, dirty to clean to minimize the worker's hazards.



**FIGURE 18.17**  
Typical structures for mobile lab or shower facilities.

In a similar decision analysis, timing of when it is prudent to stop cleaning, removing equipment, removing materials, and reducing hazardous constituents from facilities must be established. This is a twist on when is the best time to stabilize the facility for demolition. The cost of each path forward compared with what condition the facility is in (from a physical, safety, radiological, chemical, regulatory, and environmental standpoint) determines when the switch should occur from decontamination to stabilization in preparation for remote demolition. Again, remember to take into account verification requirements and costs as well as the different pathways the waste must be disposed of. Even within the facility D&D process, the cost of segregating waste for disposal may dictate that more or less cleaning needs to occur.

It has been mentioned before that routine monitoring could not only be used to meet day-to-day operating requirements but also be valuable in establishing characterization of the facility. Other actions that have to be performed to meet operating requirements should be evaluated early in the process to see if they can satisfy multiple functions for D&D. The requirement to monitor waste could also be used to segregate different types of waste. Inventorying waste can be used to package waste. Training classes could be set up to meet several training requirements at once. Standard operating procedures could be set up to meet safety, quality, energy, and configuration requirements as part of the day-to-day activities. A flexible project baseline set up in an innovative manner can enhance and meet the requirements of the up-front planning, design, field, budgeting, reporting, and close out activities.

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## 18.22 Conclusion: The Act of D&D and the Art of Balance

The act of D&D and the art of balance is an evolutionary process that must end the study and planning of how to do the process and drive the process into field performance to meet the D&D mission of completing work, as illustrated in Figure 18.18. The project can be driven to exceed all kinds of risk, quality, safety, procurement, efficiency, other direct

**FIGURE 18.18**

Waste management truck leaves the gate.

costs, and political standards, but, at the end of meeting all the requirements, completion of the work can only be accomplished by *doing the work*.

The art of balance can be applied on any project. The project manager can balance the different functions within a project for greatest efficiency. He can balance the different phases of a project for greatest effectiveness. He can balance which activities are performed based on the physical conditions. He can balance the equipment and labor being used to both current and projected project conditions. He can use design-flexible systems rather than re-tooling when the project conditions change. He can concentrate on project requirements, regulatory compliance, and direction rather than on maintaining operational procedures. He can use graded and tailored approaches. He can manage risk and quality to acceptable levels. He can balance the worker's expertise with training of the work at hand and support this with properly detailed procedures by matching the right worker to the right job. He can integrate safety and hazards recognition into the day-to-day activities to create an inherently safe environment. He can anticipate the changing conditions that require modifications to the D&D process. And finally, he can establish a system of reporting field status that ties the common understanding of the contract with invoicing with activities with baseline with budget with the work being performed. To accomplish a true balance in D&D work, an effective communication system must be established at all levels of the organization and with all the entities involved understanding how they contribute or inhibit accomplishment of the work.



## Section III

### Introduction

# *Related Engineering and Analytical Processes*

**Kenneth D. Kok**

This section contains the description of related engineering and analytical processes that are used generally in nuclear engineering related to the design and operation of nuclear processes. Chapters 19 and 20 describe the safety evaluations that are used for nuclear facilities. Chapter 19 introduces the risk assessment and safety analysis process that is used for nuclear reactors that are licensed in the United States by the Nuclear Regulatory Commission (NRC). This process has evolved from a relatively simple safety analysis used in the 1950s to a detailed risk assessment process that is used today. Chapter 20 describes the process used in the United States by the Department of Energy for safety analysis of its facilities. It is more prescriptive and less probability and risk based than the process used by the NRC.

Chapter 21 provides a discussion of criticality. Criticality occurs when sufficient fissionable material, such as U-235, is brought together under the appropriate conditions so that a self-sustaining reaction occurs. This reaction is required to make a reactor operate in a controlled manner, but is not desirable if it is not controlled.

Chapter 22 “Heat Transfer, Thermal Hydraulic, and Safety Analysis” and Chapter 23 “Thermodynamics and Power Cycles” are analytical tools used by engineers to evaluate reactor and power-producing systems. Heat transfer and thermal hydraulics are not only important in the operation of nuclear reactors, they are also critical in the evaluation of how the systems will respond under upset conditions. The chapter on thermodynamics is included to show how the energy generated by the reactor is transferred by the reactor cooling system to the turbine power generating system used to produce electricity.

Chapter 24 looks at some of the economic evaluations that can be applied to nuclear reactor systems. The calculations of costs throughout the fuel cycle and the cost of construction of the nuclear plant determine what the cost of the generated electricity will be. Chapter 25 discusses radiation protection, which is important to persons working with radioactive materials in any setting and also important to the safety of the general public. A simple form of radiation protection is sun block to protect the skin from natural solar radiation, or a lead apron by a dental technician to protect a person while X-rays are being taken.

Finally, Chapter 26 introduces the impact of low levels of radiation and raises the question concerning how it is regulated and the impact of those regulations.

# 19

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## *Risk Assessment and Safety Analysis for Commercial Nuclear Reactors*

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Yehia F. Khalil

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### **19.1 Background: Status of Operating Commercial Nuclear Reactors in the United States**

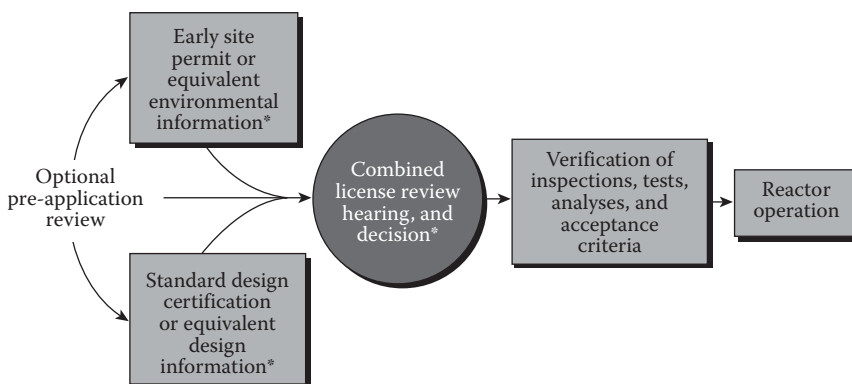
As described in NUREG/BR-0298 (Rev. 2), since the 1960s and the inception of the Atomic Energy Commission (AEC) and its successor, the Nuclear Regulatory Commission (NRC), about 126 operating licenses (OL) for commercial nuclear power plants in the United States have been approved. Of the 126 plants, 103 plants are currently operating and the remaining 23 plants are shut down and in different stages of decommissioning. The 103 operating nuclear power plants provide about 21% of the electric power in the United States. Of the 103 operating nuclear plants, 35 are boiling water reactors (BWRs) and the remaining are pressurized water reactors (PWRs).

The last commercial nuclear power plant OL was issued in 1993. Since then, the US nuclear industry went into a “dormant period,” and only recently has the momentum for building new nuclear power plants resurged. Specifically, 28 new plants at 19 sites are proposed by the nuclear industry. The new plants are based on five different reactor designs:

- GE nuclear energy economic simplified boiling water reactor (ESBWR, 1500 MW<sub>e</sub>): currently under the NRC staff review
- GE Nuclear Energy Advanced Boiling Water Reactor (ABWR, 1350 MW<sub>e</sub>): design approved in July 1994 and certified in May 1997
- Westinghouse Advanced Pressurized Water Reactor (APWR): System 80+ standard plant design approved in July 1994 and certified in May 1997
- Westinghouse standard plant design AP-1000, 1000 MW<sub>e</sub>: design approved in September 2004; also, AP-600 (600 MW<sub>e</sub>) standard plant design, design approved in September 1998 and certified in December 1999
- AREVA's European Evolutionary Pressurized Reactor (EPR, 1600 MW<sub>e</sub>): currently under NRC staff review

Figure 19.1 outlines the relationships among early site permits (ESPs), standard design certifications (SDCs), and the combined licenses (COLs). Figure 19.2 highlights potential opportunities for public hearings during the ESP review process. An estimated schedule (by fiscal year) for the proposed new reactor licensing applications is shown in Figure 19.3.

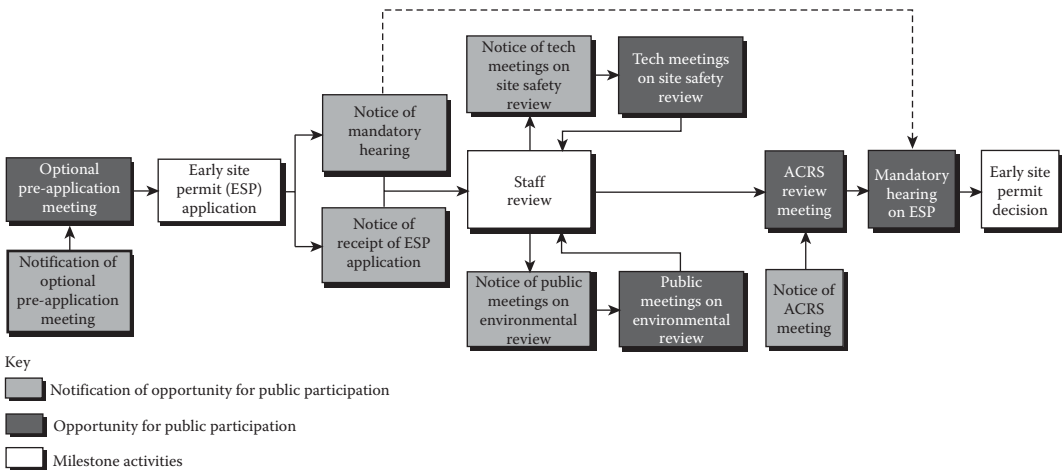
Table 19.1 shows 2010 consortia to test the COL process as well as potential applicants for COLs.



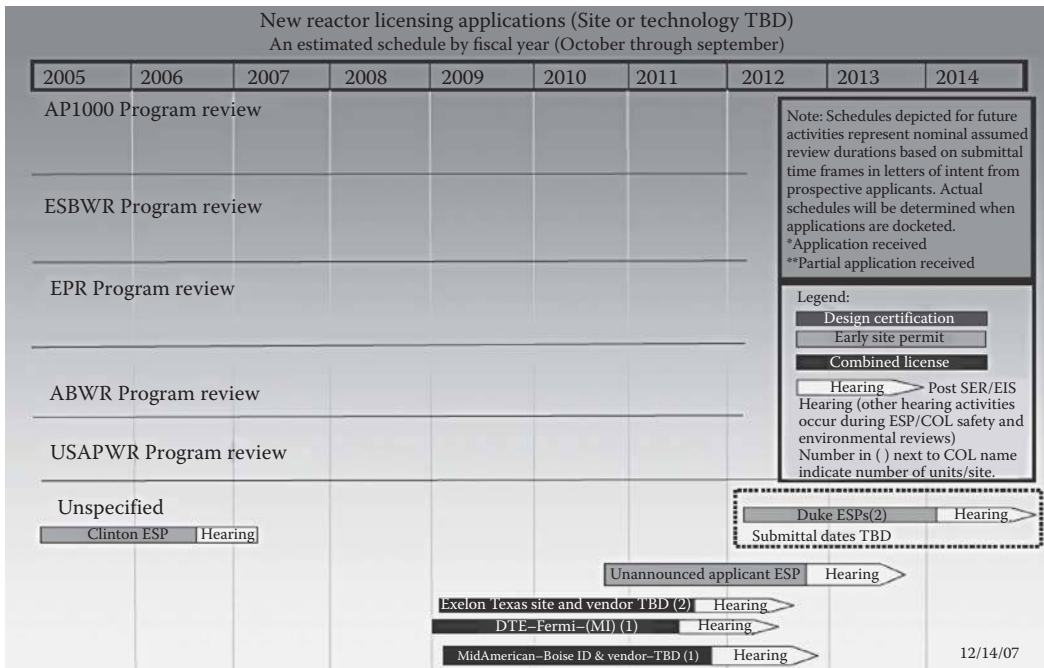
\*A combined license application can reference an early site permit, a standard design certification, both, or neither. If an application does not reference an early site permit and/or a standard design certification, the applicant must provide an equivalent level of information in the combined license application.

**FIGURE 19.1**

Relationships among early site permits (ESPs), standard design certifications (SDCs), and the Combined Licenses (COLs). (From Nuclear Power Plant Licensing Process, NUREG/BR-0298, Rev. 2, July 2004.)



**FIGURE 19.2** Opportunities for public hearings during the ESP review process. (From Nuclear Power Plant Licensing Process, NUREG/BR-0298, Rev. 2, July 2004. With permission; *Licensing New Nuclear Power Plants*, NEI Fact Sheet, September 2000.)



**FIGURE 19.3** New reactors licensing application—an estimated schedule by fiscal year. (From Nuclear Power Plant Licensing Process, NUREG/BR-0298, Rev. 2, July 2004.)

**TABLE 19.1**

2010 Consortia to Test the Combined License (COL) Process as Well as Potential COL Applicants

Utility	Reactor Design & Site	Comments
NuStar <sup>a</sup>	AP-1000 (TVA/Bellefonte)	Lead Consortium
	ESBWR (Grand Gulf)	
Dominion	ESBWR (North Anna)	Lead Consortium
TVA	ABWR (TVA/Bellefonte)	Lead Consortium
UniStar		
(Constellation Energy and AREVA)	EPR (Calvert Cliffs)	Potential COL application in 2008
Entergy	ESBWR (River Bend)	Potential COL application in 2007
Duke	AP-1000 (Greenfield)	Potential COL application in 2006
Progress Energy	TBD (Florida or Carolina)	Potential COL application in 2008
Southern Nuclear	AP-1000 or ESBWR (Vogtle)	Potential COL application in 2008

<sup>a</sup> Progress, TVA, Duke, Constellation, Exelon, Entergy, EDF, FP&L, Bechtel, Southern Nuclear, Westinghouse.

## 19.2 NRC Reactor Licensing Process and Required Permits

The US Nuclear Regulatory Commission (US NRC) has the responsibility for licensing, regulating, and monitoring the operation of all the US commercial nuclear power plants.

Currently, nuclear power plants are licensed under a two-step process as described in Part 50 of the Code of Federal Regulations Title 10 (10 CFR 50). However, to improve the efficiency of this regulatory process, the NRC established in 1989 a new licensing process, per 10 CFR 52, which combines the construction permit and OL with conditions for operation of the plant. Congress affirmed and strengthened the new licensing process as part of the 1992 Energy Policy Act. The new licensing process is known as the COL process. Besides improving the regulatory efficiency, COL can reduce the financial risks for electric generation companies building new nuclear power plants. The issuance of COL with ITAAC (inspections, tests, analyses, and acceptance criteria) is under 10 CFR 52.97.

Granting a COL signifies the resolution of all safety issues associated with the plant. However, no applicant licensee has yet been through the entire COL process. The NRC currently estimates that the review and approval of the first set of COLs may require up to four years (NEI Fact Sheet, September 2007).

The NEI Fact Sheet (September 2007) reports that currently two nuclear licensees (Grand Gulf and North Anna) that proposed to build new nuclear reactors have sites already approved by the NRC through the ESP process. Because those licensees “banked” the sites approvals, that portion of the approval process will not be required when they submit their COL applications. Other nuclear licenses that plan to apply for COLs in the near future are preparing applications for site approval simultaneously with their COL applications, and, hence, they will request site approval as part of the COL process.

In either process (10 CFR 50 or 10 CFR 52), before a nuclear power plant can be built and operated, approval must be obtained from the NRC. In both licensing processes, the NRC oversees the construction and operation of a facility throughout its lifetime to ensure regulatory compliance with the NRC’s mission of protecting public health and safety, the environment, and the common defense and security.

Sections 19.2.1 and 19.2.2 provide the details of the old licensing process (10 CFR 50) and the new licensing process (10 CFR 52), respectively. Section 19.2.3 describes the ESP process and Section 19.2.4 discusses the design certification process.

### **19.2.1 Two-Step Licensing Process (10 CFR 50)**

In this process, each commercial nuclear power plant application must undergo a safety review (SR), an environmental review, and antitrust review by the NRC staff. The applicant must submit a safety analysis report (SAR) that contains the plant design information and criteria for the proposed nuclear reactor as well as comprehensive data on the proposed site. Also, this document should discuss various postulated accident scenarios and the plant-specific safety features that are designed to prevent accidents or mitigate their consequences should they occur. Furthermore, the application must contain a comprehensive assessment of the environmental impact of the proposed plant. Finally, the applicant must submit information for antitrust reviews of the proposed plant.

After an application to construct a nuclear plant is submitted, the NRC staff determines whether it contains sufficient information to satisfy the Commission requirements for a detailed review. If the application is accepted, the NRC holds a public hearing near the proposed site to familiarize the public with the safety and environmental aspects of the proposed application, including the planned location, the type of plant to be build, the regulatory process, and the provisions for public participation in the licensing process. Numerous public meetings of this type are typically held during the course of the reactor licensing process.

To maintain permanent records of correspondences, all documents related to the application are placed in the agency document management system, ADAMS, and in the NRC Public Document Room located in Rockville, Maryland. The NRC issues a press release to media near the proposed plant announcing receipt of the application and sends copies of the announcement to federal, state, and local officials. In addition, a notice of receipt of the application is published in the Federal Register (FR).

The NRC staff reviews the application to determine whether the nuclear plant design meets all applicable regulations per 10 CFR Parts 20, 50, 73, and 100, respectively. The review includes the following elements:

- (a) Site characteristics that include surrounding population, seismology, meteorology, geology, and hydrology
- (b) Nuclear plant design
- (c) Anticipated recovery actions of the plant to postulated accidents
- (d) Plant operations, including the applicant's technical qualifications to operate the plant
- (e) Discharges from the plant into the environment (i.e., radiological effluents)
- (f) Emergency plans (EP)

After the NRC staff completes its review, it prepares a Safety Evaluation Report (SER) summarizing the anticipated effects of the proposed facility on public health and safety.

Each application to construct, or operate, a nuclear power plant is reviewed by the Advisory Committee on Reactor Safeguards (ACRS). This review starts early in the licensing process, and a series of meetings with the applicant and the NRC staff are typically

held during the review process. After the ACRS completes its review process, it submits the results in a report to the NRC.

In accordance with the National Environmental Policy Act, the NRC staff performs an environmental review to evaluate potential environmental effects and anticipated benefits of the proposed plant. After completing this review, the NRC issues a draft Environmental Impact Statement (EIS) for comments by the appropriate federal, state, and local agencies as well as by the public. When this step is completed, the NRC issues a Final EIS that addresses all comments received.

According to the Atomic Energy Act (AEA), a public hearing must be held before a construction permit is issued to the nuclear licensee. The public hearing is conducted by a three-member Atomic Safety and Licensing Board. The board members include an attorney, who acts as chairperson, and two recognized technical experts. Members of the public may submit written or oral statements of concerns to the licensing board to be entered into the hearing record. Alternatively, they may petition to intervene as full parties in the hearing.

#### **19.2.1.1 Limited Work Authorization**

The issuance of a Limited Work Authorization (LWA) is under 10 CFR 50.10(e). In some cases, the NRC may authorize the licensee to do some construction at the site prior to the issuance of a construction permit. This authorization is known as an LWA and is done at the risk of the licensee. The LWA may be granted only after the licensing board has made all of the National Environmental Policy Act (NEPA) findings required by the Commission's regulations for authorizing construction and has determined that there is reasonable assurance that, from a public health and safety standpoint, the proposed site is a suitable location for a nuclear power reactor of the proposed type and capacity.

After the construction permit is issued, the applicant must, if it did not as part of the original application, submit a final safety analysis report (FSAR) to support its application for an OL. The FSAR describes the final design of the plant and its operating and emergency procedures. The NRC prepares a final SER for the OL, and also the ACRS makes an independent evaluation and presents its findings and recommendations to the NRC.

While a public hearing is neither mandatory nor automatic for OL applications, the NRC publishes a notice in the FR that it received an application for an OL, has accepted it for review, and is considering issuance of the license. This notice allows the public to request a hearing. If a public hearing is held, the same decision process as described for the construction permit hearing still applies.

#### **19.2.2 Combined License (COL) Process (10 CFR 52)**

Similar to the construction permit under the two-step licensing process (10 CFR 50), the COL authorizes construction of the commercial nuclear power plant. Essentially, the COL must contain the same information required in an application for an OL issued under 10 CFR 50. It must also specify the inspections, tests, and analyses that the applicant must perform. In addition, the document must specify acceptance criteria that are necessary to provide reasonable assurance that the plant is constructed and will be operated in conformity with all applicable regulatory requirements. If the ESP and design certification are not referenced, then the NRC reviews the technical and environmental information as described for the two-step licensing process (10 CFR 50). Finally, there is also a mandatory hearing for a COL process.



After issuing a COL, the NRC staff authorizes operation of the plant only after verifying that the applicant licensee completed the required inspections, tests, and analyses and those acceptance criteria were fully met.

During the construction period and at periodic intervals, the NRC publishes notices of these completions in the FR. Before the date scheduled for the initial loading of fuel in the nuclear reactor by at least 180 days, the NRC publishes a notice of intended operation of the plant in the FR. At this point, there is an opportunity for a public hearing; however, the NRC will consider petitions for a hearing only if the petitioners provide evidence (*prima facie*) that the licensee has not met, or will not meet, the acceptance criteria. The NRC determines whether the acceptance criteria have been fully met, before it allows the applicant licensee to operate the plant.

### 19.2.3 Early Site Permit (ESP) Process

An ESP process addresses issues related to site safety, environmental characteristics, and emergency preparedness (EP). These issues are independent of the plant-specific design. The ESP application must evaluate all potential physical barriers to developing an acceptable EP. The ESP application contains the following information:

- Site boundaries
- Locations and descriptions of any industrial, military, or transportation facilities and routes
- Seismic, meteorologic, hydraulic, and geologic data
- Existing and projected future population of the surrounding area
- Evaluation of alternative sites
- Proposed general location of each plant planned to be on the site
- Number, type, and power level of the plants planned for the site
- Maximum discharges (effluents) from the site
- Type of plant cooling system (ultimate heat sink UHS) to be used
- Radiation dose consequences of postulated accidents
- Plans for coping with emergency situations

The NRC documents its findings on site safety characteristics and emergency planning in an SER. Also, findings related to environmental protection issues are to be documented in draft and final EIS.

#### 19.2.3.1 LWA

The ESP may allow for an LWA that enables the applicant to perform nonsafety site preparation activities in advance of the issuance of the COL. After the NRC staff and the ACRS complete their SRs and evaluations, the NRC issues a notice in the FR calling for a mandatory public hearing. The ESP is initially valid for a period between 10 and 20 years, and can be renewed for 10–20 years.

As described in the NEI Fact Sheet (September 2007), it takes about 1–2 years to develop an ESP application; depending on whether it is a “greenfield” site or a site adjacent to an existing facility. After the ESP application is submitted, the process of the Commission’s review and approval (including the public hearing) may take about three years.

Figure 19.1 shows the relationships among ESPs, SDC, and the COL. Figure 19.2 delineates potential opportunities for public hearings and involvements during the ESP review process.

#### 19.2.4 Design Certification

Independent of a specific plant site, the NRC may approve and certify a standard nuclear plant design through a rulemaking process and the design certification would be valid for 15 years. In another words, the design certification process allows the plant designers to secure an advance NRC approval of standard plant designs. Later, these plant designs can be ordered, licensed for a particular site, and built.

An application for a standard plant design certification must contain proposed ITAAC for the standard design. Additionally, the application must demonstrate compliance with the Commission's relevant regulations.

Typically, the SR of the application is based primarily on the information provided by the applicant under oath. An application must contain a level of design information sufficient to enable the NRC to reach a final conclusion on all safety questions related to the plant design. Essentially, a design certification application should provide a complete nuclear plant design, with the exception of site-specific design features such as the UHS and the intake structures.

The application must cover the design basis; the limits on operation; and a safety analysis of structures, systems, and components (SSC) of the plant as a whole. The scope and contents of the application are equivalent to the level of detail found in an FSAR for a currently operating commercial nuclear power plant. The NRC staff prepares an SER that describes its review findings of the plant design and how such a design meets all applicable regulations.

Once a design certification application has been submitted, the NRC takes 36 to 60 or more months to complete its review and rulemaking, depending on whether the NRC staff has previously reviewed and approved the selected reactor technology.

In addition to the NRC reviews, the ACRS reviews each application for an SDC, together with the NRC staff's SER, in a public meeting. Upon determining that the application meets the relevant standards and regulatory requirements, the NRC staff drafts a rule to issue the SDC as an appendix to the 10 CFR 52 regulations. Members of the public may submit written or oral concerns on the proposed design certification rule; however, at its discretion, the NRC may decide to hold a public hearing to address those concerns.

The issues that are resolved in a design certification rulemaking are subject to a more restrictive change process than issues that are resolved under other licensing processes. That is, the NRC cannot modify a certified design unless it finds that the design does not meet the applicable regulations in effect at the time of the design certification, or if it is necessary to modify the design to ensure adequate protection of the public health and safety.

An application for a COL under 10 CFR 52 can incorporate by reference a design certification and/or an ESP. The advantage of this approach is that the issues resolved during the design certification rulemaking and the ESP hearing processes are precluded from reconsideration later at the COL stage.

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## 19.3 Overview of Probabilistic Risk Assessment (PRA)

### 19.3.1 Historical Background

In 1967, F. R. Farmer of the United Kingdom proposed that the probabilities as well as consequences of potential accidents need to be estimated to assess the associated risk. Farmer used I-131 as a surrogate for consequences. By plotting the probability and consequence of each postulated accident, one could distinguish those with high risk from those with low risk. He proposed a boundary line as a criterion for acceptable risk. Farmer's work was a conceptual breakthrough in nuclear reactor safety analysis. Farmer takes full credit as the originator and pioneer of PRA.

The Reactor Safety Study (RSS) started in 1972 as directed by the US AEC, and it took about three years for the study to be completed.

In this study, the Surry plant (a Westinghouse PWR) and the Peach Bottom plant (a GE BWR) were used as reference designs. The study estimated the risk from the operation of 100 reactors in the United States and concluded that the risk is small. The major contributors to plant risk were found to be caused by transients and small loss of coolant accident (LOCA) initiators as compared with large LOCA events.

The publication of the RSS (WASH-1400 report) in 1975 resulted in serious industry and public concerns and caused major controversial issues about the unreliability and incompleteness of the claims made in the study. As a result, in July 1977, the NRC organized a group of experts led by Harold Lewis to examine the results claimed by the WASH-1400 study. Based on the findings of the Lewis Committee (NUREG/CR-0400), the NRC withdrew its endorsement of the RSS. The following are excerpts from the NRC statement of 1978:

The Commission has reexamined its views regarding the study in light of the Review Group critique.

The Commission withdraws any explicit or implicit part endorsement of the Executive Summary.

The Commission does not regard as reliable the Reactor Safety Study's numerical estimates of the overall risk of reactor accidents.

In the wake of the 1979 Three Mile Island (TMI) accident, the nuclear utility industry pursued several initiatives to improve plant safety by better understanding operational risks and plant-specific vulnerabilities. One of the most notable efforts was the Industry Degraded Core Rulemaking (IDCOR) program in the early 1980s. The primary focus of the IDCOR program was to better understand the consequences of accident scenarios that extend beyond the design basis of the plant. In addition to the IDCOR program, several plant-specific PRA studies were developed such as the Oconee, Zion, Indian Point, and Seabrook studies. On the regulatory side, in 1983 the NRC published a PRA procedures guide (NUREG/CR-2300). This report continues to be a useful resource to the PRA practitioners. In 1986, the NRC safety goals policy statement was issued, and in 1988 the Commission issued Generic Letter (GL) 88-20 requesting each nuclear licensee to conduct an individual plant examination for the identification of severe accident vulnerabilities for internal events (known as the IPE study) and for external events (known as the IPEEE study), respectively. The GL 88-20, however, did not require the quantification of

off-site consequences in terms of the radiological dose and health effects. The 1995 NRC final policy statement (FR #300238, SECY-89-102) encouraged the nuclear licensees to use PRA insights to the extent possible for supporting their plant operation, for making risk-informed decisions, and for requesting regulatory relief from those overly conservative operational burdens. However, the Commission cautioned that the utilization of PRA insights does not replace the use of defense-in-depth (DID) principles and the traditional deterministic methods. Table 19.2 provides a timeline of key documents and events of most interest to the PRA community.

**TABLE 19.2**

Timeline of Key Documents and Events of Most Interest to the PRA Community

Events/Documents	Timeline
WASH-740 (The Brookhaven Report)	1957
TID-14844, "Calculation of distance factors for power and test reactor sites," Technical Information Document, United States Atomic Energy Commission	1962
Farmer paper at IAEA Symposium on Containment	1967
NUREG-75/014 (WASH-1400)	1975
NUREG/CR-0400, Lewis Risk Assessment Review Group Report to the Nuclear Regulatory Commission	1978
NRC, Statement on Risk Assessment and the RSS Report (WASH-1400)	1978
TMI-2 accident	1979
NUREG-0660, NRC Action Plan Development as a Result of the TMI-2 Accident, U.S. Nuclear Regulatory Commission, Rev. 1	1980
Industry PRA studies: Oconee, Zion, Indian Point, and Seabrook	1981–1983
NUREG/CR-2300: PRA Procedures Guide	1983
NUREG-0880	1983
NRC Safety Goals	1983, 1986
NRC Policy statement on severe reactor accidents	1985
Chernobyl accident	1986
Individual Plant Examinations (NRC GL 88-20)	1988
SECY-89-1102, Implementation of the Safety Goals	1990
IPEEE for External Events	1991
NUREG-1150 - five reference plants: Surry-1, Zion-1, Sequoyah-1, Peach Bottom-2, and Grand Gulf-1	1991
NRC final policy statement on the use of PRA methods in nuclear regulatory activities	1995
EPRI PSA Applications, EPRI TR-05396	1995
Risk-Informed Regulatory Guides (General guidance 1.174/Rev. 1 in November 2002, IST 1.175, GQA 1.176, TS 1.177, and ISI 1.178)	1998
NRC New Reactor Oversight Process (ROP)	1998
SECY-98-144: NRC White Paper on Risk-Informed and Performance-Based Regulation	1999
SECY-99-256: NRC Rulemaking Plan for Risk-Informing Special Treatment Requirements	1999
SECY-99-264: NRC Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR 50	1999
Risk-Informing Part 50	2000
ASME PRA Standard	2002
Revised NUREG/CR-6595 on containment failure modes and bypass events	
SECY-04-0210, Status of accident sequence precursor program and the development of standardized plant analysis risk models	2004
NUREG/CR-6928: Industry-Average Performance for Components and Initiating Events at US Commercial Nuclear Power Plants	2007

The risk-informed and performance-based (RI-PB) regulations were introduced to the nuclear industry by the NRC in the 1997 time frame. The primary objective of these regulations is to make the regulatory process more efficient by focusing the Commission resources on what is important to protect public health and safety. For the nuclear utilities, this objective resulted in classifying the plant SSC into risk-significant (RS) and non-risk-significant SSCs. More plant resources should be focused on maintaining the reliability and availability of the risk-significant SSCs and, hence, plant safety would be improved using this cost-effective strategy.

Starting in 1998, the Commission produced the following guidance documents to assist the nuclear licensees in understanding the minimum requirements to meet the intent of risk-informed regulations. These documents are as follows:

- Regulatory Guide 1.174 and Standard Review Plan Chapter 19: Regulatory guidance documents
- Regulatory Guide 1.175 and Standard Review Plan Section 3.9.7: Inservice testing (IST)
- Regulatory Guide 1.176: Graded quality assurance (GQA)
- Regulatory Guide 1.177 and Standard Review Plan Section 6.1: Technical specification changes (TSC)
- Regulatory Guide 1.178 and Standard Review Plan Section 3.9.6: Inservice inspection (ISI)

As the nuclear licensees started to use PRA insights to support risk-informed applications such as extension of the allowed outage time of the emergency diesel generators, and plant technical specifications (TS) changes, PRA model quality and fidelity became the focal point for the nuclear licensees and the regulators. As a result, the owners, groups such as the Westinghouse WOG, the former Combustion Engineering CEOG, and the BWR owners group, formed PRA peer review committees to verify the quality of the licensees' PRA models. The intent of the peer review process can be described as follows:

1. Assess the technical quality and adequacy of a PRA for potential risk-informed applications for which the PRA may be used.
2. Identify areas of consistency or inconsistency in the treatment of issues important to plant risk and implementing risk-informed applications.
3. Ensure that the PRA model represents the plant as built and as operated.

In addition to the PRA peer review process, the American Society of Mechanical Engineers (ASME) sponsored an initiative to develop the so-called ASME PRA Standard. This document establishes requirements for the PRAs used to support risk-informed decisions for commercial nuclear power plants. The ASME PRA Standard (ASME-RA-S-2002) also establishes a range of PRA capabilities for applicability to different risk-informed decisions, as described by the PRA Capability Categories I, II, and III, respectively.

### 19.3.2 Scope of PRA

Risk has different types, such as individual risk, societal risk, ecological risk, and economic risk. Risk can also have different meanings, such as the maximum thinkable (probable or credible) outcome of an undesirable event, worst-case outcome, and most

probable undesired outcome. In this context, the worst-case scenarios are generally based on conservative assumptions that yield the maximum loss level as a result of failure of all possible layers of DID.

The overarching approaches to risk assessment can be sorted into two major groups. The first group represents qualitative approaches that provide less accurate risk-related information, while the second group represents those quantitative approaches that provide quantifiable results based on actual field reliability data, predictions of the so-called physics of failure (POF) models, test data, or expert opinion solicitation. Examples of risk assessment approaches that fall into the first group include the design and process failure mode and effects analyses, hazard and operability analysis, hazard checklists, and influence diagrams. Fault tree analysis (FTA), event tree analysis (ETA), fault tree/event tree linking, decision trees, consequence analysis, and cost/benefit analysis are examples of the quantitative approaches to risk analysis.

In the nuclear field, the PRA is a quantitative approach intended to answer three fundamental questions, known as the PRA risk triad:

- What could go wrong?
- How likely it is for the undesirable events to happen?
- What are the consequences should the undesirable events happen?

The question of what can go wrong is answered by identifying the accident initiators (called the initiating events; IE). The PRA analyst then needs to answer the questions of: "With what likelihoods?" and "With what potential consequences?" These latter two questions can be answered by quantifying the initiating events frequencies and the risk associated with each accident sequence, respectively.

The author of this chapter contends that in answering the first question ("What can go wrong?") the PRA analyst has to "think the unthinkable."

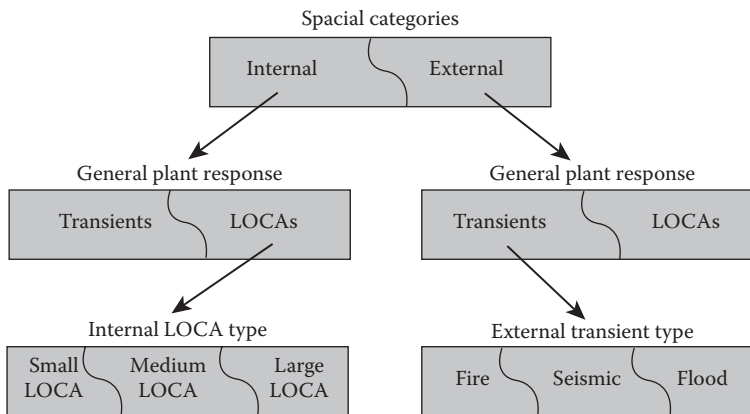
A plant-specific PRA model provides quantitative results in the form of point estimates and probability distributions that postulated risk events may occur in a nuclear power plant. Two figures of merit are commonly used by the PRA analysts to demonstrate how a nuclear power plant meets the safety goals put forth by the NRC, namely the core damage frequency (CDF) and the large early release frequency (LERF). These figures of merit are applicable to all modes of operation of a nuclear power plant. For the shutdown mode, other figures of merit are equally important, such as the time to core uncovering and time to core boiling.

A full-scope PRA model comprises the following three levels.

### 19.3.3 Level-I PRA/Front-End Analysis

The main tasks of Level-I PRA analysis include the assessment of

- Frequencies of initiating events (IE); Figure 19.4 shows IE categorization
- Human reliability analysis (HRA)
- Systems reliability models
- Dominant accident sequences (internal events and internal flooding [IF])
- Dominant external events including fire, seismic, tornadoes, and high winds
- Contribution of each dominant accident sequence to the total plant CDF
- Fault tree/event tree linking; Figure 19.5 illustrates the linkage between an event tree and the associated fault trees



**FIGURE 19.4**  
Initiating events (IE) categorization.

The internal events PRA is commonly known as the independent plant examination (IPE) and the external events PRA is referred as the IPEEE.

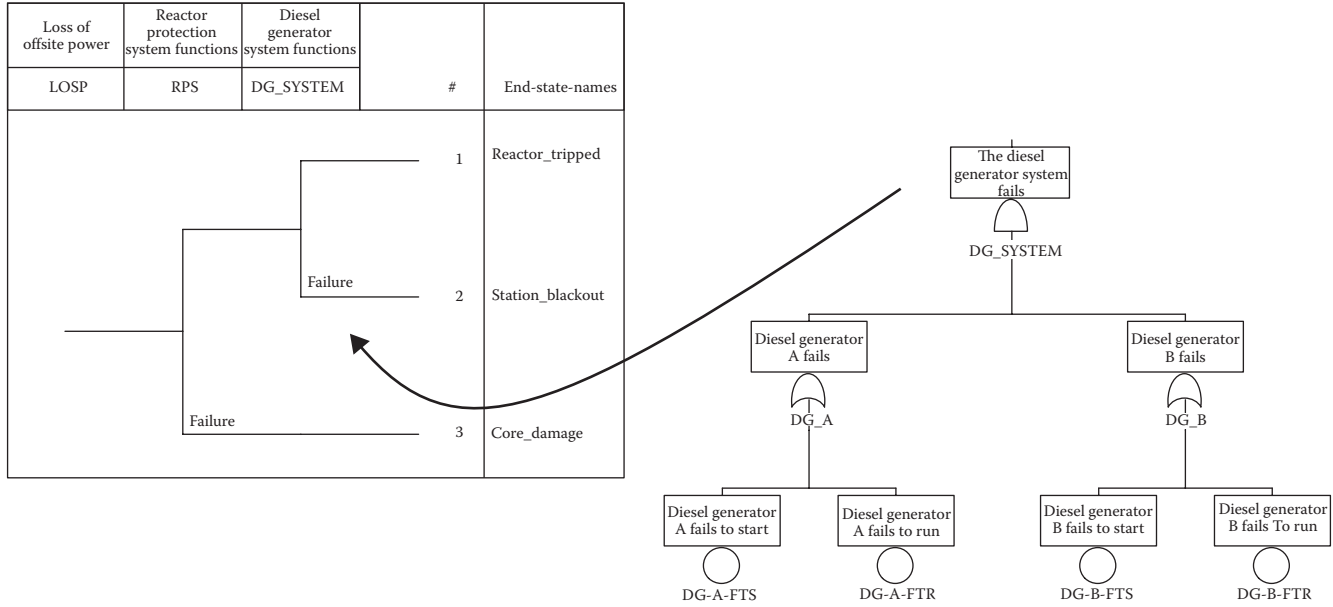
For Level-I PRA tools, the most commonly known used by the nuclear industry include CAFTA for FTA, ETA-II for ETA, and the HRA calculator for human error analysis. The Idaho National Engineering Laboratory developed for the NRC a software package called the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE). The features of the SAPHIRE package include FTA, ETA, and uncertainty analysis.

#### 19.3.4 Level-II PRA/Back-End Analysis

Level-II RPA deals with

- Severe accident progression starting from the plant damage states, which are the end states of Level-II PRA
- Operator recovery actions to prevent core damage progression to vessel breach and containment failure
- Severe accident phenomenology such as high-pressure melt ejection leading to direct containment heating (DCH), in-vessel steam explosion leading to alpha-mode containment failure, ex-vessel steam explosion potentially leading to vessel relocation, hydrogen burn, liner melt-through, and basemat melt-through
- Source term evaluation and categorization, for example, LERF
- Containment thermal hydraulic (T/H) response during severe accident progression
- Examples include containment failure modes (early versus late), fragility curves, and failure mechanisms

The most commonly used Level-II PRA tools include CAFTA for FTA, ETA-II code for containment analysis, and the modular accident analysis program (MAAP) for severe accident simulation. The MAAP code is also used for equipment success criteria development, although some PRA practitioners use the RELAP code for this purpose.



**FIGURE 19.5**  
Event tree/fault tree linking.



### 19.3.5 Level-III PRA

The intent of Level-III PRA is to assess

- Radionuclide releases to the environment (radiological dose assessment)
- Nuclear aerosol plume dispersion models
- Risk to the public health including early and late fatalities

The most commonly known Level-III PRA tools include the CRAC code and the MACCS code.

The plant-specific PRA model contains key elements such as

1. Accident initiators based on plant's specific design. These are called special initiators such as loss of offsite power and loss of service water.
2. Accident sequences based on expected system responses and operator actions following plant procedures (e.g., EOPs, AOPs, and plant-specific severe accidents guidelines).
3. System models (such as MFW, AFW, HPSI, LPSI, and SW) that represent installed system designs and their support systems.
4. Equipment failure rates based on plant failure data, generic historical data, or a combination of both.
5. Maintenance unavailabilities based on plant maintenance procedures, frequencies, and practices.
6. Human error rates calculated using plant-specific procedures and simulator results.

### 19.3.6 Uses of PRA

Nuclear power plants risk practitioners use PRA methods and tools for

- Identifying plant risks and vulnerabilities
- Identifying risk-significant SSCs and, hence, help focusing the plant limited resources on what is critical to plant safety
- Improving equipment availability and reliability while balancing reliability versus maintenance unavailability
- Addressing plant license amendment requests using risk-driven insights
- Managing outage risk
- Managing online risk and supporting PRA daily demand work and maintenance activities
- Ensuring plant compliance with the NRC safety goals and requirements
- Assisting the nuclear licensee in responding to the NRC GL, Information Notices, and other regulatory requirements

Table 19.3 delineates the main features of probabilistic risk assessment.

**TABLE 19.3**

Main Features of Probabilistic Risk Assessment

<b>PRA Is</b>	<b>PRA Is Not</b>
A living document to be updated frequently (typically, major updates are done at frequencies ranging from every 18 months and up to three or sometimes five years).	PRA document is a one-time effort and represents the plant as built only.
The PRA model has to be updated to ensure that it represents the built, as built and as operated (i.e., the model should reflect design changes, retrofits, and changes in plant operating procedures).	
A risk management and decision-making tool that can be used during all plant operational modes.	PRA is the sole basis of regulatory decisions.
PRA insights should be used in conjunction with other engineering deterministic approaches.	Plant risk management and decision making should be based on PRA insights only.
PRA is a supplement to the regulatory process.	
The PRA model is driven by a blend of stochastic events, physical phenomena, thermal hydraulics, plant operating procedures, technical specifications, HRA, and DID principles.	The PRA model is physics based.
The PRA model provides risk information that contains aleatory and epistemic uncertainties.	The quantification results of the PRA model are 100% certain.

## 19.4 Aleatory and Epistemic Uncertainties in PRA

Probabilistic values that are used in the risk model represent subjective information that measures the degree of belief of the PRA analyst who developed the model. Hence, there are uncertainties associated with these probabilistic values and parameters in a risk model. There are two types of uncertainties, as detailed below.

### 19.4.1 Aleatory Uncertainty

This type of uncertainty is sometimes referred to as the “irreducible uncertainty” because it is caused by the stochastic and unpredictable nature of events. Even the knowledge and opinions of the subject matter experts (SMEs) cannot be expected to reduce aleatory uncertainty; however, their knowledge may be useful in quantifying the uncertainty.

### 19.4.2 Epistemic Uncertainty

This type of uncertainty is caused by the lack of complete knowledge about the behavior of the component, system, or the physical phenomena of interest. However, the epistemic uncertainty can be eliminated, or at least reduced, with sufficient theoretical studies, tests, and use of SME’s judgment. Hence, the epistemic uncertainty is sometimes referred to as the reducible uncertainty.

Parameters uncertainty is generally epistemic in nature as one simply does not know the correct values for the model’s input parameters. Probability is always a measure of the degree of belief of the analyst.

As a result, quantitative risk estimates should be expressed in terms of probability distributions rather than being presented as point estimates. However, one of the technical

challenges in the treatment of epistemic uncertainty is the encoding of probability distributions based on expert judgments and scientific evidence such as experimental results and mode predictions based on the POF of the phenomenon or failure mechanism being analyzed.

Fortunately, the Bayesian approach allows incorporation of all available relevant information into the assessment of probabilities. It allows quantification of both epistemic as well as aleatory uncertainties, and the combination of their effects into a single probability value of an undesirable event, or into a single probability distribution for the consequences of assumed risk. Once the uncertainties are determined, they can be propagated through the risk model using techniques such as Monte Carlo simulation.

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# 20

## *Nuclear Safety of Government-Owned, Contractor-Operated Nuclear Facilities*

Arlen R. Schade

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## 20.1 Introduction

This chapter covers the laws and processes that have defined the nuclear safety program since 1946. The joint responsibility of the Atomic Energy Commission (AEC) for promotion and regulation is discussed. The separation of the AEC into the Energy Research and Development Administration (ERDA) and the Nuclear Regulatory Commission (NRC) and their respective missions are discussed.

ERDA was left with the mission of production of nuclear weapons and other nuclear missions. ERDA still had a promotion and regulation responsibility for the nuclear facilities that they owned. In 1977, ERDA was incorporated into the U.S. Department of Energy (U.S. DOE). The U.S. DOE was formed in response to the 1973 energy crisis.

The Chernobyl nuclear reactor accident caused several reviews to be made of the DOE weapons nuclear facilities by the National Research Council. While no imminent safety hazards were identified, a number of suggestions were made to improve nuclear safety.

In 1988, Congress created the Defense Nuclear Facility Safety Board (DNFSB). The DNFSB was given oversight authority over DOE defense-related nuclear facilities.

To encourage the development of nuclear power reactors, Congress passed the Price-Anderson Amendment Act (PAAA). This law indemnified the owners of nuclear facilities from potential unbounded claims if there was a nuclear accident that caused harm to the public. The law was revised in 1988 to require DOE to develop an enforcement process for violations of nuclear requirements.

To support this enforcement process, DOE developed several DOE orders, guides, and manuals. These documents were to be used by the contractors to prepare documented safety analysis (DSA) and technical safety requirements (TSRs).

This chapter includes a discussion of how a hazard category is determined for a nuclear facility and gives a thumbnail discussion of the development of a hazard and accident analysis.

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## **20.2 Laws That Have Shaped the Current DOE Nuclear Safety Process**

This section discusses the laws that have defined the processes for developing the hazard and accident analysis and the controls necessary to ensure the safe operation of government-owned contractor-operated nuclear reactors and nuclear facilities.

### **20.2.1 Atomic Energy Act of 1946**

Following the end of World War II in August 1945, it was recognized that the emerging atomic energy technology would require congressional legislation to ensure that atomic energy would be properly regulated. The Atomic Energy Act (AEA) of 1946 was sponsored by Senator Brien McMahon, who chaired the U.S. Senate Special Committee on Atomic Energy. The AEA of 1946 was passed and signed into law after intensive debate by politicians, military planners, and atomic scientists. The Act became effective January 1, 1947 (AEA 1946, Public Law 585–79).

The AEA of 1946 reflected America's postwar optimism, with Congress declaring that atomic energy should be employed not only in the form of nuclear weapons for the nation's defense but also to promote world peace, improve the public welfare, and strengthen free competition in private enterprise.

The AEA of 1946 defined how the U.S. Government would regulate the use of the new nuclear technology. The development of nuclear weapons and atomic energy uses for the public were placed under civilian control as opposed to military control. The AEA of 1946 established the U.S. AEC that was chartered to develop and produce future atomic weapons and was also chartered to facilitate the peaceful uses of nuclear energy.

One of the most significant provisions of the AEA of 1946 was the prohibition of sharing atomic technology with other powers, even with the U.S. allies. The AEA of 1946 stated, "All data concerning the manufacture or utilization of atomic weapons, the production of fissionable material, or the use of fissionable material in the production of power" are restricted data unless the information has been declassified. The phrase "all data" included every suggestion, all data concerning the manufacture or utilization of atomic weapons, the production of fissionable material, or the use of fissionable material in the production of power.

This was the development of "born secret" or "classified at birth" information. The assumption was that any atomic energy information was classified until it was declassified. Many believed that to be an infringement on free speech. It was a novel new concept in the U.S. law and is still the law.

The rest of the countries of the world saw this as an action by the United States to maintain an atomic energy monopoly. The restriction on sharing of atomic technology angered the United States' closest ally, the United Kingdom, which had openly shared their information with the United States during the Manhattan Project. They and other countries embarked on their own national programs.

The AEC succeeded the Manhattan Engineer District of the U.S. Army Corps of Engineers. David Lilienthal was appointed the first Chairman of the AEC. Congress gave the new civilian Commission extraordinary power and independence to carry out its mission. To provide the Commission exceptional freedom in hiring scientists and professionals, Commission employees were exempt from the Civil Service system. Because of the need for great security, all production facilities and nuclear reactors

would be government-owned, with all technical information and research results under Commission control. The National Laboratory system was established from the facilities created under the Manhattan Project. Argonne National Laboratory was one of the first laboratories authorized under this legislation as a contractor-operated facility dedicated to fulfilling the new Commission's mission.

### **20.2.2 Atomic Energy Act of 1954**

The AEA of 1946 was replaced by the AEA of 1954 (Public Law 80–703). The major change was to substantially modify the AEA to be supportive of a civilian nuclear power industry. The changes were necessary to implement the President Eisenhower's Atoms for Peace program.

The AEA of 1954 assigned the AEC the functions of encouraging the use of nuclear power and regulating its safety. The AEC regulatory programs sought to ensure public health and safety from the hazards of nuclear power without imposing excessive requirements that would inhibit the growth of the industry. This was a difficult goal to achieve, especially in a new industry, and within a short time, the AEC programs stirred considerable controversy. During the 1960s, an increasing number of critics charged that the AEC regulations were insufficiently rigorous in several important areas, including radiation protection standards, nuclear reactor safety, plant setting, and environmental protection. On the other hand, the Utilities were dissatisfied with the process and antinuclear special interests used the regulatory process to slow down the licensing process that increased the cost of construction of nuclear power plants.

Production, utilization, and research nuclear facilities and activities were generally regulated for nuclear safety through the AEC Program or AEC Site Office.

### **20.2.3 Price–Anderson Act of 1957**

The Price–Anderson Act of 1957 (Public Law 85–256) limited the liability of the nuclear industry in the event of a nuclear accident in the United States. Because of the safety uncertainty posed by the nuclear industry in the United States, private insurance companies were unwilling to fully underwrite a nuclear power plant. The lack of financial security would have hindered the development of the nuclear industry. The federal government intervened with this amendment to the AEA of 1954.

The purpose of the Price–Anderson Act of 1957 was to ensure the availability of a large pool of funds to provide prompt and orderly compensation of members of the public who incur damages from a nuclear or radiological incident no matter who might be liable. The Act provides “omnibus” coverage, that is, the same protection available for a covered licensee or contractor extends through indemnification to any persons who may be legally liable, regardless of their identity or relationship to the licensed activity. Because the Act channels the obligation to pay compensation for damages, a claimant need not sue several parties but can bring its claim to the licensee or contractor.

The Price–Anderson Act required AEC contractors to enter into agreements of indemnification to cover personal injury and property damage to those harmed by a nuclear or radiological incident, including the costs of incident response or precautionary evacuation and the costs of investigating and defending claims and settling suits for such damages.



The scope of the Act includes nuclear incidents in the course of the operation of power reactors, test and research reactors, AEC nuclear and radiological facilities, and transportation of nuclear fuel to and from a covered facility.

#### **20.2.4 Energy Reorganization Act of 1974**

Under the Energy Reorganization Act of 1974 (Public Law 93–438), the AEC was abolished, and two new federal agencies were established in its place to administer and regulate atomic energy activities: the ERDA and the NRC. Licensing and oversight of nuclear power plants and use of radioactive material was now to be performed by the new NRC. The majority of the licensed nuclear power plants in operation today were licensed under the NRC process. The NRC function was to establish by rule or order, and to enforce, such standards to govern these uses as “the Commission may deem necessary or desirable to protect health and safety and minimize danger to life or property.” Commission action under the AEA of 1974 were required to conform to the Act’s procedural requirements, which provide an opportunity for hearings and Federal judicial review in many instances. The AEA of 1974 gave the Commission its collegial structure and established its major offices.

ERDA was responsible for the management of development and production of nuclear weapons, promotion of nuclear power, and other energy-related work, the production, utilization, and research facilities and to also develop and enforce safety requirements. ERDA and NRC lost their exemption from the Civil Services *Program*.

#### **20.2.5 DOE Organization Act of 1977**

In 1977, the responsibilities of the ERDA were transferred to the newly established DOE. Several federal agencies have been established to handle various aspects of U.S. energy policy, dating back to the creation of the Manhattan Project and the subsequent AEC. The impetus for putting them all under the auspices of a single department was the 1973 energy crisis, in response to which President Jimmy Carter proposed creation of the department. The DOE is a cabinet-level department of the U.S. Government responsible for energy policy and nuclear safety. The DOE was established by the Department of Energy Organization Act. Its purview includes the nation’s nuclear weapons program, nuclear reactor production for the U.S. Navy, energy conservation, energy-related research, radioactive waste disposal, and domestic energy production. DOE also sponsors more basic and applied scientific research than any other U.S. Federal Agency. Most of this is funded through its system of DOE National Laboratories (Department of Energy Organization Act of 1977, Public Law 95–91).

In the United States, all nuclear weapons deployed by the U.S. Department of Defense (DOD) are “owned” by the DOE and are actually on loan to DOD. The DOE has the federal responsibility for the design, testing, and production of all nuclear weapons. DOE in turn uses contractors to carry out its responsibilities, design of the nuclear components of the weapon (Los Alamos National Laboratory and Lawrence Livermore National Laboratory), engineering of the weapon systems (Sandia National Laboratory), manufacturing of key components (Los Alamos National Laboratory), testing (Nevada Test Site), and final weapon/warhead assembling/dismantling (Pantex).

### 20.2.6 PAAA of 1988

The 1988 PAAA (Public Law 100–408) is an amendment to the AEA of 1954. The Amendment continues the indemnification of DOE contractors from costs related to public liability. It differed from the original act in two principal ways:

1. It made Price–Anderson coverage mandatory for all management and operating (M&O) contractors, subcontractors, and suppliers conducting nuclear activities for DOE. (For the purposes of the statute, “nuclear” includes “radiological.”)
2. Congress mandated that DOE change its methods of managing nuclear activities at contractor operated sites by requiring DOE to undertake enforcement actions against indemnified contractors for violations of nuclear safety requirements. The establishment of enforcement sanctions as a method of ensuring compliance with safety requirements is intended to minimize the risk to workers and the public.

The 1988 PAAA indemnification applied to all DOE contractors and subcontractors. This was at the expense of making every contractor or subcontractor liable for fines for violating a nuclear safety requirement. The not-for-profit institutions that managed the national laboratories were exempted from the fines.

### 20.2.7 Energy Policy Act of 2005 (EPACT)/PAAA of 2005

On August 8, 2005, the Energy Policy Act of 2005 (EPACT) (Public Law 108–58) became law. EPACT, among other things, contained sections 601 through 610, known as the PAAA of 2005. The Price–Anderson Act resides at section 170 of the AEA of 1954. The Price–Anderson Act at section 170d indemnifies DOE’s contractors and subcontractors under contracts that involve a risk of public liability for a nuclear incident. The indemnification provides a structure that assures that monies are readily available to remedy damage to the public that may occur as the result of a nuclear incident. The indemnity applies as matter of right, not dependent on contractual implementation.

The PAAA 2005 altered the indemnity in three major ways. The statute

1. Establishes prior law, establishes a specific \$10 billion liability limit for the indemnity. Previously, the indemnification had to be computed in accordance with a statutorily prescribed computational method.
2. Directs that the DOE Secretary to adjust the indemnification at least once every five years.
3. Increases the indemnification for damage outside the United States resulting from a nuclear incident occurring within the United States or involving Federal radiological materials from \$100 million to \$500 million.

In addition, the PAAA 2005 removed from the Secretary the duty to determine whether civil penalties for violation of DOE nuclear safety directives or regulations by other not-for-profit contractors should be remitted. It further removes the exemptions from civil penalties for seven named M & O contractors, their subcontractors, and suppliers, reflected in Note I to the clause at 970.250-70 (June 1996). In place of the named exempt entities, the PAAA 2005 provides that civil penalties for any not-for-profit contractors, subcontractors,

or suppliers, as defined in the PAAA 2005, “may not exceed the total amount of fees paid within any 1-year period (as determined by the Secretary) under the contract under which the violation occurs.” Also, the changes to civil penalties in the PAAA 2005 do not apply to any violation of the AEA occurring under a contract entered into before August 8, 2005, the date of enactment of the PAAA 2005.

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## **20.3 Independent Oversight**

In the late 1980s, public pressure similar to the pressure that caused the earlier AEC split into the NRC and ERDA began to challenge DOE’s self-regulation for their nuclear facilities.

### **20.3.1 National Research Council Reviews (1989)**

After the Chernobyl accident, DOE asked the National Research Council to review the large reactors owned by the DOE to determine if these reactors were vulnerable to large accidents that could have significant radiological consequences. The National Research Council issued several reports that addressed safety, management, and technical issues, but found that the U.S. production reactors did not have comparable design flaws.

Congress then directed DOE to request the National Reach Council to review the health, safety, and environmental issues at the installations that were involved in the manufacture of nuclear weapons. In all there were 17 plants that were included in the review scope. The National Research Council provided their recommendations in *The Nuclear Weapons Complex: Management for Health, Safety, and the Environment (1989) Commission on Physical Sciences, Mathematics, and Applications (CPSMA) (National Research Council Reviews, 1989)*. The Commission’s report addressed many generic issues that dealt with the culture associated with nuclear weapon production. Redirecting the culture to ensure that health, safety, and environmental considerations are integral to making operational decisions was one of the major recommendations.

### **20.3.2 Defense Nuclear Facility Safety Board (DNFSB)**

For nearly half a century, the DOE and its predecessor agencies operated the nation’s defense nuclear weapons complex without independent external oversight. In the late 1980s, it became increasingly clear to members of Congress that significant public health and safety issues had accumulated at many of the aging facilities in the weapons complex. As an outgrowth of these concerns, Congress created the DNFSB in 1988 through Public Law 100–456 as an independent oversight organization within the Executive Branch charged with providing advice and recommendations to the Secretary of Energy “to ensure adequate protection of public health and safety” at DOE’s defense nuclear facilities. Note that this scope does not include DOE’s nondefense nuclear facilities.

The Board is responsible for an independent oversight of all activities affecting nuclear safety within DOE’s nuclear weapons complex. Prior to the end of the nuclear arms race, the nuclear weapons complex concentrated on the design, manufacture, test, and maintenance of the nation’s nuclear arsenal. The complex is now engaged in the cleanup of contaminated sites and facilities, disassembly of nuclear weapons to achieve arms control objectives, maintenance of the smaller stockpile, and storage and disposition of excess fissionable materials. All of these hazardous activities must be carried out in the strict observance of health and safety requirements.

To ensure that these safety requirements are adequate, the Board's enabling statute, 42 U.S.C. § 2286 et seq., requires the Board to review and evaluate the content and implementation of DOE health and safety standards applicable to the design, construction, operation, and decommissioning of defense nuclear facilities. The Board must then recommend to the Secretary, including shutdown and decommissioning phases, any specific measures, such as changes in the content and implementation of those standards that the Board believes should be adopted to ensure that the public health and safety are adequately protected. The Board is also required to review the design of new defense nuclear facilities before construction begins, as well as modifications to older facilities, and to recommend changes necessary to protect health and safety. Review and advisory responsibilities of the Board continue throughout the full life cycle of facilities.

Congress gave the Board a variety of powers to achieve its mission. Primary among these is the power to issue a recommendation to the Secretary of Energy. Although the Secretary is permitted to reject Board recommendations, in practice the Secretary has not chosen to do so over the 17 years of Board operations. In addition to recommendations, the Board may conduct investigations, issue subpoenas, hold public hearings, gather information, conduct studies, and establish reporting requirements for DOE. The Board is required by statute to report to Congress each quarterly (initially annually) concerning its oversight activities, its recommendations to the Secretary of Energy, and improvements in safety achieved at defense nuclear facilities as a result of its activities.

The DNFSB issued more than 40 formal sets of recommendations to the Secretary of Energy on health and safety issues for the DOE defense nuclear facilities, as listed in Table 20.1 (<http://dnfsb.gov/pub.docs/dnfsb/rec.html>). The recommendations fell into two categories: facility-specific recommendations and multifacility or program recommendations. The number of issues resulting in recommendations identified from CY 2000 has been considerably less than the number identified from CY 1991 through CY 1999. All but one of the post-CY 1999 issues addressed generic multisite or program issues. Table 20.1 includes the recommendations through February 2008. Two recommendation letters have not had a formal identification number assigned. The DNFSB recommendations have been instrumental in moving the DOE to issue appropriate standards to ensure nuclear safety.

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## 20.4 DOE Nuclear Safety Regulatory Process Post-1988

In the late 1980s, DOE initiated tasks to provide new DOE orders that addressed nuclear safety documents and nuclear criticality safety. These documents were necessary to establish DOE nuclear safety policy and requirements in anticipation of the change to a regulated process. In 1991–1992, the DOE issued four DOE orders that defined the requirements for preparing Unreviewed Safety Questions (USQs) (DOE Order 5480.21), TSRs (DOE Order 5480.22), Safety Analysis Reports (DOE Order 5480.23), and Nuclear Criticality Safety (DOE Order 5480.24). They were used for the specification of nuclear safety requirements and used in conjunction with 10 CFR, Part 830 (10 CFR 830), Subpart A—*Quality Assurance (QA) Requirements*—for enforcement action under the PAAA regulatory process. These orders were deactivated on the issuance of 10 CFR 830, Subpart B—*Safety Basis Requirements*—and replaced by DOE guides.

**TABLE 20.1**

## DNFSB Recommendations

<b>Number</b>	<b>Recommendation</b>
2008-X	Integration of DOE-STD-1189, <i>Integration of Safety into the Design Process</i>
2008-Y	Independent validation of line management's implementation of new or substantially revised safety basis controls
2008-1	Safety Classification of Fire Protection Systems
2007-1	Safety-Related In Situ Nondestructive Assay of Radioactive Materials
2005-1	Nuclear Material Packaging
0204-2	Active Confinement Systems
2004-1	Oversight of Complex, High-Hazard Nuclear Operations
2002-3	Requirements for the Design, Implementation, and Maintenance of Administrative Controls
2002-2	Weapons Laboratory Support of the Defense Nuclear Complex
2002-1	Quality Assurance for Safety-Related Software
2001-1	High-Level Waste Management at the Savannah River Site
2000-2	Configuration Management, Vital Safety Systems
2000-1	Prioritization for Stabilizing Nuclear Materials
1999-1	Safe Storage of Fissionable Material Called "Pits"
1998-2	Safety Management at the Pantex Plant
1998-1	Resolution of Safety Issues Identified by DOE Internal Oversight
1997-2	Continuation of Criticality Safety
1997-1	Safe Storage of Uranium-233
1996-1	In-Tank Precipitation System at the Savannah River Site
1995-2	Safety Management
1995-1	Improved Safety of Cylinders Containing Depleted Uranium
1994-5	Integration of Rules, Orders, and Other Requirements
1994-4	Deficiencies in Criticality Safety at the Oak Ridge Y-12 Plant
1994-3	Rocky Flats Seismic and Systems Safety
1994-2	Low Level Waste Management
1994-1	Improved Schedule for Remediation in the Defense Nuclear Facilities Complex
1993-6	Maintaining Access to Nuclear Weapons Expertise
1993-5	Hanford Waste Tanks Characterization Studies
1993-4	Deficiencies in Criticality Safety at Oak Ridge Y-12 Plant
1993-3	Improving Technical Capability
1993-2	Criticality Safety
1993-1	Concerning Standards Utilization in Defense Nuclear Facilities
1992-7	Training and Qualification
1992-6	Operational Readiness Reviews (ORRs)
1992-5	Discipline of Operation in a Changing Defense Nuclear Facilities Complex
1992-4	Multifunction Waste Tank Facility at the Hanford Site
1992-3	Operational Readiness of the HB-Line at the Savannah River Site
1992-2	DOE's Facility Representative Program at Defense Nuclear Facilities
1992-1	Operational Readiness of the HB-Line at the Savannah River Site
1991-6	Radiation Protection for Workers and the General Public at DOE Defense Nuclear Facilities
1991-5	Power Limits for K-Reactor Operation at the Savannah River Site
1991-4	DOE's Operational Readiness Review Prior to Resumption of Plutonium Operations at the Rocky Flats Plant

(Continued)

**TABLE 20.1 (Continued)**

## DNFSB Recommendations

Number	Recommendation
1991-3	DOE's Comprehensive Readiness Review Prior to Initiation of the Test Phase at the Waste Isolation Pilot Plant (WIPP)
1991-2	Closure of Safety Issues Prior to Restart of K-Reactor at the Savannah River Site
1991-1	Strengthening the Nuclear Safety Standards Program for DOE's Defense Nuclear Facilities
1990-7	Safety at the Single-Shell Hanford Waste Tanks
1990-6	Criticality Safety at Rocky Flats Plant
1990-5	Systematic Evaluation Program at the Rocky Flats Plant
1990-4	Operational Readiness Review at Rocky Flats Plant
1990-3	Safety at Single-Shell Hanford Waste Tanks
1990-2	Design, Construction, Operation, and Decommissioning Standards at Certain Priority DOE Facilities
1990-1	Operator Training at Savannah River Facilities Prior to Restart of K, L, and P Reactors

**20.4.1 DOE Order 5480.21: Unreviewed Safety Questions**

The concept of the USQs was established to allow contractors to make physical and procedural changes and to conduct test and experiments without prior DOE (DOE Order 5480.21) approval, as long as these changes did not explicitly or implicitly affect the authorization basis of the facility or result in a TSR change. The intent of this order was to provide contractors with the flexibility needed to conduct day-to-day operations and to require that those issues with a potential impact on the authorization basis, and, therefore, the safety of the facility be brought to the attention of DOE, thus maintaining the proper safety focus. The authorization basis is described in documents such as the facility safety analysis report, other safety analyses, hazard classification documents, the TSRs, DOE-issued safety evaluation reports, and facility-specific commitments made in compliance with DOE orders or policies.

DOE policy is that

1. Each nuclear facility develops procedures to implement the USQ review process consistent with the provisions described in this order.
2. Any changes made to a facility that directly or indirectly affect the facility authorization basis, and, therefore, its safety be reviewed in accordance with the provisions of this order.
3. Primary responsibility, authority, and accountability for the direction and management of the USQ process reside with the line management of the facility organization responsible for the design and safety analyses.

**20.4.1.1 DOE G 424.1-1A: Implementation Guide for Use in Addressing Unreviewed Safety Question Requirements**

DOE Guides (DOE G 424.1-1A) are part of the DOE Directives System and are issued to provide supplemental information regarding the Department's requirements as contained in rules, orders, notices, and standards. Guides also provide acceptable methods for implementing these requirements.

This guide may be used by all contractors, including contractors for National Nuclear Security Administration (NNSA), for DOE Hazard Category 1, 2, or 3 nuclear facilities.

This guide was developed to support DOE Order 5480.21 USQs. The guide has been revised to provide guidance for the requirements defined in 10 CFR 830, Subpart B, §203 (10 CFR 830.203)—*Unreviewed Safety Question Process*. This guide imposes no requirements.

#### **20.4.2 DOE Order 5480.22: Technical Safety Requirements**

The purpose of DOE Order 5480.22 was to clearly state the requirement to have TSRs prepared for DOE nuclear facilities and to delineate the criteria, content, scope, format, approval process, and reporting requirements of these documents and revisions thereof.

It is DOE policy that nuclear facilities operate within Program Secretarial Officer (PSO) approved TSRs that prescribe the bounds for safe operation of these facilities to protect the health and safety of the public and reduce risk to workers.

TSR means the conditions, safe boundaries, and the management, or administrative controls (ACs) necessary to ensure the safe operation of a nuclear facility and to reduce the potential risk to the public and facility workers from uncontrolled releases of radioactive materials or from radiation exposures due to inadvertent criticality. A TSR consists of safety limits, operating limits, surveillance requirements, ACs, use and application instructions, and the basis thereof.

Section 10 CFR 830.205 of the Nuclear Safety Management rule, requires DOE contractors responsible for Category 1, 2, or 3 DOE nuclear facilities develop TSRs. These TSRs identify the limitations to each government-owned contractor-operated (GOCO) nuclear facility based on the DSA and any additional safety requirements established for the facility. Although not required by the Rule 830, there also may be a need to establish TSRs for the safe operation of radiological facilities (DOE Order 5480.22).

##### **20.4.2.1 DOE G 423.1-1: Implementation Guide for Use in Developing TSRs**

This guide was developed to support DOE Order 5480.22 TSRs. It provides guidance for the requirements defined in 10 CFR 830.203—*Unreviewed Safety Question Process*.

The TSR rule requires contractors to prepare and submit TSRs for DOE approval. This guide provides guidance in identifying important safety parameters and developing the content for the TSRs that are required by 10 CFR 830.205.

The appendix to Subpart B of the Nuclear Safety Management rule specifies the types of safety limits, operating limits, surveillance requirements, and ACs that define the safety envelope necessary to protect the health and safety of the public and workers. The TSR derivation chapter in the DSA is the key component that provides the basis for TSRs.

This guide provides elaboration for the content of TSRs. In providing this guidance, it is recognized that the diversity of DOE facilities may necessitate varying degrees of emphasis to be placed on some of the TSR sections, but the following guidance is intended to be generally applicable. This guide imposes no requirements (DOE G 423.1-1).

#### **20.4.3 DOE Order 5480.23: Nuclear Safety Analysis Reports**

The purpose of DOE Order 5480.23 was to establish requirements for contractors responsible for the design, construction, operation, decontamination, or decommissioning of nuclear facilities to use for developing safety analyses that established and evaluated the adequacy of the safety bases of the facilities.

It is DOE policy that nuclear facilities and operations be analyzed to identify all hazard and potential accidents associated with the facility and the process systems, components,

equipment, or structures, and to establish design and operational means to mitigate these hazards and potential accidents. The results of these analyses are to be documented in DSAs. The identified hazard and safety analyses are to be approved by DOE (DOE Order 5480.23).

DOE has developed several DOE standards and guides that prescribe acceptable processes for developing DSA, TSRs, and USQ determinations. The standards are referenced as a “Safe Harbor” methodology in 10 CFR 830, Subpart B—*Safety Basis Requirements*. These standards and guides include the following.

#### **20.4.3.1 DOE-STD-1027: Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports**

The purpose of this DOE-STD-1027-92 was to establish guidance for the preparation and review of hazard categorization and accident analysis techniques as required in DOE Order 5480.23—*Nuclear Safety Analysis Reports*. As issued, DOE Order 5480.23 required further guidance to ensure consistency across all nuclear facilities within the DOE complex. This standard is identified as the safe harbor methodology for determining the hazard category of a facility. The hazard category determines the safe harbor methodology, the formality of the hazard and accident analysis, and the extent of required detail. A facility that has a hazard categorization of less than “3” is not subject to 10 CFR 830, Subpart B.

The DOE Standard DOE-STD-1027 imposes no new requirements on nuclear facilities. Instead, it focuses on the definition of the standard identifying nuclear facilities required to have DSAs to comply with 10 CFR 830.

1. The hazard categorization methodology to be applied to all facilities
2. The accident analysis techniques appropriate for the graded approach addressed in the order

DOE Order 5480.23 and its attached guidance document provide directions on the use of the graded approach. This report is intended not to supersede that direction but to supplement and clarify it. Methods other than those suggested in this guide may be considered for applying the graded approach, but they must be justified whenever grading is applied.

#### **20.4.3.2 DOE-STD-3009: Preparation Guide for the U.S. Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses**

This Standard (DOE-STD-3009) describes a DSA preparation method that is acceptable to the DOE as delineated for those specific facilities listed in Table 20.2. This table is from Appendix A, “General Statement of Safety Basis Policy,” to Subpart B, “Safety Basis Requirements,” of 10 CFR 830. It was developed to assist Hazard Category 2 and 3 facilities in preparing DSAs that will satisfy the requirements of 10 CFR 830. Hazard Category 1 facilities are typically expected to be Category A reactors for which extensive precedents for SARs already exist.

Guidance provided by this Standard is generally applicable to any facility required to document its safety basis in accordance with 10 CFR 830. For new facilities in which conceptual design or construction activities are in progress (i.e., Preliminary Documented Safety Analysis [PDSAs]) elements of this guidance may be more appropriately handled as an integral part of the overall design requirements process (e.g., preliminary design to design criteria). The methodology provided by this standard focuses more on characterizing



**TABLE 20.2**

## DSA Safety Harbor Methodologies

Reactors	Nuclear regulatory commission (NRC) regulatory guide 1.70, standard format and content of safety analysis reports for nuclear power plants
Nonreactor nuclear facilities	DOE STD 3009–2000, <i>Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports</i> .
Nuclear facilities with limited operational life	DOE STD 3009–2000, <i>Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports</i> , or DOE STD 3011–94, <i>Guidance for Preparation of DOE 5480.22 (TSR) and DOE 5480.23 (SAR) Implementation Plans</i> .
Nuclear facilities in deactivation or the transition surveillance and maintenance	DOE STD 3009–2000, <i>Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports</i> ; or DOE STD 3011–94, <i>Guidance for Preparation of DOE 5480.22 (TSR) and DOE 5480.23 (SAR) Implementation Plans</i> .
The decommissioning of a DOE nuclear facility	DOE STD 3009–2000, <i>Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports</i> ; or DOE STD 3011–94, <i>Guidance for Preparation of DOE 5480.22 (TSR) and DOE 5480.23 (SAR) Implementation Plans</i> . Deriving hazard controls based on the safety and health programs, the Work Plans, the Health and Safety Plans, and the Emergency Response Plans.
Environmental restoration activities	Using the method in DOE-STD–1120–98 or successor document, and Using the provisions in 29 CFR 1910.120 (or 29 CFR 1926.65 for construction activities) for developing a Safety and Health Program and a site-specific Health and Safety Plan (including elements for Emergency Response Plans, conduct of operations, training and qualifications, and maintenance management.
A DOE nuclear explosive	Developing its documented safety analysis in two pieces: A Safety analysis report for the nuclear facility that considers the generic nuclear explosive operations and is prepared in accordance with DOE-STD–3009, Change Notice No. 1, January 2000, or successor document, and A Hazard Analysis Report for the specific nuclear explosive operations prepared in accordance with DOE-STD–3016–99, Hazards Analysis Reports for Nuclear Explosive Operations, February 1999, or successor document.
A DOE Hazard Category 3 nonreactor nuclear facility	Using the methods in Chapters 2–5 of DOE-STD–3009, Change Notice No. 1, January 2000: The basic description of the facility/activity and its operations, including safety structures, systems, and components. A qualitative hazards analysis, and The hazard controls (consisting primarily of inventory).
Transportation activities	Preparing a Safety Analysis Report for Packaging in accordance with DOE-O–460.1A, <i>Packaging and Transportation Safety</i> , October 2, 1996, and Preparing a Transportation Safety Document in accordance with DOE-G–460.1–1, <i>Implementation Guide for Use with DOE O 460.1A, Packaging</i> .
Transportation and onsite transfer of nuclear explosives, nuclear components, Naval nuclear fuel elements, Category I and Category II special nuclear materials, special assemblies, and other materials of national security.	Preparing a Safety Analysis Report for Packaging in accordance with DOE-O–461.1, <i>Packaging and Transportation of Materials of National Security Interest</i> , September 29, 2000, or successor document, and Preparing a Transportation Safety Document in accordance with DOE-M–461.1–1, <i>Packaging and Transfer of Materials of National Security Interest</i> .
Alternate methodologies for DSAs	A Contractor may also propose “alternate methods” that are considered being more effective than the safe harbor methods to prepare the DSAs. DOE approval for the specific circumstances for which you will use the alternate method must be obtained in advance.

facility safety (i.e., back-end approach) with or without well-documented information than on the determination of facility design (i.e., front-end approach). Accordingly, contractors for facilities that are documenting conceptual designs for PDSAs should apply the process and format of this standard to the extent it is judged to be of benefit.

Beyond conceptual design and construction, the methodology in this standard is applicable to the spectrum of missions expected to occur over the lifetime of a facility (e.g., production, shutdown/standby, decontamination, and decommissioning). As the phases of facility life change, a suitable methodology is provided for use in updating an existing DSA and in developing a new DSA if the new mission is no longer adequately encompassed by the existing DSA (e.g., a change from production operations to decontamination and decommissioning). This integration of the DSA with changes in facility mission and associated updates should be controlled as part of an overall safety management plan.

For facilities transitioning into D&D, the safety basis of the D&D operations is documented throughout a DSA. This DSA, of which the principal emphasis is on the D&D operations themselves, provides the necessary analysis and supporting information to describe the facilities as they undergo shutdown, deactivation, decontamination, and decommissioning or mantlement. The facility consists of the physical building, its constituent components, and the actual processes of D&D being performed. Physical buildings and constituent components targeted for D&D are briefly described in Chapter 2. Detailed descriptions are reserved for the actual D&D processes, which are the focus of evaluation in Chapters 3 and 4 for each stage of major configuration change. Also included are the temporary engineering and ACs used to maintain the safety basis. This description and evaluation would envelop major configurations during the D&D operations for which the authorization basis is sought. This is consistent with the intent of DSAs for operating facilities where all operations conducted are not detailed in the DSA.

#### **20.4.3.3 DOE-HDBK-3010: Airborne Release Fractions (ARFs)/Rates and Respirable Fractions (RFs) for Nonreactor Nuclear Facilities**

The purpose of this document is to provide a compendium and analysis of experimental data from which ARFs and RFs may be derived (DOE-HDBK-3010). Such values are needed to determine quantities of radioactive material driven airborne for the purpose of estimating the scope of the potential release spectrum and potential downwind consequences from a given facility or activity. The information provided in this handbook aids in making such estimates. The handbook discusses the following major topics:

1. Source term formula: Provides a computational formula for using this information.
2. Applicability of data: Distinguishes the proper use of information.
3. Accident stresses: Identifies the types of accident conditions for which this information is applicable.
3. Handbook organization: Explains the presentation of information and use of examples.

The data in this handbook can be used in a variety of applications, such as safety and environmental analyses, and to provide information relevant to system and experiment design. However, these data and the analyses of the data contained herein need to be critically evaluated for applicability in each situation in which they are used, and they represent only one source of information in a complete safety analysis or design process.

#### **20.4.3.4 DOE-STD-3011: Guidance for Preparation of Basis for Interim Operation (BIO) Documents**

This Standard (DOE-STD-3011) provides a DOE-approved methodology for preparing a BIO document. The BIO document is an acceptable form of DSA in accordance with Table 20.2.

A BIO developed under DOE Order 5480.23 was supposed to be a bridge between safety documentation that existed prior to the order and a DOE-STD-3011-2002-compliant SAR. It was not expected to be fully compliant with the order. However, a BIO under 10 CFR 830 must fully satisfy the requirements of 10 CFR 830.202 and 830.204.

The methodology in this standard has been determined to be acceptable for the following types of Hazard Category 1, 2, and 3 nuclear facilities and activities:

1. Nuclear facility with a limited operational life (five years or less)
2. The deactivation of a nuclear facility
3. The transition surveillance and maintenance of a nuclear facility

#### **20.4.3.5 DOE-STD-1120-2005: Integration of Environment, Safety, and Health into Facility Disposition Activities**

DOE-STD-1120 provides guidance for preparing a DSA for decommissioning of nuclear facilities, as well as environmental restoration activities that involve work not done within a permanent structure (DOE-STD-1120-2005). Methodologies provided in this standard are compliant with 10 CFR 830, Subpart B—*Safety Basis Requirements*. The standard is written to make the maximum utilization of the provisions of 29 CFR 1910.120 or 29 CFR 1926.65—*Hazardous Waste Operations and Emergency Response* (HAZWOPER). Derivation of controls is also necessary for facility decommissioning projects that involve more than “low level residual fixed radioactivity.”

As stated in DOE-STD-3009-94, a preparation guide for U.S. DOE Nonreactor Documented Safety Analysis reports, “It is not the intention of the DSA to cover safety as it relates to the common industrial hazards that make up a large portion of basic OSHA regulatory compliance.” Therefore, in the context of 10 CFR 830, Subpart B requirements, the scope of HAZWOPER is taken to include those hazards, associated controls, and safety and health (S&H) programs that must be identified and maintained within a Hazard Category 1, 2, or 3 facility’s safety basis.

Existing S&H programs that are in place to meet DOE directives and standards, as implemented through the Integrated Safety Management (ISM) process, may be acceptable mechanisms for meeting HAZWOPER S&H program requirements. Where applicable to the safety basis, these programs should be described in the DSA.

#### **20.4.3.6 DOE G 421.1-2: Implementation Guide for Use in Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830**

Title 10 CFR 830, Subpart B—*Safety Basis Requirements*—requires the contractor responsible for a DOE nuclear facility to analyze the facility, the work to be performed, and the associated hazards and to identify the conditions, safe boundaries, and hazard controls necessary to protect workers, the public, and the environment from adverse consequences (DOEG 421.1-2). These analyses and hazard controls constitute the safety basis on which the contractor and DOE rely to conclude that the facility can be operated safely.

Performing work consistent with the safety basis provides reasonable assurance of adequate protection of workers, public, and the environment. This guide elaborates on the DSA development process and the safe harbor provisions of the Appendix to 10 CFR 830, Subpart B.

#### **20.4.4 DOE Order 5480.24: Nuclear Criticality Safety**

DOE Order 5480.24 defined the criteria that were to be used to develop the criticality safety program. In general, the order referenced the Accreditation process for developers of American National Standards (ANS/ANSI) as the criteria for an adequate nuclear criticality safety program. Two levels of verification of acceptable nuclear risks were identified. The first is the “double contingency” process. To be double contingent means that two unlikely process events are required before a criticality is possible.

1. Criticality safety is comprehensively addressed and receives an objective review, with all identifiable risks reduced to acceptably low levels, and the management authorization of the operation is documented.
2. The public, workers, Government property, and essential operations are protected from the effects of a criticality incident.

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### **20.5 10 CFR 830 Regulation**

Congress directed DOE in the PAAA of 1988 that its methods of managing nuclear activities at contractor-operated sites be modified. DOE was directed to develop a process so that it may undertake enforcement actions against indemnified contractors for the violations of nuclear safety requirements. The establishment of enforcement sanctions as a method of ensuring compliance with safety requirements is intended to minimize the risk to workers and the public (Code of Federal Regulation, Part 820).

DOE issued 10 CFR 820 Procedural Rules for DOE Nuclear Activities (23) effective September 16, 1993. These procedural rules are used in applying its substantive regulations and orders relating to nuclear safety. These procedural rules are intended to be an essential part of the framework through which DOE deals with its contractors, subcontractors, and suppliers to ensure that its nuclear facilities are operated in a manner that protects public and worker safety and the environment. In particular, this part sets forth the procedures to implement the provisions of the PAAA of 1988 which subjects DOE contractors to potential civil and criminal penalties for violations of DOE rules, regulations, and orders relating to nuclear safety (DOE Nuclear Safety Requirements).

#### **20.5.1 10 CFR 830 Subpart A: QA Requirements**

This subpart established the QA requirements for contractors conducting activities, including providing items or services that affect, or may affect, nuclear safety of DOE nuclear facilities (10 CFR 830.120). This subpart became effective January 10, 2001. Subpart A was used for most of the PAAA fines (except for 10 CFR 835—*Radiation Protection* (25)) that have been levied to date. Almost any serious event or noncompliance can be linked to a

deficiency in the QA program (QAP), for example, inadequate design reviews, procedures, work packages, and training (Code of Federal Regulation, Part 830 Subpart A; Part 835).

The QA criteria are identified in 10 CFR 830.122—*Quality Assurance Criterion*—and further discussed in DOE G 414.2A and DOE O 414.1C.

#### **20.5.1.1 DOE G 414.1-2A: QA Management System Guide for Use with 10 CFR 830, Subpart A, QA Requirements**

Contractors were required to submit a QAP and to update it annually as required to keep it current. The QAP must address the following management, performance, and assessment criteria (DOE G 414.1-2A):

- Criterion 1—Management/Program
- Criterion 2—Management/Personnel Training and Qualification
- Criterion 3—Management/Quality Improvement
- Criterion 4—Management/Documents and Records
- Criterion 5—Performance/Work Processes
- Criterion 6—Performance/Design
- Criterion 7—Performance/Procurement
- Criterion 8—Performance/Inspection and Acceptance Testing
- Criterion 9—Assessment/Management Assessment
- Criterion 10—Assessment/Independent Assessment

This guide adds no requirements.

#### **20.5.1.2 DOE O 414.1C: QA**

The objectives of this order (DOE 0414.1C) are

1. To ensure that DOE, including NNSA, products, and services meet or exceed customers' expectations.
2. To achieve QA for all work based on the following principles.
3. That quality is assured and maintained through a single, integrated, effective QA program (i.e., management system).
4. That management support for planning, organization, resources, direction, and control is essential to QA.
5. That performance and quality improvement require thorough, rigorous assessment and corrective action.
6. That workers are responsible for achieving and maintaining quality.
7. That environmental, safety, and health risks and impacts associated with work processes can be minimized while maximizing the reliability and performance of work products. To establish QA process requirements to be implemented under a QAP for the control of suspect/counterfeit items (S/CIs), safety issue corrective actions, and safety software.

This Guide imposes no requirements.

### 20.5.2 10 CFR 830, Subpart B: Safety Basis Requirements

Subpart B of 10 CFR Part 830—*Nuclear Safety Management*—requires contractors and operators of Hazard Categories 1, 2, and 3 DOE nuclear facilities to develop and maintain a safety basis and to perform work in accordance with that safety basis. The major components of the safety basis for a nuclear facility include the DSA and the TSRs.

The DSA is the safety analysis for a Hazard Category 1, 2, or 3 DOE nuclear facility. Section 830.204 of 10 CFR, Part 830, requires that DSAs and TSRs be developed for these facilities. The DSA is the safety analysis for a Hazard Category 1, 2, or 3 DOE nuclear facility. Section 830.204 of 10 CFR, Part 830, requires that DSAs (Code of Federal Regulation, Part 830, Subpart B):

1. Describe the facility and the work to be performed; and the facility
2. Categorize the facility in accordance with DOE-STD-1027
3. Evaluate all accident conditions that are presented by natural and/or manmade hazards
4. Derive the hazard controls, including TSRs, to eliminate, limit, or mitigate identified hazards, and define the process for maintaining the hazard controls current at all times and controlling their use
5. Define the characteristics of the safety management programs necessary to ensure the safe operation of the facility, including a criticality safety program, where applicable. Depending on the type of nuclear facility and where it is in its life cycle, the DSA format will typically be one or more of the following:
  - a. SAR
  - b. BIO
  - c. HSP (Health and Safety Plan)
  - d. HAR (Hazard Analysis Report)

DOE contractors for Hazard Category 1, 2, and 3 DOE nuclear facilities must use DOE-approved methods to develop their DSAs. These methods will be either a safe harbor methodology or an approved alternate.

Safe harbor methodologies are methods for developing a DSA that have already been approved by DOE for use in the specific circumstances described. Table 20.2 lists the safe harbor methodologies. These methodologies will allow the development of a DSA by a method that is appropriately graded for the work and the hazards.

The safe harbor methodologies are methods identified in standards developed by DOE or the NRC, or defined in regulations promulgated by the Occupational, Safety, and Health Administration (OSHA). These standards are based on many years of experience with the types of facilities and activities to which they may be applied. Prior DOE approval to use the safe harbor methods in accordance with the stated provisions in 10 CFR 830 is not required. DOE approval is required to use a method other than the safe harbor method. In addition, whether a safe harbor method is used or not used to develop the DSA, DOE approval is required for the final DSA. The rule, standards, and guide should be consulted for the specific provisions and limitations for each safe harbor method.

In cases where two or more safe harbors are applicable, the contractor should choose the method that is cost-effective for the particular facility. Note that some of these categories overlap, for instance.

The Safety Harbor methods are listed in Table 2 of Appendix A to Subpart B of Part 830 (Table 20.2 of this document). Note that successor documents may be used where applicable. A number of the DOE standards are discussed in Section 20.4, DOE Nuclear Safety Regulatory Process Post 1988 of this document.

### **20.5.3 DOE PAAA Enforcement of Nuclear Safety Requirements**

The purpose of DOE's enforcement program is to enhance and protect the radiological safety of the public, workers at DOE facilities, through a process that encourages the effective understanding and proper implementation of nuclear safety requirements; critical self-assessment of activities; and the timely identification, open and prompt reporting, and prompt, aggressive correction of noncompliance conditions by DOE contractors.

The Office of Price-Anderson Enforcement implements the congressionally mandated nuclear safety enforcement program in accordance with 10 CFR 820. The enforcement process provides fine mitigation for violations of nuclear safety requirements that are promptly reported to the Office of Enforcement and for which prompt corrective and effective actions are taken.

DOE has issued several documents that define the enforcement process. These are

#### ***20.5.3.1 DOE-HDBK-1085: DOE Enforcement Program Roles and Responsibilities Guidance Handbook***

This Handbook (DOE-HDBK-1085-95) provides detailed guidance and procedures to implement the General Statement of DOE Enforcement Policy as described in 10 CFR 820. The guidance provided in this Handbook is qualified, however, by the admonishment to exercise discretion in determining the proper disposition of each potential enforcement action. The Enforcement and Investigation Staff will apply a number of factors in assessing each potential enforcement situation. The purpose of the DOE enforcement policy is to improve nuclear safety for the workers and the public.

#### ***20.5.3.2 Operational Procedure, Identifying, Reporting, and Tracking Nuclear Safety Noncompliances (June 1988)***

This operational procedure is one of a series prepared by the Office of Enforcement and Investigation to provide information on the conduct of the DOE's statutory nuclear safety enforcement program. This document describes DOE's philosophy on effective identification and reporting of nuclear safety noncompliances and identifies threshold reporting criteria that will be acceptable in evaluating noncompliances reported in DOE's Noncompliance Tracking System. The criteria are derived from portions of DOE Order 232.1A, Occurrence Reporting and Processing of Operations Information, and its attendant Manual (<http://tis.eh.doe.gov/enforce/>).

#### ***20.5.3.3 Operational Procedure, Enforcement of DOE Nuclear Safety Requirements under PAAA of 1988 (June 1988)***

This operational procedure (DOE Nuclear Safety Requirements, 1988)

1. Defines when Enforcement Letters may be used instead of a Notice of Violation
2. States that mitigation for identification and reporting will generally not be given for self-disclosing events, or when timely effort to restore compliance is not taken

3. Identifies considerations that may be made when determining whether to exercise discretion in the application of a civil penalty or reduction in severity
4. Clarifies the types of violations that normally fall into the different severity level classifications and the circumstances when a severity level III violation may warrant a civil penalty
5. Identifies DOE's authority to apply the base civil penalty on a per-day basis, particularly when the violation is continuing and its significance warrants more than the single day base civil penalty
6. States that a Final Notice of Violation will not be issued if the Preliminary Notice is not contested
7. Defines the circumstances that would be considered "willful" in evaluating a violation

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## 20.6 Development of a DOE-STD-3009 DSA

Developing a DSA requires that an inventory of potential hazards within the facility be developed. One method of identifying potential hazards is to use a checklist. Normally, a team comprised of engineering, operations, and safety would do a walk down of the facility. Many sites have an inventory system that identifies chemicals and their quantities. This inventory system should be queried to determine what chemicals are in the facility. Radioactive material inventory and sealed source inventory systems should also be used to determine their inventories. A typical hazard identification checklist is provided in Table 20.3. Dispositions are either identified as a Standard Industrial Hazard, screened out due to limited quantities of hazardous material, or are carried forth as an initiator of an event or the hazard from an event. The criteria identified in Table 20.4 are used to determine which hazards require further review.

After the hazards have been identified, the potential events that are initiated by the hazard or have the potential during an event to cause harm to the onsite worker or members of the public are identified. Table 20.5 provides an example of how the hazards and events are identified for a typical nuclear facility.

The next task is to bin the identified events into frequency bins. Table 20.6 shows the frequency bins used for the safety analysis of the DOE Nuclear facility accident analysis.

Once the events have been binned, the next step is to estimate the frequency of an event occurring and the consequence level of the event on the facility worker, on-site worker, and the public, using Table 20.7.

Once the frequency and the consequence of the events are determined, the risk level can be developed by using Table 20.8.

The frequency and consequence information is then used to complete Table 20.9. From Table 20.8, the controls that are needed to appropriately prevent or mitigate events that impact workers are selected. Most of the controls will only require the mitigation or prevention that is provided by the established safety management programs that are integral to the ISM program.



**TABLE 20.3**

Example of a Hazard Identification Table

<b>Item</b>	<b>Hazard Energy Source or Material</b>	<b>Exists (Y/N)</b>	<b>Description</b>	<b>Disposition</b>
<b>1.0</b>	<b>Electrical</b>			
1.1	Battery banks			
1.2	Cable runs			
1.3	Diesel generators			
1.4	Electrical equipment			
1.5	Heaters			
1.6	High voltage (>600 V)			
1.7	Locomotive, electrical			
1.8	Motors			
1.9	Power tools			
1.10	Pumps			
1.11	Service outlets, fittings			
1.12	Switchgear			
1.13	Transformers			
1.14	Transmission lines			
1.15	Wiring/underground wiring			
1.16	Other			
<b>2.0</b>	<b>Thermal</b>			
2.1	Boilers			
2.2	Bunsen burner/hot plates			
2.3	Electrical equipment			
2.4	Electrical wiring			
2.5	Engine exhaust			
2.6	Furnaces			
2.7	Heaters			
2.8	Lasers			
2.9	Steam lines			
2.10	Welding surfaces			
2.11	Welding torch			
2.12	Other			
<b>3.0</b>	<b>Pyrophoric material</b>			
3.1	Pu and U metal			
3.2	Other			
<b>4.0</b>	<b>Spontaneous combustion</b>			
4.1	Cleaning/decontamination solvents			
4.2	Fuels (gasoline, diesel fuel)			
4.3	Grease			
4.4	Nitric acid and organics			
4.5	Paint solvents			
4.6	Other			
<b>5.0</b>	<b>Open flame</b>			
5.1	Bunsen burners			
5.2	Welding/cutting torches			
5.3	Other			

(Continued)

**TABLE 20.3 (Continued)**

Example of a Hazard Identification Table

Item	Hazard Energy Source or Material	Exists (Y/N)	Description	Disposition
<b>6.0</b>	<b>Flammables</b>			
6.1	Cleaning/decontamination solvents			
6.2	Flammable gases			
6.3	Flammable liquids			
6.4	Gasoline			
6.5	Natural Gas			
6.6	Paint/paint solvent			
6.7	Propane			
6.8	Spray paint			
6.9	Other			
<b>7.0</b>	<b>Combustibles</b>			
7.1	Paper/wood products			
7.2	Petroleum based products			
7.3	Plastics			
7.4	Other			
<b>8.0</b>	<b>Chemical reactions</b>			
8.1	Concentration			
8.2	Disassociation			
8.3	Exothermic			
8.4	Incompatible chemical mixing			
8.5	Uncontrolled chemical reactions			
<b>9.0</b>	<b>Explosive material</b>			
9.1	Caps			
9.2	Dusts			
9.3	Dynamite			
9.4	Electric squibs			
9.5	Explosive chemicals			
9.6	Explosive gases			
9.7	Hydrogen			
9.8	Hydrogen (batteries)			
9.9	Nitrates			
9.10	Peroxides			
9.11	Primer cord			
9.12	Propane			
9.13	Other			
<b>10.0</b>	<b>Kinetic (linear and rotational)</b>			
10.1	Acceleration/deceleration			
10.2	Bearings			
10.3	Belts			
10.4	Carts/dollies			
10.5	Centrifuges			
10.6	Crane loads (in motion)			
10.7	Drills			
10.8	Fans			
10.9	Firearm discharge			

(Continued)

**TABLE 20.3 (Continued)**

Example of a Hazard Identification Table

Item	Hazard Energy Source or Material	Exists (Y/N)	Description	Disposition
10.10	Fork lifts			
10.11	Gears			
10.12	Grinders			
10.13	Motors			
10.14	Power tools			
10.15	Pressers/shears			
10.16	Rail cars			
10.17	Saws			
10.18	Vehicles			
10.19	Vibration			
10.20	Other			
<b>11.0</b>	<b>Potential (pressure)</b>			
11.1	Autoclaves			
11.2	Boilers			
11.3	Coiled springs			
11.4	Furnaces			
11.5	Gas bottles			
11.6	Gas receivers			
11.7	Pressure vessels			
11.8	Pressurized system (e.g., air)			
11.9	Steam headers and lines			
11.10	Stressed members			
<b>12.0</b>	<b>Potential (height/mass)</b>			
12.1	Cranes/hoists			
12.2	Elevated doors			
12.3	Elevated work surfaces			
12.4	Elevators			
12.5	Lifts			
12.6	Loading docks			
12.7	Mezzanines			
12.8	Floor pits			
12.9	Scaffolds and ladders			
12.10	Stacked material			
12.11	Stairs			
12.12	Other			
<b>13.0</b>	<b>Internal flooding sources</b>			
13.1	Domestic water			
13.2	Fire suppression piping			
13.3	Process water			
13.4	Other			
<b>14.0</b>	<b>Physical</b>			
14.1	Sharp edges or points			
14.2	Pinch points			
14.3	Confined space			
14.4	Tripping			

(Continued)

**TABLE 20.3 (Continued)**

Example of a Hazard Identification Table

Item	Hazard Energy Source or Material	Exists (Y/N)	Description	Disposition
<b>15.0</b>	<b>Radiological material</b>			
15.1	Radiological material			
<b>16.0</b>	<b>Hazardous material</b>			
16.1	Asphyxiants			
16.2	Bacteria/viruses			
16.3	Beryllium and compounds			
16.4	Biologicals			
16.5	Carcinogens			
16.6	Chlorine and compounds			
16.7	Corrosives			
16.8	Decontamination solutions			
16.9	Dusts and particles			
16.10	Fluorides			
16.11	Hydrides			
16.12	Lead			
16.13	Oxidizers			
16.14	Poisons (herbicides, insecticides)			
16.15	Other			
<b>17.0</b>	<b>Ionizing radiation sources</b>			
17.1	Contamination			
17.2	Electron beams			
17.3	Radioactive material			
17.4	Radioactive sources			
17.5	Radiography equipment			
17.6	X-ray machines			
17.7	Other			
<b>18.0</b>	<b>Nonionizing radiation</b>			
18.1	Lasers			
18.2	Other			
<b>19.0</b>	<b>Criticality</b>			
19.1	Fissile material			
<b>20.0</b>	<b>Nonfacility events</b>			
20.1	Aircraft crash			
20.2	Explosion			
20.3	Fire			
20.4	Power outage			
20.5	Transportation accident			
20.6	Other			
<b>21.0</b>	<b>Vehicles in motion</b>			
21.1	Airplane			
21.2	Crane/hoist			
21.3	Forklifts			
21.4	Heavy construction equipment			
21.5	Helicopter			

(Continued)

**TABLE 20.3 (Continued)**

Example of a Hazard Identification Table

Item	Hazard Energy Source or Material	Exists (Y/N)	Description	Disposition
21.6	Train			
21.7	Truck/car			
<b>22.0</b>	<b>Natural phenomena</b>			
22.1	Earthquake			
22.2	Flood			
22.3	Lightning			
22.4	Rain/hail			
22.5	Snow/freezing weather			
22.6	Straight wind			
22.7	Tornado			

Source: BJC/OR-1112 Facility Hazard Categorization/Classification and Hazard Analysis Application Guide, Revision 4, January 2007 (32). Prepared for the U.S. Department of Energy.

**TABLE 20.4**

Standard Industrial Hazard Screening Criteria

Hazard	Criteria/Measure	Guidance
Radiological Hazards	DOE-STD-3009	RQs from 40 CFR 302.4, Appendix B as discussed in this
Criticality	DOE-STD-3009	Exemption criteria are defined by the BJC NCS program, BJCNS-1003
Nonradiological Chemical and Toxicological Hazards	DOE-STD-3009	Any chemical and toxicological hazard that requires additional assessment as identified by an Emergency Management Hazard Survey. As an alternate the Reportable Quantities (RQs) contained in 10 CFR 302.4 may be used. See the additional discussion in this appendix.
X-ray equipment	Does not meet ANSI X-ray standards	Applicable national codes and standards (e.g., ANS N537/NBS123)
Flammable Materials	N/A	Considered as a contributor/initiator for fire events
Reactive material	N/A	Screened according to RQ, TQ, TPQ screening values or SARA #00-26
Chemical compatibility	N/A	Screened according to RQ, TQ, TPQ screening values or SARA #00-26
Lasers	Class III nonenclosed beam Class IV	ANSI Z136.1 <i>Safe Use of Lasers Classifies Lasers in Classes I through IV</i>
Electrical	>600 V or >600 V and >24 mA or >50 J stored energy at 600 V	NEC identifies these as systems requiring special considerations
Kinetic energy	“Unique or Unusual” high kinetic energy sources (e.g., high energy flywheels, large centrifuges)	Many high kinetic energy systems are capable of causing personnel injury. Most of these (e.g., cars, trucks, forklifts, cranes) are SIHs unless an initiator for another significant event. Unique systems (e.g., high energy flywheels, large centrifuges) are not considered SIHs
Pressure	Stored energy > 0.1 lb TNT(1.4 × 10 <sup>5</sup> ft-lb) or Pressure >3000 psig	High hydraulic pressures and pressurized gas bottles are SIHs. Large volumes of compressed gases are not routine
Temperature	Temperatures that could act as an initiator	High temperature systems are SIHs but an evaluation is required if the temperature could result in an overpressure, creation of toxic products or cause a fire.

(Continued)

**TABLE 20.4 (Continued)**

## Standard Industrial Hazard Screening Criteria

Hazard	Criteria/Measure	Guidance
Biohazards	As identified by Health Physic or Industrial Hygiene organization	NA
Asphyxiants	Oxygen content less than 18%	Asphyxiants do not have TLV and cannot be handled as toxic material. Consider areas that could entrap asphyxiants and areas storing cylinders of asphyxiants

Source: BJC/OR-1112 Facility Hazard Categorization/Classification and Hazard Analysis Application Guide, Revision 4, January 2007. Prepared for the U.S. Department of Energy.

Notes: ANS= American Nuclear Society, ANSI= American National Standard Institute, N/A= Not Applicable, NEC= National Electric Code, RQ= Reportable Quantity, SIH= Standard Industrial Hazard, TLV= Threshold Limit Value, TPQ= Threshold Planning Quantity, TQ= Threshold Quantity.

**TABLE 20.5**

## Hazard Sources and Potential Event Types

Hazard Energy Source or Material Group	Potential Events by Category
Electrical	E-1: Fire—In combination with combustible/flammable material E-2: Explosion—In combination with explosive material E-4: Direct exposure—Worker injury
Thermal	E-1: Fire—In combination with combustible/flammable material E-2: Explosion—In combination with explosive material E-4: Direct exposure—Worker injury E-5: Criticality—Increased concentration
Pyrophoric material	E-1: Fire—Pyrophoric; may serve as ignition source for larger fires E-2: Explosion—In combination with explosive material
Spontaneous combustion	E-1: Fire—May serve as ignition source for larger fires E-2: Explosion—In combination with explosive material
Open flame	E-1: Fire—In combination with combustible/flammable material E-2: Explosion—In combination with explosive material E-4: Direct exposure—Worker injury
Flammables	E-1: Fire—In combination with ignition source
Combustibles	E-1: Fire—In combination with ignition source
Explosive material	E-2: Explosion—In combination with ignition source E-3: Loss of confinement—Missiles (in combination with ignition source) E-5: Criticality—Loss of configuration or spacing
Chemical reactions	E-1: Fire—Fire or other thermal effect E-2: Explosion—Explosion or over-pressurization E-3: Loss of confinement—Toxic gas generation E-5: Criticality—Increased concentration, precipitation of material
Kinetic (Linear and rotational)	E-3: Loss of confinement—Impacts, acceleration/deceleration, missiles E-4: Direct exposure—Worker injury E-5: Criticality—Loss of configuration or spacing

(Continued)

**TABLE 20.5 (Continued)**

Hazard Sources and Potential Event Types

Hazard Energy Source or Material Group	Potential Events by Category
Potential (Pressure)	E-3: Loss of confinement—Impacts, missiles E-4: Direct Exposure—Worker injury E-5: Criticality—Loss of configuration or spacing
Potential (Height/Mass)	E-3: Loss of confinement—Impacts (falling objects), dropping E-4: Direct Exposure—Worker injury E-5: Criticality—Loss of configuration or spacing
Internal flooding sources	E-3: Loss of confinement—Ground/surface water runoff E-5: Criticality—Increased moderation
Physical	E-3: Loss of confinement—Puncture, dropping E-4: Direct exposure—Worker injury
Radiological material	All Events—Potentially releasable material
Hazardous material	All Events—Potentially releasable material
Ionizing radiation sources	All Events—Potentially releasable material
Nonionizing radiation	E-1: Fire—Thermal effects in combination with combustible/flammable material E-2: Explosion—Thermal effects in combination with explosive material E-4: Direct exposure—Worker injury Other—May interfere with equipment operation
Criticality	E-5: Criticality—Criticality
Nonfacility events	May lead to any event category (E-1 through E-5)
Vehicles in motion	May lead to any event category (E-1 through E-5)
Natural phenomena	May lead to any event category (E-1 through E-5)

Source: BJC/OR-1112 Facility Hazard Categorization/Classification and Hazard Analysis Application Guide, Revision 4, January 2007. Prepared for the U.S. Department of Energy.

**TABLE 20.6**

Frequency Evaluation Levels

Frequency Level	Acronym	Frequency	Qualitative Description
Anticipated	A	$10_{-1} > f = 10_{-2}/\text{yr}$	Incidents that may occur several times during the lifetime of the facility (incidents that commonly occur)
Unlikely	U	$10_{-2} > f = 10_{-4}/\text{yr}$	Events that are not anticipated to occur during the lifetime of the facility
Extremely Unlikely	EU	$10_{-4} > f = 10_{-6}/\text{yr}$	Events that will probably not occur during the lifetime of the facility
Beyond Extremely Unlikely	BEU	$f < 10_{-6} /\text{yr}$	All other Events

Source: BJC/OR-1112 Facility Hazard Categorization/Classification and Hazard Analysis Application Guide, Revision 4, January 2007. Prepared for the U.S. Department of Energy.

**TABLE 20.7**

Consequence Evaluation Levels for Hazard Receptors

Consequence Level	Public	On-Site Worker	Facility Worker
	MOI location, shortest distance to the site boundary	On-site receptor location not less than 100 m or facility boundary from the point of release. For elevated doses use point of highest doses	Involved worker within facility boundary. Use highest dose within facility boundary
High (H)	Considerable off-site impact >25 rem <sup>a</sup> TEDE or >ERPG-2/TEEL-2	Considerable on-site impact >100 rem TEDE or >ERPG-3/TEEL-3	<sup>b</sup> Facility worker hazards are typically protected with SMPs. For Safety Significant designation, consequence levels such as prompt death, serious injury, or significant radiological and chemical exposure should be considered
Moderate (M)	Only minor off-site impact >1 rem TEDE or >ERPG-1/TEEL-1	Considerable on-site impact >25 rem TEDE or >ERPG-2/TEEL-2	
Low (L)	Negligible off-site impact <1 rem or <ERPG-1/TEEL-1	Minor on-site impact <25 rem or <ERPG-2/TEEL-2	

Source: BJC/OR-1112 Facility Hazard Categorization/Classification and Hazard Analysis Application Guide, Revision 4, January 2007. Prepared for the U.S. Department of Energy.

Notes: ERPG = Emergency Response Planning Guideline, TEDE = Total Effective Dose Equivalent, MOI = Maximally Exposed Off-site Individual, TEEL = Temporary Emergency Exposure Limit.

<sup>a</sup> Off-site consequences >25 rem from operational accidents must be protected with Safety Class Structures, Systems, and Components (SSCs) independent of frequency. Follow DOE-STD-3009 for manmade external and natural phenomenon events.

<sup>b</sup> Occupational Radiation Protection, unintended (incidental) releases of sufficiently high frequency are considered a part of normal operations governed by 10 CFR 835.

**TABLE 20.8**

Qualitative Risk Ranking Bins

Frequency → Consequence ↓	Beyond Extremely Unlikely $f < 10_{-6}/\text{yr}$	Extremely Unlikely $10_{-6} \leq f < 10_{-4}/\text{yr}$	Unlikely $10_{-4} \leq f < 10_{-2}/\text{yr}$	Anticipated $f \geq 10_{-2}/\text{yr}$
High	III	II	I	I
Moderate	IV	III	II	I
Low	IV	IV	III	III

Source: BJC/OR-1112 Facility Hazard Categorization/Classification and Hazard Analysis Application Guide, Revision 4, January 2007. Prepared for the U.S. Department of Energy.

Events that have a risk level I or II are carried forth to the accident analysis. The accident analysis will conduct a qualitative/quantitative analysis to determine the radiological or chemical consequences to the on-site worker or to the public at the site boundary. From the proposed controls, identified in Table 20.7, controls are selected to prevent or mitigate the impact to on-site workers and the public. These controls are carried forth to the TSRs and become the "operating license" for the DOE contractor.



**TABLE 20.9**

Hazard Evaluation Table for (Facility X)

Event Number	Event Category	Event Description	Causes	Unmitigated			Preventive Features	Mitigating Features		
				Freq Level	Consequence Level	Risk Bin		Freq Level	Consequence Level	Risk Bin
1	E-6	Vehicle accident followed by a fire from spilled/ leaked fuel or hydraulic fluid (leaked material enters facility) Location: 7826, 7834 Release Mechanism: Impact and spill, thermal release MAR: Facility Inventory ICs: Packaging for stored waste; walls of underground bunkers prevent spilled fuel from entering the bunkers; walls of underground bunkers serve as vehicle barriers	Vehicle accident: operator error (excessive speed, inattention) operator illness/ mechanical failure of vehicle improper maintenance on vehicle AND Subsequent fire: vehicle fuel (leaking tank) hydraulic fluid (leaking lines/cylinders) incapacitation (e.g., heart attack) miscellaneous combustibles associated with vehicle AND Ignition sources: electrical (vehicle wiring, generator, power tools)	EU	Radiological Worker – High On-site – High Public – Moderate (7834)/Low (7826) Chemical Worker – Low On-site – Low Public – Moderate Physical Worker – SIH	II II III IV IV III SIH	<b>SSCs:</b> None Administrative: Transportation Safety Program (establish speed limits onsite roads)	EU	Radiological Worker – Low On-site – Moderate	IV III

Source: BJC/OR-1112 Facility Hazard Categorization/Classification and Hazard Analysis Application Guide, Revision 4, January 2007. Prepared for the U.S. Department of Energy.

Notes: EU = Extremely Unlikely, MAR = Material at Risk, IC = Initial Condition, SIH = Standard Industrial Hazard.

The selection of which controls to select is based on several general principles:

- Prevention should be chosen over mitigation.
- Passive design features should be chosen over active systems.
- Active systems and components should be chosen over ACs.
- Specific administrative controls (SACs) should be chosen over safety management program programmatically controls.
- Program programmatically controls should be chosen over a general credit of a safety management program programmatically.

These controls are then discussed in the appropriate chapter of the DSA and carried forth to the TSRs. The TSRs form the equivalent of an operating license. Violations of these nuclear safety requirements are reportable under the PAAA and may subject the contractor to fines.

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# 21

## *Neutronics*

Ronald E. Pevey

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In this chapter, we turn our attention to the general techniques and tools that are used to analyze two situations in nuclear engineering that involve neutron chain reactions (sometimes called “criticalities”): (1) nuclear reactors, which involve planned criticalities purposely invoked in a reactor core to serve an engineering purpose (e.g., electricity production, propulsion, creation of medical isotopes); and (2) excure criticalities, which are (usually) unplanned criticalities outside of a reactor core and are accidents to be avoided.

We consider each of these situations in the same chapter because they both involve similar physical and mathematical processes. However, as we will see, the differences in the two situations cause the analysis strategies to be quite different, even though they use some of the same analytical tools and data.

Our approach in this chapter will begin with an overview in Section 21.1 of the common physical and mathematical aspects that are important to the study of criticalities, and then follow this in Sections 21.2 and 21.3 with the particular concepts, analytical approaches, tools, and data that are used in the disciplines of nuclear reactor analysis and nuclear criticality safety analysis, respectively.

## 21.1 Background Mathematics and Physics

From the time that a neutron chain reaction was first envisioned to its experimental verification with the CP-1 critical pile in 1942 to the advanced reactor designs of today, the principal measure of the propensity for a system to sustain a neutron chain reaction has been the  $k$ -effective eigenvalue. In this section, we review the mathematical and physical background of criticality that both reactor physics and criticality safety have in common. This will involve several steps:

1. A discussion of the physical parameters of fission—the cross sections, energy deposition, and other physical properties
2. A review of the basic equation that governs the transport of neutrons in energy, space, direction, and time
3. A discussion of the  $k$ -effective eigenvalue, looking at both the physical background and the mathematical solution of the Boltzmann equation
4. A discussion of the primary methods of solving this equation

### 21.1.1 Neutron Chain Reactions

What makes a neutron chain reaction possible is the fact that a reaction (fission) that is induced by a single neutron reacting with an atom of a fissionable nuclide (most commonly  $^{235}\text{U}$ ,  $^{233}\text{U}$ , or  $^{239}\text{Pu}$ ) has the potential of causing the nuclide to break apart, or fission, with the release of energy and additional neutrons. If a configuration of fissionable material can be set up in which one of those released neutrons goes on to produce another fission, then the chain reaction will go on indefinitely.

One metric for criticality is the  $k$ -effective eigenvalue, which can be looked at in several ways. The simplest is as the average number of fission events that result from the neutrons emitted from a single original fission event. If this number is 1, then the rate of fission reactions will be maintained with time. If greater than 1, the reaction rate will increase and if less than 1, it will decrease:

$$\begin{aligned}
 k_{\text{effective}} &= \frac{\text{fissions in generation } N+1}{\text{fissions in generation } N} \\
 &= \frac{\text{fission neutron production rate}}{\text{neutron leakage rate} + \text{neutron absorption rate}}
 \end{aligned}
 \tag{21.1}$$

This is the generational view of  $k$ -effective.

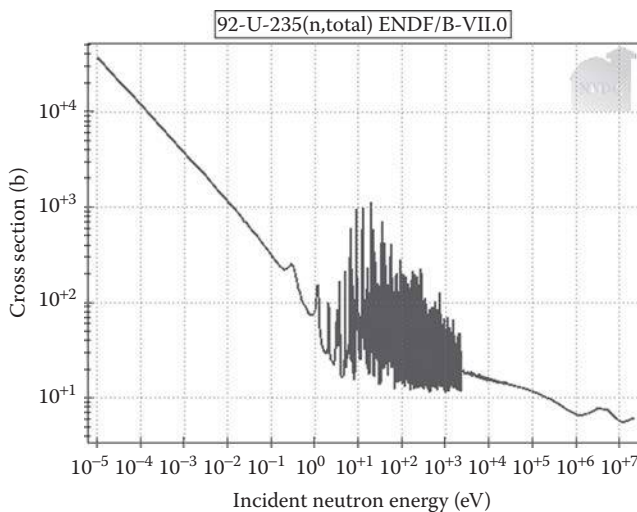
An alternate view is the reaction rate point of view, also shown in Equation 21.1, for which we look at a snapshot of reactions rates at an instant in time. In this point of view,  $k$ -effective is the ratio of the rate at which neutrons are being produced by fission reactions divided by the rate at which neutrons are disappearing due to absorption and leakage events.

In the reaction rate point of view, the important physical parameters are the neutron cross sections, which give the probabilities of various reactions occurring when neutrons interact with nuclei. The most important reaction types are fission, absorption (the sum of fission and nonfission capture), and scattering reactions. Particular forms and energy shapes of the cross sections have a strong effect on criticality.

### 21.1.1.1 Absorption

The energy-dependent cross section for a neutron reacting with  $^{235}\text{U}$  is shown in Figure 21.1. This reaction covers all reactions in which a neutron interacts with a nucleus (both from compound nucleus formation and potential scattering). This cross section is an example of what is called a *microscopic* cross section, usually denoted by a lower case Greek letter sigma, which applies to an individual nucleus and has units of area (1 barn =  $10^{-24}$  cm<sup>2</sup>). The mental picture is of the target area that a spherical target nucleus presents to an oncoming neutron (although the fact that this area depends on how fast the neutron is approaching weakens the analogy a bit). This cross section is a problem-independent property of the nuclide.

We also need to be aware of the *macroscopic* cross section that is used to determine reaction rates. The macroscopic cross section applies to a material as a whole and has units of inverse centimeters (usually interpreted as the probability of interaction per cm of travel of a free neutron). This cross section is problem-dependent, is usually denoted by the upper case Greek sigma, and is related to the microscopic cross section through the relationship:



**FIGURE 21.1**

Total cross section for  $^{235}\text{U}$ . (Data from <http://www.nndc.bnl.gov/sigma/>. With permission.)

$$\Sigma_x(E) = \sum_{i=1}^I N_i \sigma_{ix}(E) \quad (21.2)$$

where:

$N_i$  represents the atom densities (in atoms/unit volume) in the target

$x$  represents a reaction type

The two most dominant features of the total cross section in Figure 21.1 are the (1) presence of tall, narrow resonances in the intermediate energy range (actually, the resonances continue into higher energies but cannot be individually resolved experimentally) and (2) smooth, but rising, values as neutron energy decreases. Both of these features arise from the energy shape of resonances, which follow the Breit–Wigner shape for compound nucleus formation:

$$\sigma_{\text{CN}}(E) = \sigma_0 \left( \frac{E_0}{E} \right)^{1/2} \frac{1}{1+y^2}; \quad y = \frac{2}{\Gamma}(E-E_0) \quad (21.3)$$

where  $E_0$  denotes the peak energy of a given resonance. For values of  $E$  close to  $E_0$ , the ratio provides the distinctive resonance shape with energy, with the leading square root term being effectively constant. For values of  $E$  close to zero, the opposite occurs: the squared term is effectively constant and the leading square root term results in the characteristic inverse velocity shape at very low energies, from the additive effect of all of the resonances. (Note the log–log slope of  $-1/2$  in energy, corresponding to  $1/v$  dependence.)

### 21.1.1.2 Fission

The fission cross section for  $^{235}\text{U}$  is shown in Figure 21.2. The cross section general shape looks just like the total cross section in Figure 21.1, but scaled down. This is because the fission reaction is just one piece of the total cross section—it represents the total cross section multiplied by the probability that an interacting neutron causes a fission event.

Of more usefulness to our consideration of criticality is the value  $\eta$ , defined as

$$\eta(E) = \frac{v(E)\sigma_f(E)}{\sigma_a(E)} \quad (21.4)$$

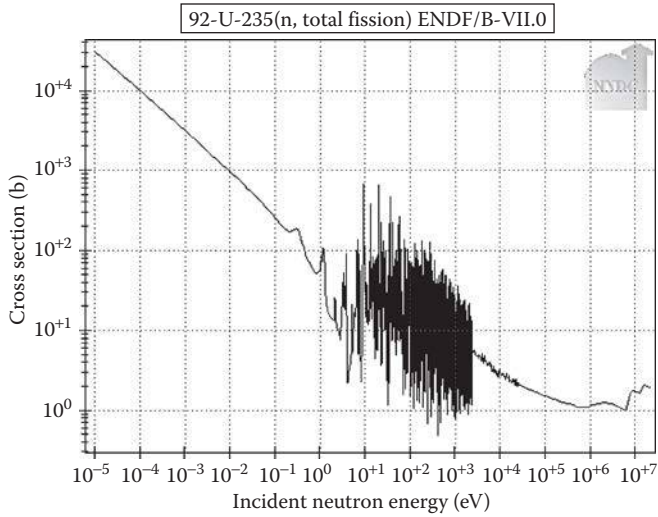
where:

$\eta(E)$  represents the number of fission neutrons emitted from a fission induced by a neutron of energy  $E$

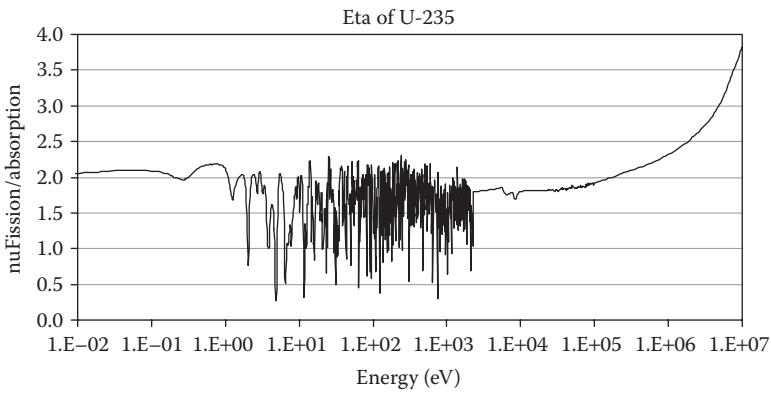
$\eta$  represents the number of fission neutrons that result per absorbed neutron in the fissile isotope, as a function of energy of the absorbed neutron

Figure 21.3 shows this ratio for  $^{235}\text{U}$ ; its usefulness lies in the view that it gives us of which neutron energies are most effective for the production of fission neutrons. Interestingly, it turns out that low-energy neutrons and high neutron energies are effective for this purpose, but that neutrons with energies between these two ranges (which also correspond with resonance energies) are less effective. This is the reason that nuclear reactors are generally designed to use either fast or thermal neutrons.





**FIGURE 21.2**  
Fission cross section for <sup>235</sup>U. (Data from <http://www.nndc.bnl.gov/sigma/>. With permission.)



**FIGURE 21.3**  
 $\eta$  value for <sup>235</sup>U.

**21.1.1.3 Scattering**

The scattering reaction is also important for criticality considerations because of the effect it has on reducing the neutron energy of the scattered neutron. The most important effect comes from elastic scattering, the predominant form of scattering in the energy range of interest to reactor and criticality problems. The neutrons that come out of an elastic scattering event have emerging energies that have been fractionally decreased by up to  $(1 - \alpha)$ , where

$$\alpha = \left( \frac{A-1}{A+1} \right)^2 \tag{21.5}$$

and  $A$  is the atomic mass of the target nucleus (in multiples of neutron mass). Note that this fraction increases dramatically for light nuclei. The practical result of this is that the

presence of low-mass nuclei will substantially lower the average neutron energy (a process called “moderation”) for a reactor or critical configuration. The most effective moderator is hydrogen, for which  $A \cong 1$  and a scattered neutron can lose up to 100% of its initial energy.

### 21.1.2 Neutron Transport Equation

The transport of neutrons through material is governed by the neutral particle Boltzmann transport equation, one form of which is given by

$$\begin{aligned} \vec{\nabla} \cdot \vec{J}(\vec{r}, E) + \Sigma_t(\vec{r}, E)\phi(\vec{r}, E) = \frac{\chi(\vec{r}, E)}{k_{\text{eff}}} \int_0^{\infty} dE' v(E') \Sigma_f(\vec{r}, E') \phi(\vec{r}, E') \\ + \int_0^{\infty} dE' \Sigma_s(\vec{r}, E' \rightarrow E) \phi(\vec{r}, E') \end{aligned} \quad (21.6)$$

where:

- $\phi(\vec{r}, E), \vec{J}(\vec{r}, E)$  are the flux and current as a function of position and energy
- $\chi(\vec{r}, E)$  distribution of fission neutrons in position and energy

This equation gives the balance of the different reactions that can occur with neutrons at a given point and energy, along with the  $k$ -effective eigenvalue, denoted by  $k_{\text{eff}}$ . Each of the terms in the equation denotes a particular loss mechanism for the neutrons (on the left-hand side of the equation) or gain mechanisms for neutrons (on the right-hand side):

- Streaming rate of loss:  $\vec{\nabla} \cdot \vec{J}(\vec{r}, E)$
- Interaction rate of loss:  $\Sigma_t(\vec{r}, E)\phi(\vec{r}, E)$
- Fission neutron production rate:  $\chi(\vec{r}, E) \int_0^{\infty} dE' v(E') \Sigma_f(\vec{r}, E') \phi(\vec{r}, E')$
- Rate of neutrons emerging from scattering reactions:  $\int_0^{\infty} dE' \Sigma_s(\vec{r}, E' \rightarrow E) \phi(\vec{r}, E')$

The engineering use of these terms comes when the equation is solved for the flux and current, and then integrated over all energies and over spatial regions of interest to get total neutron reaction rates and overall neutron leakage rates. This equation can be manipulated to solve for  $k$ -effective by picking the spatial integration region such that all of the fissionable material is included and such that escaping neutrons will not re-enter the problem and cause fission. When that is done, we get

$$\langle \vec{\nabla} \cdot \vec{J}, 1 \rangle + \langle \Sigma_t, \phi \rangle = \frac{1}{k_{\text{eff}}} \langle v \Sigma_f, \phi \rangle + \langle \Sigma_s, \phi \rangle \quad (21.7)$$

which uses the shorthand notation

$$\langle a, b \rangle = \int_V d\vec{r} \int_0^{\infty} dE a(\vec{r}, E) b(\vec{r}, E) \quad (21.8)$$

This equation can be reorganized to get

$$k_{\text{eff}} \equiv \frac{\langle v \Sigma_f, \phi \rangle}{\langle \vec{\nabla} \cdot \vec{J}, 1 \rangle + \langle \Sigma_a, \phi \rangle} \quad (21.9)$$

which is the mathematical equivalent of the last term of Equation 21.1.

### 21.1.3 Need for Neutron Distribution Knowledge

Although the determination of  $k$ -effective represents an important use of the flux profile (in space and energy), for reactor analysis the flux is also needed to obtain other engineering parameters that we need to know in the design of reactors.

Because the purpose of reactor analysis is to contribute to the design of a machine that delivers energy, perhaps the most important use of the calculated flux is to determine the amount of energy that will be deposited in various spatial regions (usually fuel elements) in the reactor. We find this energy by integrating the flux over the deposition region utilizing an energy deposition factor for each of the different neutron reactions that can occur:

$$E_{\text{deposited}} = \sum_{i=1}^I N_i \sum_x \int_0^{\infty} \varepsilon_{ix}(E) \int_V \sigma_{ix}(\vec{r}, E) \phi(\vec{r}, E) d\vec{r} dE \quad (21.10)$$

where the  $\varepsilon_{ix}(E)$  factor represents the energy deposition for reaction  $x$  in isotope  $i$  at energy  $E$ .

Other uses of the neutron flux in the reactor analysis process are in

1. The determination of the slowly time-varying nuclide densities that occur in an operating reactor (referred to as “burnup”), which includes the build-in of other fissionable isotopes (referred to as “conversion” or “breeding”)
2. The determination of asymptotic concentrations of neutron-absorbing intermediate fission products whose presence have a strong effect on subsequent  $k$ -effective and neutron flux distributions
3. The analysis of quickly varying time dependence of  $k$ -effective and neutron flux distribution, which gives us the needed information to control reactor total power and power spatial distribution changes with time
4. The control of the reactor neutron spectrum (i.e., energy dependence), which has a strong effect on the reactor operating characteristics

Each of these will be considered in Section 21.2.2, which covers the reactor analysis process.

### 21.1.4 Solving for the Neutron Flux

In general, the time dependence of the neutron flux—the quickly varying change due to power changes and control issues, and the slower varying changes due to material changes from reactor burnup—are handled with special techniques involving dedicated computer codes; these will be described in Section 21.2.

The computer codes that are used to solve for the time-independent, steady-state neutron flux fall into two categories:

1. Deterministic: Discrete ordinates, integral transport, diffusion theory
2. Stochastic: Monte Carlo

Each of these has their advantages and disadvantages; both types are in our toolset because the relative advantages and disadvantages of each result in our use of both types for various situations.

### 21.1.4.1 Deterministic Solutions

Deterministic methods involve the numerical subdivision of the independent variables of space, energy, and direction into computational subdivisions, with a subsequent reformulation of the continuous-variable Boltzmann Equation into a set of discrete-variable equations for each *phase cell* (i.e., the combination of a space subdivision, an energy subdivision, and a direction subdivision). The flow of particles through space causes a linkage among the spatial subdivisions, and the scattering process causes a linkage among the energy and directional subdivisions. Specialized deterministic computer codes solve these coupled linear algebra equations for the neutron flux in each phase cell, and the desired flux integrals are approximated by summations over the appropriate cells to get the engineering parameters (including *k*-effective) of interest in the analysis.

Because the basic formation of the deterministic solution requires this extensive subdivision, deterministic methods naturally provide the detailed information in space and energy that the reactor design process requires. In addition to this wide range of information that they deliver, the deterministic methods also tend to be fast, accurate, and amenable to acceleration and convergence improvement methods from well-developed numerical analysis techniques.

The three predominant variations of deterministic methods are as follows:

1. The *discrete ordinates* method subdivides all three independent dimensions—space, energy, and direction. Space is divided into a regular one-dimensional (1D), two-dimensional (2D), or three-dimensional (3D) grid; energy utilizes the multigroup method; and direction is handled by calculating the flux only in particular directions (discrete ordinates). Each of the spatial cells is linked to its immediate neighbors through the flow of particles through the intervening boundaries; energy and direction subdivisions are linked through scattering. The amount of computer time that the solution requires is strongly dependent on the complexity of the geometry and the resolution of the spatial grid, the number of energy groups used, and the number of directions followed. This is the slowest, but most accurate, of the deterministic methods. It is generally used for pin-cell and fuel assembly calculations for situations for which regular geometries either apply or can be reasonably approximated.
2. The *integral transport* method (also called the *collision probability* method) subdivides only space and energy. Space is again divided into a grid, but the grid regions tend to be larger and less regular than for discrete ordinates. The energy variable is again handled using the multigroup method. The basic relationship (based on an integral form of the Boltzmann transport equation) is that each of the spatial regions is linked to each of the other spatial regions; the direction of particle travel does not show up explicitly in the equation but is buried in the spatial region-to-region linkage parameters. The loss of spatial resolution and a usual assumption that scattering is isotropic cause these solutions to be faster, but less accurate, than discrete ordinates solutions. The integral transport method is frequently used for pin-cell and fuel assembly calculations with nonregular geometry, but for which the isotropic scattering approximation can be reasonably employed.
3. The *diffusion theory* method simplifies the directional dependence by assuming a nearly isotropic flux (i.e., an approximately equal number of neutrons traveling in all directions). The spatial treatment again utilizes a regular grid of spatial elements connected to their immediate neighbors and the energy treatment again employs the multigroup approach. This is the simplest and fastest of the

deterministic methods, but the direction approximation is the most extreme, limiting its usefulness near sources or strong absorbers. The primary use of diffusion theory is for full-reactor calculations in which reactor assemblies have been homogenized, so that strong absorbers and fission/scattering sources have been mathematically spread out.

#### 21.1.4.2 Stochastic Solutions

Stochastic (Monte Carlo) methods solve for the neutron flux by simulating particle transport rather than by numerically solving the Boltzmann transport equation (although the simulational algorithm can be derived from the equation itself as well as from the physics).

The Monte Carlo particle simulation involves use of an algorithm in which the decisions faced by a neutron in its lifetime are simulated in the computer code, in the order in which they come up in the physics. For each Monte Carlo decision, the outcome is determined from the mathematical probability densities that nuclear physics experiments give us for the physical effect being simulated.

For criticality problems, neutrons are born in “batches” or “generations” of particles (generally hundreds to thousands of particles for each one of hundreds to thousands of generations). The decisions that are faced by each fission-produced neutron are

1. An initial position is chosen for the particle based on fission sites from the previous generation (or assumed for the first generation).
2. An initial energy is chosen from the known probability distribution of fission neutrons from the appropriate fissionable nuclide (and possibly depending on the energy of the neutron inducing the fission).
3. An initial direction is chosen from an isotropic distribution.
4. The distance traveled to the next collision site is chosen. Actually, this is done in two steps: the number of mean free paths to the next collision is chosen and then this is translated into a distance based on the materials and geometry of the problem. (This second step takes up the majority of the computer time needed by the Monte Carlo code.) If this distance is greater than the distance to the edge of the volume being calculated, the particle is deemed to have escaped and the history is terminated.
5. The type of collision is determined from the macroscopic cross sections of the material present at the collision site. If the reaction is a neutron absorption reaction, the particle history is terminated; if the absorption reaction is fission, the location of the collision is stored as a potential starting point for the next generation of neutrons.
6. If the collision is a scattering reaction, the new energy and direction are chosen from the scattering distributions for the material, and the simulation returns to step 4.

As each of the simulated particles live its life, the code keeps track of information of interest: average flux values in the regions of the problem, energy deposition in the regions of the problems, the number of new fission neutrons produced by the generation. This last one is of particular interest since the ratio of new fission neutrons to the number of neutron particle histories followed in the generation gives an estimate of  $k$ -effective for the generation, using Equation 21.1. The statistical processing of these  $k$ -effective estimates gives the code its final estimate of  $k$ -effective, with statistical uncertainty (estimated standard deviation).

TABLE 21.1

Comparison of Advantages and Disadvantages of Deterministic and Stochastic Methods

Method	Advantages	Disadvantages
Deterministic	Fast (1D,2D);	Slow (3D);
Discrete ordinates	Accurate for simple geometries;	Multigroup energy treatment required;
Integral transport	Complete spatial, energy, angular map of the flux;	Geometry must be approximated;
Diffusion theory	1/N (or better) convergence;	Large computer memory required;
	Follows rare particles well	User must determine accuracy with repeated calculations
Stochastic	Exact geometry;	Slow (1D, 2D, 3D);
Monte Carlo	Continuous energy possible;	Large computer time requirements;
	Estimate of accuracy given;	1/N <sup>1/2</sup> convergence
	Follows common particles well	

This simplified description has assumed that the exact physical probabilities are utilized to determine the outcome of every decision; when this is done, the resulting simulation is termed an *analog* simulation. More sophisticated statistical treatments are included in modern computer codes that utilize nonphysical distributions with corrections (in a defined particle *weight*) to keep the results of the simulation unbiased; these can be shown to improve the efficiency of the simulation. These methods are called “variance reduction” methods, although this is somewhat of a misnomer because many of these methods increase efficiency by saving computer time, not by reducing variance. The exact theory and technique for doing this is beyond the scope of this handbook but is well described in Monte Carlo descriptions such as in Lewis and Miller (1993).

A key feature of Monte Carlo codes is that the accuracy of the results delivered by the method depends on the number of particle histories followed ( $N$ ) by the ratio  $1/\sqrt{N}$ . This means that a decrease of a factor of 10 in the uncertainty of any parameter requires the number of histories to be increased by a factor of 100. This fact severely limits the efficiency of Monte Carlo methods to deliver precise results, especially for region-dependent values of flux and energy deposition. The result is that Monte Carlo methods are generally much more efficient in calculating system-wide parameters such as  $k$ -effective than they are for calculating detailed spatial parameters such as energy deposition profiles.

### 21.1.4.3 Summary of Deterministic vs. Stochastic Advantages and Disadvantages

Table 21.1 gives a synopsis of the relative advantages and disadvantages of the deterministic and stochastic solution strategies.

## 21.2 Reactor Core Analysis

### 21.2.1 General Overview of Reactor Analysis

The process of analyzing a nuclear reactor core over its lifetime is generally broken down into several distinct tasks:

1. Calculating the reactor properties—primarily the  $k$ -effective eigenvalue and the fission power distribution—at distinct “snap-shots” of reactor material properties and temperatures over the projected life of the core

2. Calculating, in iteration with the previous step, the changing material distributions in the reactor—taking into account depletion of the fuel, buildup of fission products, control absorber concentrations and distributions, reshuffling of fuel elements, and replacement of used fuel
3. Performing kinetics calculations at distinct points of the core lifetime to determine the control parameters and predicted dynamic response of the reactor to prescribed design-basis accidents

Calculation of the fission power distribution in Step 1 involves the handling of a tremendous amount of geometric details of the reactor because the reactor might contain tens of thousands of fuel elements, control elements, instrument ports, coolant flow channels, structural components, etc. In addition to the spatial detail, the designer needs to know the neutron flux distributions in energy and direction as well. This is an overwhelming amount of detail.

### 21.2.2 Basic Static Reactor Analysis Approach

The result of this situation is that the typical reactor analysis process involves a “divide and conquer” attack in which the spatial and energy details are not handled simultaneously, but instead a multistep synthesis process is followed that gradually increases the spatial detail while relaxing the energy detail. We will describe this processing in seven steps.

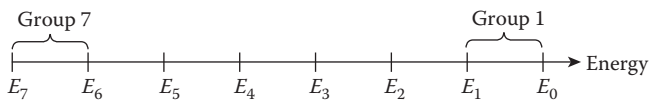
#### Step 1: Finegroup nonresonance cross section creation

The first step involves all of the energy detail and none of the spatial detail, with the creation of finegroup cross sections from the full-energy-dependent cross sections for each of the isotopes and each of the reactions of each isotope. Although we refer to this step as the first step, it is actually usually performed prior to the beginning of the design process since it does not depend on the individual characteristics of the reactor being designed or analyzed.

Multigroup cross section theory involves the discretization of the energy dimension by replacing the continuous energy variable with  $G$  energy groups, with  $G$  usually in the range 50–500, traditionally numbered from high to low energy as shown in Figure 21.4.

The basis of the multigroup approximation involves conserving reaction rates over each of the energy groups; this involves the definition of the group  $g$  parameters for a given isotope as

$$\phi_g = \int_{E_g}^{E_{g-1}} \phi(E) dE \quad (21.11)$$



**FIGURE 21.4**  
Discretization of the energy variable into energy groups.

$$\sum_{t_g} = \frac{\int_{E_g}^{E_{g-1}} \phi(E) \sum_t(E) dE}{\int_{E_g}^{E_{g-1}} \phi(E) dE} \quad (21.12)$$

$$S_g = \int_{E_g}^{E_{g-1}} S(E) dE \quad (21.13)$$

$$\chi_g = \int_{E_g}^{E_{g-1}} \chi(E) dE \quad (21.14)$$

$$v\Sigma_{f_g'} = \frac{\int_{E_g}^{E_{g-1}} \phi(E) v\Sigma_f(E) dE}{\phi_g} \quad (21.15)$$

$$\Sigma_{s_g' \rightarrow g} = \frac{\int_{E_g}^{E_{g-1}} \int_{E_g'}^{E_{g-1}'} \phi(E') \Sigma_s(E' \rightarrow E) dE' dE}{\phi_{g'}} \quad (21.16)$$

The continuous energy cross sections are obtained from the Evaluated Nuclear Data Files (ENDF/B), which include smooth cross sections up to at least 20 MeV, secondary scattered neutron energy and directional distributions, thermal scattering kernel properties, and resonance parameters.

The key to the accuracy of the multigroup approach is good information on the expected energy shape of the neutron flux over the range of the individual groups, although the need for good approximations decreases as the number of energy groups increases (which is the reason that this step begins with enough groups to qualify as a “finegroup”). The most common approximation for the energy spectrum shape is the so-called “fission-1/ E-Maxwellian” approximation:

$$\phi(E) = \begin{cases} c_1 \frac{E}{(kT)^2} \exp\left(-\frac{E}{kT}\right), & \text{if } E < E_{\text{thermal}} \\ 1/E, & \text{if } E_{\text{thermal}} < E < E_{\text{fast}} \\ c_2 \chi(E), & \text{if } E > E_{\text{fast}} \end{cases} \quad (21.17)$$

which can be shown to be a reasonable energy shape for a thermal reactor. The constants  $c_1$  and  $c_2$  in the equation are chosen to provide continuity at the breakpoints  $E_{\text{thermal}}$  and  $E_{\text{fast}}$  which are usually placed around 0.5 eV and 0.5 MeV, respectively.

### Step 2: Resonance processing

One of the physical assumptions that go into the 1/E flux shape in the slowing-down range in Equation 21.17 is that the primary neutron reaction in this energy range is elastic scattering. This assumption can break down near the previously described absorption resonances, resulting in deep dips in the energy flux shape at these energies as neutrons are strongly absorbed in the resonances. Because



the energy flux shape serves as a weighting function in Equation 21.11 through 21.16, these severe disruptions in the flux shape only seriously affect the resulting finegroup cross sections for the resonance nuclides themselves. This fact results in Equation 21.11 through 21.16 doing a reasonable job of calculating finegroup cross sections except for resonance nuclides at energy groups corresponding to resonance energies.

The second step improves on the assumed energy flux shape by calculating the flux dips around specific resonances and applying these improved spectra in Equations 21.11 through 21.16, resulting in improvements to the finegroup cross sections for these isotopes. The actual method used depends very strongly on the material contents of the fuel, especially for actinide isotopes—so is sensitive to fuel burnup—and also depends on the geometry immediately around each fuel element.

Compared with the spatial independence of Step 1, Step 2 uses a minimal amount of spatial information—usually just the fuel element dimensions and pitch between fuel elements—although with increasing computer power more spatial information is being incorporated into this step. This information about the sizes of the fuel and moderator (coolant) regions is very important in determining the resulting neutron spectrum because of the effect of neutrons that begin with a high energy, escape the fuel element into the surrounding moderating coolant, and then stream from the moderator back into the fuel element. The effect of the moderated neutrons is usually to dramatically increase the  $k$ -effective of the reactor; because of this, thermal reactors are generally designed to be a little under-moderated (i.e., with a little less than optimum amount of moderating coolant volume per pin) so that a power transient that increases power—increasing temperature and reducing coolant density—will result in a reduction in  $k$ -effective.

As an example of the resulting spectrum, a simple approximation of the slowing-down region neutron balance (through the narrow resonance approximation) and a simple spatial approximation of a lump of resonance absorber surrounded by an infinite “sea” of moderator results in a simplified neutron balance relation of

$$[\Sigma_m^F + \Sigma_t^F(E)]\phi_F(E) = [1 - P_{F0}(E)] \left[ \int_E^{E/\alpha_F} \frac{dE'}{E'} \frac{\Sigma_s^F(E')\phi_F(E')}{1 - \alpha_F} + \frac{\Sigma_m^F}{E} \right] + P_{F0}(E) \frac{\Sigma_m^F + \Sigma_t^F(E)}{E} \quad (21.18)$$

where:

$\Sigma_m^F, \Sigma_t^F, \Sigma_s^F$  are the macroscopic cross sections for moderator scattering, total interactions, and fuel scattering in the fuel

$\phi_F(E)$  is the energy-dependent flux in the fuel (normalized for  $1/E$  moderator flux shape)

$P_{F0}(E)$  is the probability that a neutron released (uniformly) in the fuel has its next collision in the moderator

This results in an approximate neutron energy spectrum in the fuel of

$$\phi_F(E) = \frac{\Sigma_s^F + \Sigma_m^F + P_{F0}(E)[\Sigma_t^F(E) - \Sigma_s^F(E)]}{[\Sigma_t^F(E) + \Sigma_m^F]E} \quad (21.19)$$

This spectrum (after a lattice correction) is used to calculate improved problem-dependent finegroup reaction cross sections for resonance nuclides in resonance-range energy groups using Equations 21.11 through 21.16.

### Step 3: Pin-cell analysis

In the third step, we combine the finegroup microscopic cross sections produced in the first two steps according to the fuel isotope densities to create burnup-dependent macroscopic finegroup cross sections. These are used in a transport calculational model utilizing a more complete model of each of the fuel pins, including the fuel, gap, clad, coolant, and moderator materials and geometry.

Traditionally, the nonorthogonality of the geometry has led to the integral transport method being used for this step; in recent years, however, the discrete ordinates method has been expanded to include nonorthogonal geometries, and so is increasingly called upon to perform this calculation. At the same time, as computer processing speeds increase, the Monte Carlo method is also becoming a more viable choice.

The final result of this step is the average fuel, clad, and moderator flux levels in each of the finegroup energy groups. These fluxes are used to “smear” the entire fuel pin region into equivalent multigroup cross sections that are subsequently applied to the fuel element as a whole, but utilizing a smaller number of energy groups—typically 10 to 50 groups. These *broadgroup* cross sections are found from the original finegroup cross sections for the individual spatial elements (fuel, clad, and moderator) using the relation

$$\sigma_{iG}^{\text{pin}} = \frac{\sum_{r \in \text{pin}} V_r \sum_{g' \in G} \hat{\sigma}_{rtg'} \hat{\phi}_{rg'}}{\sum_{r \in \text{pin}} V_r \sum_{g \in G} \hat{\phi}_{rg}} \quad (21.20)$$

for finegroups  $g$  inside broadgroup  $G$ .

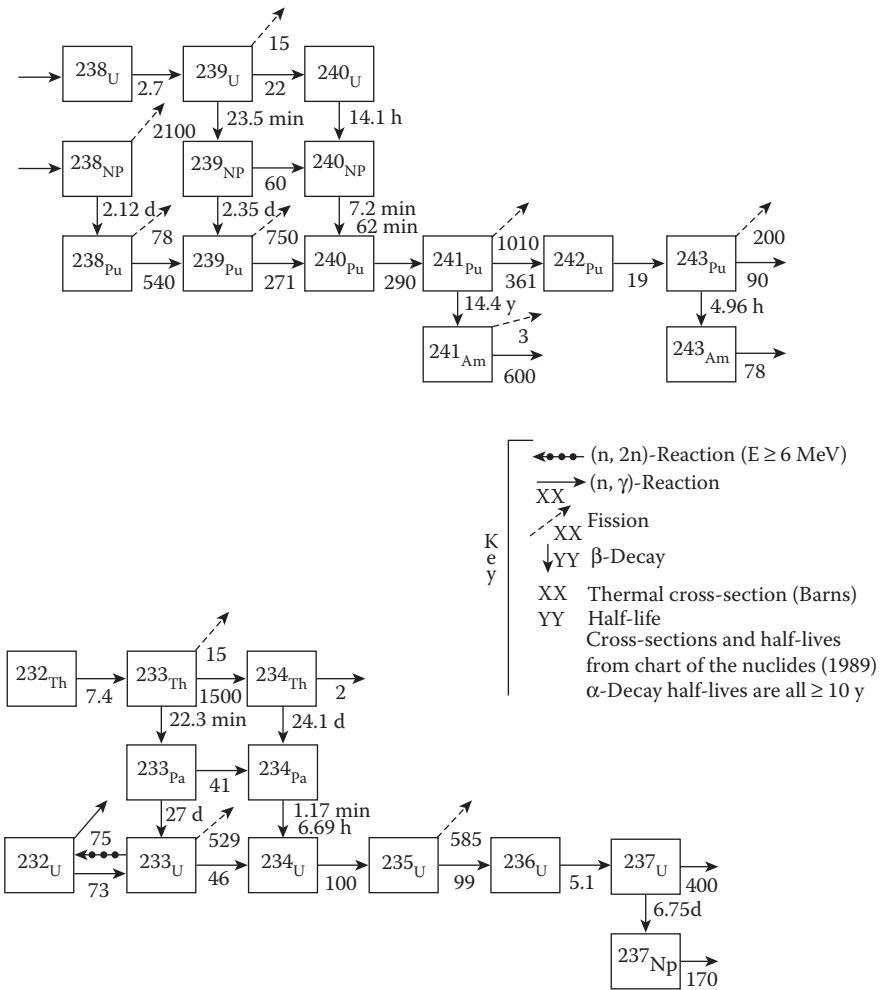
### Step 4: Lattice analysis

The fourth step is similar in approach and goal to the third but is applied to a fuel assembly rather than a fuel pin. The equivalent smeared broadgroup cross sections for each of the basic fuel pin types are fashioned together to form each of the assembly types in the reactor, and each of these assembly types are similarly calculated with a transport code and “smeared” to form equivalent microscopic *fewgroup* cross sections. Again, as the spatial scope expands, the energy representation decreases, this time to 2–10 energy groups.

Again, the traditional calculational tool for this step has been integral transport (collision probability) methods, but discrete ordinates and Monte Carlo are seeing increased use.

### Step 5: Burnup analysis

The fifth step of reactor analysis “burns” the reactor for a user-specified period of time to update the isotopic contents, then returns to Step 2. The reactor composition changes with time due to irradiation effects and radioactive decay. These effects must be taken into account in the design of the reactor, in fact may be the primary design consideration of the reactor itself (e.g., the high flux isotope reactor at the Oak Ridge National Laboratory).



**FIGURE 21.5** Transmutation-decay chains for U-238 and Th-232. (From Knief, R.A., *Nuclear Engineering*, Taylor & Francis Group, Washington, DC, 1992. With permission.)

Reactor isotopic changes occur for two reasons: irradiation effects (primarily due to fission and transmutation) and radioactive decay. Figure 21.5 shows the transmutation decay chains for U-238 and Th-232.

In general, an equation can be written for each isotope in the reactor, with some combination of these terms involving decay and transmutation to and from the isotope:

$$\frac{dN_j(t)}{dt} = \gamma_j \sum_g \Sigma_{fg} \phi_g + \sum_{i \neq j} (\lambda^{i \rightarrow j} + \sum_g \sigma_g^{i \rightarrow j} \phi_g) N_i - (\lambda^j + \sum_g \sigma_{ag}^j \phi_g) N_j \quad (21.21)$$

where:

- $\gamma_j$  is the fraction of fission events resulting in isotope  $j$
- $\lambda^{i \rightarrow j}$  is the decay constant from isotope  $i$  to isotope  $j$
- $\lambda^j$  is the total decay constant for isotope  $j$

$\sigma^{i \rightarrow j}$  is the transmutation cross section from isotope  $i$  to isotope  $j$   
 $\sigma_a^i$  is the total loss cross section for isotope  $i$

More formally, the linked equations can be translated into a “simple” matrix ordinary differential equation:

$$\frac{d\mathbf{N}(t)}{dt} = \underline{\underline{A}}_i(\phi(t_i))\mathbf{N}(t) + \underline{\underline{F}}_i(\phi(t_i)), \quad t_i < t < t_{i+1} \quad (21.22)$$

with solution

$$\mathbf{N}(t_{i+1}) = \exp[\underline{\underline{A}}_i \Delta t] \mathbf{N}(t_i) + \underline{\underline{A}}_i^{-1} \{ \exp[\underline{\underline{A}}_i \Delta t] - 1 \} \underline{\underline{F}}_i \quad (21.23)$$

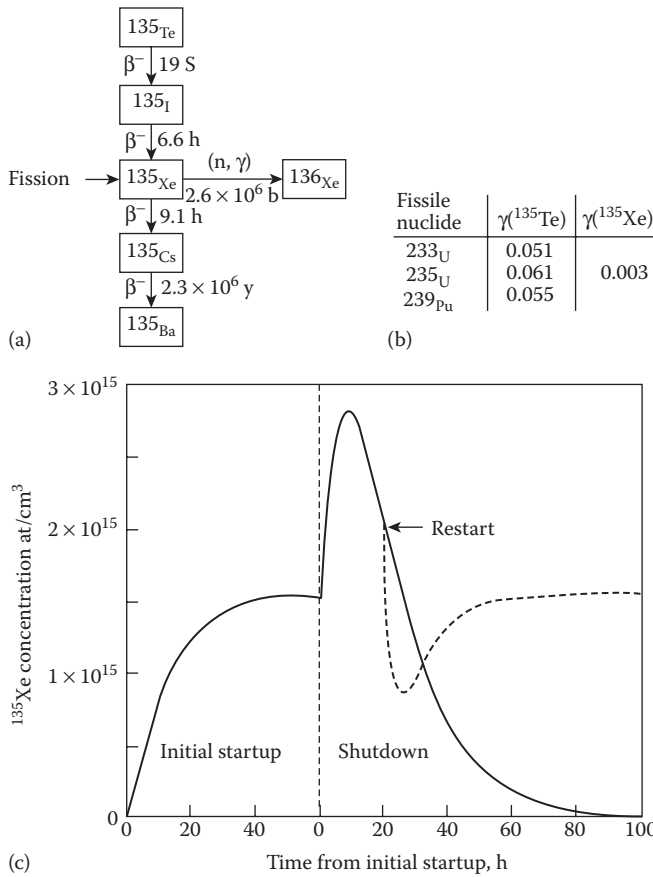
This equation uses  $\Delta t$  for the “burnup time steps” that the user specifies (and are required because  $A$  and  $F$  depend on flux). The flux at  $t_i$  is used for times up to  $t_{i+1}$ , when the flux terms of the  $A$  and  $F$  matrices are updated.

The primary consequence of burnup is a drop in  $k$ -effective as the fuel burns out and fission products are built up. This drop is compensated by the build-up of new fissile isotopes (notably Pu-239 from U-238 neutron absorption in uranium-fueled reactors). Generally, boiling water reactors and pressurized water reactors replace the fuel in stages, with fresh fuel assemblies replacing the most burned-out assemblies at scheduled shutdowns with nonreplaced assemblies often moved (shuffled) to new positions to optimize the reactor operating characteristics.

A second consequence of particular importance for thermal reactors is the build-in of the intermediate fission product chain isotopes with particularly high absorption cross sections, notably Xe-135 and Sm-149. The decay chain that produces Xe-135 is shown in Figure 21.6, along with a typical time profile for the Xe-135; Figure 21.7 shows the same for Sm-149. Note from the time profile that both of these reach an equilibrium value in an operating reactor (at a level that depends on the flux level) within a few days. When the reactor is shut down, Sm-149 (which is stable) rises to a new equilibrium, whereas Xe-135 rises initially and then decays. When the reactor is restarted, the equilibrium for both of these isotopes re-establishes itself.

Because of the expected drop in  $k$ -effective, extra fuel is built into the reactor at the beginning of the core life, which in turn requires a more extensive control system to counteract this initial “extra”  $k$ -effective. This is referred to as “excess reactivity” (with “reactivity” generally defined as the  $(k\text{-effective}-1)/k\text{-effective}$ ). Many core designs include built-in absorbers, referred to as “burnable poisons” that reduce the control problems that this excess reactivity introduces. These absorbers reduce the initial  $k$ -effective—allowing less absorbing control systems to be used—but are generally designed to burn up before the end of the cycle so as to not shorten the core lifetime. (Alternatively, burnable poisons can be used to extend the reactor cycle life with the same control system.)

With the new contents after the burnup step, the process loops back and repeats Steps 2–5 for the new fuel pin contents for the next burnup step. This iterative sweep through the analysis steps continues until the end of the reactor cycle. Once the entire cycle is complete, the fuel assembly contents and fewgroup cross sections are known as a function of burnup—often expressed as functional fits of



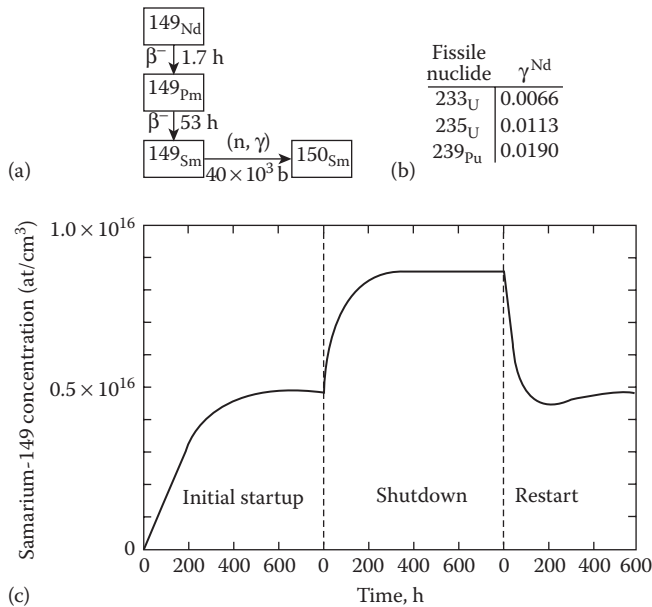
**FIGURE 21.6** Characteristics of Xe-135 under representative LWR conditions: (a) transmutation-decay chain; (b) fission yields; (c) time dependence. (From Knief, R.A., *Nuclear Engineering*, Taylor & Francis Group, Washington, DC, 1992. With permission.)

each of the neutron reaction cross sections as a function of total fission energy produced by an assembly (*exposure*) in units of MW-days/assembly; these fitted cross sections are then passed to the next step.

**Step 6: Full reactor analysis**

After the types of reactor assemblies have been characterized by equivalent few-group cross sections as a function of exposure, these assemblies are used to analyze the reactor core as a whole. Each of the assemblies is placed in its appropriate location and is then allowed to be exposed based on the calculated assembly power levels across the core. This allows the analyst to accurately predict the *k*-effective of the core and the varying spatial power shape across the core and the assembly contents as the reactor ages.

Traditionally, the detailed power distribution within each assembly is deconstructed by using the combinations of the results from each of the previous reactor analysis steps. That is, the within-pin power profiles from Step 3 (based ultimately on the flux energy shapes from Steps 1 and 2) are superimposed on the power



**FIGURE 21.7** Characteristics of Sm-149 under representative LWR conditions: (a) transmutation-decay chain; (b) fission yields; (c) time dependence. (From Knief, R.A., *Nuclear Engineering*, Taylor & Francis Group, Washington, DC, 1992. With permission.)

profile across the respective assemblies, which are in turn superimposed on the full-reactor power shape from Step 6.

The calculational method used in Step 6 is usually diffusion theory, often improved through the use of “transport corrections” of various sorts. Diffusion theory, as was previously mentioned, is the simplest neutron transport calculational method. It assumes a very low-order directional flux shape, which makes it inappropriate for use in problems that involve strong isolated sources or strong isolated absorbers. It is not accurate enough for Steps 3–5 because of fuel pin and control rods; once these sources and absorbers have been “smeared” into the assembly cross sections that come from Step 5, diffusion theory is generally accurate enough for Step 6 full reactor calculations.

**Step 7: Temperature, kinetics, and control calculations**

The final step does not follow the chronological pattern of the previous steps, but instead represents special thermal hydraulic, safety, and control calculations that are performed at each of the burnup steps.

The thermal-hydraulic analysis is performed using special codes that receive the fission power spatial deposition from the reactor analysis pin power deconstruction and perform a heat transfer and fluid flow calculation to calculate the expected temperature profile throughout the core. Because neutron cross sections depend on temperature, this calculation must be iteratively linked into the previous six steps. In many cases, this linkage results in an expansion of the parametric expansions of the neutron cross sections (described as part of Step 6) to be a function of temperature and total assembly fission power.

Safety and control calculations represent “side” calculations that are performed at each burnup step, once the proper neutron cross sections are determined, to calculate reactor behavior under accident and normal reactor control conditions. From the point of view of reactor analysis, both of these build on Steps 1–6 of analysis of the reactor static conditions to determine the reactor kinetic conditions.

The equations that are solved for these analyses are the time-dependent version of the neutron transport equation:

$$\begin{aligned} \frac{1}{v} \frac{\partial \phi(\vec{r}, E, t)}{\partial t} + \vec{\nabla} \cdot \vec{J}(\vec{r}, E, t) + \Sigma_a(\vec{r}, E) \phi(\vec{r}, E, t) \\ = \frac{\chi(\vec{r}, E)}{\lambda} (1 - \beta) \int_0^{\infty} dE' v(E') \Sigma_f(\vec{r}, E') \phi(\vec{r}, E', t) \\ + \sum_i \lambda_i C_i(\vec{r}, t) \chi_i(\vec{r}, E) + \int_0^{\infty} dE' \Sigma_s(\vec{r}, E' \rightarrow E) \phi(\vec{r}, E', t) \end{aligned} \quad (21.24)$$

which differs from the previous Equation 21.6 by the addition of the time variable in the flux term, the addition of the time derivative term, and the breaking of the fission neutron production term into both prompt neutron components—which are emitted essentially instantaneously from a fission event—and delayed neutrons—which are released from fission products (called “precursors” in this context) that decay by the emission of neutrons. The  $C_i(r, t)$  in the equation represent the concentrations of representative “groups” of precursors with similar decay times. Although the delayed neutrons represent a relatively small fraction of the fission neutrons released (e.g., about 0.67% for fission in  $^{235}\text{U}$ ), they have a tremendously important effect on the reactor safety and control characteristics. In fact, they are responsible for slowing down reactor dynamics into a time range that allows human beings to control them.

Solution of the space–time Equation 21.24 is usually done either at the assembly level (Step 6), where the fission powers of each of the assemblies are the variables of the equation, or at the reactor level, where the power level of the reactor as a whole is calculated and then the powers are “distributed” among the assemblies based on the calculated power profiles from Step 6. To show how the equations work, we will assume that the second path is taken and a reactor-wide kinetic power calculation is performed.

Integrating Equation 21.24 over the entire spatial and energy range of the reactor results in the linked equation sets:

$$\frac{dn(t)}{dt} = \frac{\rho(t) - \beta}{\Lambda} n(t) + \sum_{i=1}^6 \lambda_i C_i(t) \quad (21.25)$$

with

$n(t)$  the number of neutrons in the reactor as a function of time;

$$\rho(t) = 1 - \frac{1}{k(t)} = \frac{k(t) - 1}{k(t)} \equiv \text{reactivity of the reactor}$$

$$\Lambda = \frac{\langle n \rangle}{\langle v \Sigma_f, \phi \rangle} = \frac{\langle (1/v), \phi \rangle}{\langle v \Sigma_f, \phi \rangle} = \frac{\langle (1/v), \phi \rangle / \langle \phi \rangle}{\langle v \Sigma_f, \phi \rangle / \langle \phi \rangle}$$

$$= \frac{\text{Flux averaged "inverse velocity"}}{\text{Flux averaged } v \Sigma_f \text{ cross section}} \sim \frac{1}{\bar{v} \Sigma_f}$$

$$= \text{"mean generation time"}$$

Note that the  $n(t)$  can be replaced with anything that is proportional to neutron density, usually fission power. (This substitution would change the units of  $C_i(t)$ .)

In addition to the parameters which are obtained from the reactor static fluxes, we also need to account for the  $k$ -effective changes due to second-order effects of changing the temperature distribution in the reactor, including cross sections changes in the fuel, cross sections changes in the coolant, liquid density changes, solid density changes due to expansion, and void formation in the coolant. These effects are introduced into the equation through the use of reactivity feedback coefficients, usually partial derivatives denoted by  $\alpha$ :

$$\alpha_T \equiv \frac{\partial \rho}{\partial T} = \frac{\partial}{\partial T} \left( \frac{k-1}{k} \right) = \frac{1}{k^2} \frac{\partial k}{\partial T} \cong \frac{1}{k} \frac{\partial k}{\partial T} \quad (21.26)$$

These are usually approximated by comparison of changes in  $k$ -effective due to small changes in the temperature of the fuel, moderator, cladding, and coolant:

$$\alpha_T \cong \frac{1}{k} \frac{\partial k}{\partial T} \cong \frac{k_2 - k_1}{k(T_2 - T_1)} \quad (21.27)$$

where the 1 and 2 subscripts denote two reactor states differing only by small changes in temperature of the examined region.

### 21.2.3 Basic Reactor Analysis Tools and Data

In modern reactor analysis, the calculations previously outlined are performed on computer codes specially written for the purpose. For most actual operating reactors, these computer codes are proprietary to the reactor vendors (or, in some cases, the utilities that own the reactor) and are not generally available to outside analysts. However, there are computer codes available from Reactor Safety Information Computational Center (RSICC), the computer code distribution center at the Oak Ridge National Laboratory, which can be used for reactor analysis. For the most part, the available computer codes are those that were developed at the various national laboratories (Oak Ridge National Laboratory [ORNL], Los Alamos National Laboratory [LANL], Argonne National Laboratory [ANL]), although some university and privately developed software is also available.

Of particular interest for nonproprietary use is the SCALE system, developed at ORNL, which has the SAS2 and TRITON sequences that are available to perform Steps 2–5 (with Step 1 results already embedded in the cross sections libraries provided). Adding to this capability with a diffusion theory code (e.g., VENTURE) and a kinetics code (e.g., SKINATH) gives the general analyst capability of performing basic reactor analysis.

Although this path is often chosen, for example, to determine approximate assembly contents for shipping cask source terms, it is to be expected that the proprietary codes will be more finely tuned for the analysis of the particular reactors for which the codes were written.



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### 21.3 Criticality Safety Analysis

Criticality safety analysis is a discipline aimed at the prevention or termination of inadvertent nuclear chain reactions in nonreactor environments. Although the physics and mathematics are the same as for reactor analysis, the basic approach is different because the goal is different. Because the analysis goal of criticality safety exclusively is to establish a conservative assessment of the criticality state of a given proposed operation, and the systems being analyzed have so much variety, the methodology takes a completely different form than reactor analysis methodology.

Despite these differences, we will examine the practice of criticality safety with reactor analysis as the starting point for several reasons:

- This is the path taken by many professionals in the field. Because of the need for criticality safety analyses in the various elements of the nuclear fuel cycle, many criticality safety practitioners work for companies in the nuclear energy industry. Because many of these professionals are likely to have a reactor analysis background, it makes sense to build on this background in a handbook like this one.
- The current educational curricula in nuclear engineering university departments are likely to include several courses and associated laboratories involving nuclear reactor theory and engineering, but few universities offer courses in criticality safety.
- The first part of this chapter involves nuclear reactor analysis, so we can avoid duplication by building on those ideas.

So, with this flow in mind, let us lay out the significant differences between the theory and practice of reactor analysis and that of criticality safety analysis:

Difference #1: Criticality safety analysis is interested only in the  $k$ -effective eigenvalue.

This is an oversimplification because the additional information that is available from understanding of the system-wide fluxes and reaction rates is certainly useful to the criticality safety analyst's understanding of the system being analyzed. Nonetheless, the fact that the  $k$ -effective eigenvalue is the main piece of information that the computational tools must deliver has important ramifications. The most important of these is that the advantages and disadvantages of deterministic versus Monte Carlo analysis methods strongly favor Monte Carlo for criticality safety analysts. For this reason, Monte Carlo codes are the predominant neutron transport tools used in criticality safety.

Difference #2: The neutron balance relationship is the same, but the relative importance of loss mechanisms is different.

As laid out in the Equation 21.1 definition of  $k$ -effective, the basic neutron balance relationship is between fission neutron production and neutron loss mechanisms, the two neutron loss mechanisms being neutron absorption and neutron leakage. Nuclear reactors, for which criticality must be created and maintained for extended periods, tend to be large devices—both for neutron balance reasons and because of the large fuel inventory required to deliver substantial amounts of power for long periods of time. The result of this, from the point of view of the  $k$ -effective equation, is that neutron leakage is a relatively unimportant neutron

loss mechanism. For this reason, reactor control is almost exclusively maintained through the addition and removal of neutron absorbers.

On the other hand, fissile material handling activities, for which criticality is to be avoided, the preference is to limit processing to smaller accumulations of fissile material. Therefore, neutron leakage is a much more important factor in the design of safe processes; in fact, it usually comes as a surprise to new professionals in the field of criticality safety (after a background in or study of reactor analysis) to find out what a small role neutron absorbers play in criticality safety. (The primary reason for this is the reluctance to “poison” fuel in the processing steps and the difficulty of proving that nonembedded neutron absorbers will remain effective during processing activities.)

Difference #3: There is a wider range of geometries and materials in criticality safety analyses.

Compared with the controlled environment that is purposely designed into a nuclear reactor, the range of materials and geometrical arrangements that a criticality analyst has to deal with is much wider. Likely, processing conditions involve the presence of gloveboxes, walls, floors, surrounding equipment, the operators themselves, etc., involving a wider range of material, shapes, and arrangements than a nuclear reactor analyst has to deal with.

One result of this is that there is a much higher uncertainty in  $k$ -effective in criticality safety analyses, and this uncertainty does not really disappear with processing experience. That is, in almost all criticality safety analyses—in contrast with reactor analyses—there is almost no chance that the resulting predictions of the  $k$ -effective of the process will ever be verified experimentally. This leads to a much higher dependence on the validation of criticality safety calculations with externally produced benchmark experiments. This has resulted in a tremendous industry-wide effort to document, evaluate, and distribute reliable information on available critical experiment benchmarks over a wide range of fuels, enrichments, involved materials, and geometries.

A second result of this relative complexity of materials and geometries is a further emphasis on use of Monte Carlo calculational methods, with their increased geometrical modeling capabilities.

Difference #4: Criticality safety is a smaller professional field.

Worldwide, not as many nuclear engineering professionals are involved in criticality safety analysis as are involved in reactor analysis. More importantly, perhaps, these limited number of professionals tend to be more widely dispersed into smaller groups than are usually found in reactor design and analysis organizations.

The result of this is that there is a much higher emphasis on the production and widespread use of ANSI/ANS professional standards to guide day-to-day work. Table 21.2 lists the ANSI/ANS standards that are currently active. Also, there is higher emphasis on individual involvement in professional activities, principally through participation in activities of the American Nuclear Society (ANS) Criticality Safety Division, through participation in the activities involved in production of the ANSI consensus standards (through ANSI/ANS 8), and through a more formalized qualification and certification process for criticality safety engineers. (These three emphases have combined recently in the production of

**TABLE 21.2**

Currently Active ANSI/ANS Criticality Safety Standards

ANSI/ANS #	Title
ANSI/ANS-8.1-2014	Nuclear criticality safety in operations with fissionable materials outside reactors
ANSI/ANS-8.3-1997	Criticality accident alarm system
ANSI/ANS-8.5-1996	Use of borosilicate–glass raschig rings as a neutron absorber in solutions of fissile material
ANSI/ANS-8.6-1983	Safety in conducting subcritical neutron-multiplication measurements in situ
ANSI/ANS-8.7-1998	Nuclear criticality safety in the storage of fissile materials
ANSI/ANS-8.10-1983	Criteria for nuclear criticality safety controls in operations with shielding and confinement
ANSI/ANS-8.12-1987	Nuclear criticality control and safety of plutonium–uranium fuel mixtures outside reactors
ANSI/ANS-8.14-2004	Use of soluble neutron absorbers in nuclear facilities outside reactors
ANSI/ANS-8.15-1981	Nuclear criticality control of special actinide elements
ANSI/ANS-8.17-2004	Criticality safety criteria for the handling, storage, and transportation of LWR fuel outside reactors
ANSI/ANS-8.19-2005	Administrative practices for nuclear criticality safety
ANSI/ANS-8.20-1991	Nuclear criticality safety training
ANSI/ANS-8.21-1995	Use of fixed neutron absorbers in nuclear facilities outside reactors
ANSI/ANS-8.22-1997	Nuclear criticality safety based on limiting and controlling moderators
ANSI/ANS-8.23-2007	Nuclear criticality accident emergency planning and response
ANSI/ANS-8.24-2007	Validation of neutron transport methods for nuclear criticality safety calculations
ANSI/ANS-8.26-2007	Criticality safety engineer training and qualification program
ANSI/ANS-8.27-2008	Burnup credit for LWR fuel

the ANSI/ANS 8.26 standard for training of nuclear criticality safety engineers, developed through an ANS working group.)

Difference #5: Importance of conservatism in criticality safety analyses.

Because criticality safety analyses are concerned with establishing safety (rather than with designing a machine that will work according to design specifications), criticality safety analyses are much more concerned with conservatism than with accuracy. More specifically, the ultimate result of a criticality analysis of a particular state is an answer to the question “Is it subcritical?” In answering this question, there are logically two errors that might occur: either (1) a subcritical system is erroneously declared to be critical or (2) a critical system is erroneously declared to be subcritical. In criticality safety practice, these two errors are not treated equally. The first error is of little consequence (except possible loss of some production in the process), but the second error is avoided at almost any cost.

The result of this situation is an extremely conservative biasing of criticality safety analyses toward the assumption of criticality. The most obvious manifestation of this is that the end result of the (very involved and formal) process of validating an analysis method is a determination of an upper subcritical limit (USL), which represents the value of the  $k$ -effective prediction of the particular method that is going to be considered to represent a critical state. It sometimes comes as a surprise to reactor analysts to know that a  $k$ -effective of 1.000 is not where criticality is defined in a criticality safety analysis, but a much lower value—sometimes as low as 0.80 and seldom higher than 0.95 (except for extremely well-understood processes).

With these differences in mind, we will now explore the field of criticality safety analysis.

The professional practice of criticality safety involves establishing the safety of a proposed operation—usually one involving hands-on processing by human operators. In the broadest terms, doing this involves three steps:

1. Determining the most reactive states that the system under investigation can credibly obtain
2. Establishing to a reasonable certainty that all of these states are subcritical
3. Determining and documenting the process design features and process controls necessary to guarantee that more reactive states are incredible

We will look at each of these in turn.

### 21.3.1 First Step: Determining Credible States

The most important step of a criticality safety evaluation is not computational. Determining what to analyze is a bigger part of the effort than the analysis itself. Finding out whether a clearly defined configuration of materials is subcritical or not is the easy part—and in fact, no criticality accident has ever been caused because a recognized potential state of a system was misdiagnosed. The tough part is contingency analysis: the determination of which configurations (system states) are likely to occur in the life of a process, which are unlikely to occur but are credible, and which are incredible, and the determination of what these words (“unlikely,” “likely,” “incredible,” “credible”) mean.

The beginning of this analysis process is a full, but focused, description of the process being analyzed: the equipment being utilized (including material makeup), the geometric arrangements allowed, and—most importantly—the steps of the process. The analyst must be especially careful in this description because it determines the scope of the analysis—the range of conditions that are covered in the analysis and that, therefore, will be allowed in the subsequent processing activities.

Once this description is in place, the next stage of this analysis is to determine the normal and credible accident states of the system. The methods for performing this stage vary from formal to informal, but the most common way of approaching this task is through a parametrically driven “what if?” analysis.

#### 21.3.1.1 “MAGICMERV”: Physical Parameters Affecting $k$ -Effective

The parameters that guide the contingency analysis are the nine physical factors that affect the critical state of the system. They are easily remembered using the acronym “MAGICMERV”: mass, absorption, geometry, interaction, concentration, moderation, enrichment, reflection, and geometry.

**M** (Mass): the mass of the fissile isotope(s) in each unit being processed. Usually the higher the fissile unit mass, the higher the  $k$ -effective of the system. This is the most important of the nine parameters because it is the one that is most commonly controlled.

**A** (Absorbers): the loss of neutron absorbers that are specifically depended on for criticality control. Absorber loss usually increases the  $k$ -effective of the system. As previously stated, this factor is not as important as might be assumed—especially by those

with backgrounds in reactor analysis—because neutron absorbers are not usually specifically used for criticality control. However, when they are, they must be very carefully handled (reflected in the three ANSI/ANS standards dedicated to the subject) because of the need to assure that the absorbers stay where they are supposed to be. A secondary aspect of this parameter is that many situations are implicitly depending on neutron absorbers for criticality control through neutron-absorbing nuclides that are not added to the design explicitly, but which are normally present in the equipment (e.g., the boron in Pyrex glass equipment). ANSI/ANS-8.1 specifically allows credit to be taken for such “accidental” absorbers, but it is still incumbent on the analyst to demonstrate that the absorber material in question will in fact be present. (In many analyses, absorbers such as these are eliminated from the computational model of the system to avoid the need to guarantee their presence.)

- G** (Geometry): this parameter involves the shape of the fissile unit. The most reactive shape for an individual fissile unit is a sphere; for cylindrical objects, the most reactive shape is an aspect ratio (height/diameter ratio) of about 1 (buckling equations give the optimum as 0.924 for bare cylinders). For arrays of units, the optimum H/D can be situation-dependent.
- I** (Interaction): this parameter considers interaction of the fissile unit with other fissile material, often other identical units arranged in a regular two- or three-dimensional array. The three typical situations that must be considered under this category are (1) lattice arrangements of multiple fissile units (including consideration of stacking and lifting fissile units over the lattice); (2) consideration of how close other fissile material can be brought up to the system being analyzed; and (3) the presence of “hold-up” which is small, cumulative collections of fissile material that, over time, may collect nearby (e.g., gloveboxes or room corners).
- C** (Concentration): this parameter involves controls over the allowable concentrations of fissile material solutions in liquid processing steps. This is considered separate from Mass (#1) and Volume (#9) because it can be independently measured, therefore independently controlled. Because the word “concentration” is generally associated with liquids, the concept of “density” is considered under this parameter as well for solid fissile matrices.
- M** (Moderation): this parameter involves the presence of material intermixed with the fissile material that serves to slow neutrons down from their fast birth energies to the thermal energies where the  $\eta$  value is again high, but where leakage is much lower.
- E** (Enrichment): this is the mass fraction of the fissile isotope within its element; the most common use of this is the fraction of  $^{235}\text{U}$  of the total uranium in the fuel. It is almost always true that higher enrichment results in higher  $k$ -effective, although there are a few known situations for which this is not true.
- R** (Reflection): this refers to the presence of nonfissile material around fissile units that tend to increase reactivity by returning escaping neutrons back into the unit. Although water (often in the form of the bodies of people) is the most common reflection material, any surrounding material (e.g., structural components, floor, walls, shielding) can serve as reflectors that must be taken into account.
- V** (Volume): this is the volume of the container that holds fissile material. This is usually most important because of the limited amount of room available for moderation in case of flooding and because of its effect on the minimum spacing between fissile units.

### 21.3.1.2 Parameter-Driven Analysis: Normal Conditions, Contingencies, Controls

These nine physical parameters drive the “what if?” analysis through three iterative categories, as shown in the columns of Table 21.3. The parameters themselves are listed in the first column of the table.

The last column of the table records the controls—usually passive design features, active engineered safety systems, or administrative controls implemented through procedures and postings—that will be used to limit the impact of each of the parameters. Usually only a few of the parameters are actually controlled, with the remainder being “not applicable” (e.g., concentration controls for solid material processing) or “not controlled,” indicating that the specified parameter is allowed to achieve the most reactive contribution that is credible (e.g., enrichments up to 100%, no absorbers present). For the parameters that are controlled, it is common for multiple controls to be listed—for example, procedural controls, postings, and training on both of these.

The second column of the table records the parameter values from an assessment of the “normal” state of the system being analyzed. Despite its name, “normal” does not refer to the everyday values of the parameters, but rather records the most reactive state that is

**TABLE 21.3**

Example Contingency Analysis Table

Critical Parameter	Normal	Contingency	Controls
Mass	20% Overmass in canister	100% Overmass in canister	Postings define a maximum mass of 3.1 kg per canister; Operators trained on procedures and postings
Absorber	N/A	N/A	Analysis neither relies on nor uses absorbers
Geometry	H/D = 0.92	N/A	Process uses cylinders with H/D = 0.92; Purchasing procedures specify canister dimensions
Interaction	Single canister	Extra canister	Procedure limits carrier to one canister; Operators trained on procedures
Concentration Moderation	N/A 65% Polyethylene by volume limiting matrix	N/A N/A	Process does not involve solutions Procedure requires that containers be closed and locked; Purchasing specifications require water-tight containers
Enrichment	N/A	N/A	Worst possible enrichment (100% Pu-239) assumed
Reflection	Canister held against a surface or corner	Canister fully reflected	Building procedures require sprinkler system maintenance to assure proper operation; Sign-off maintenance sheet kept on sprinkler system
Volume	10% Increase in canister volume	N/A	Purchasing rules require maximum size of canisters is 4 liters; Operators trained to recognize normal canister

expected to be achieved during the life of a facility (usually considered to be 50–100 years) with the specified controls (in the fourth column) in place. A formal probability and risk assessment is not required to quantify this judgment, but this step instead relies on a qualitative application of engineering judgment. Again, the assessment of “not applicable” can be applied, but the existence of specific controls and the “not controlled” entries in the final column require that an assessment be made.

Each row of the third column lists the expected worst-case “credible contingency” involving the specified parameter with the specified (third column) controls in place. Although it is expected that each contingency will involve violation of at least one of the controls (otherwise the specified contingency would be considered normal), engineering judgment must again be utilized.

### 21.3.2 Second Step: Determination of the Subcriticality of Normal and Contingency States

The next step in the criticality safety analysis is to determine the criticality state of the normal state and each of the contingency states. This determination is required by ANSI/ANS-8.1, which states in paragraph 4.2.2:

Double Contingency Principle. Process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

Based on this double contingency principle, the normal case and each of the contingencies—considered one at a time—must be determined to be subcritical to establish the safety of the operation. The method that is best used to establish this subcriticality depends on the complexity of the parametric case.

The simplest, least complex way of establishing subcriticality is to use single parameter limits from ANSI/ANS-8.1 tables or figures. (See example in Tables 21.4 and 21.5 for aqueous solutions and metal units, respectively.) The values in these tables represent the limiting critical values for individual parameters, with the other parameters assumed to be at their worst possible values for single units (including interaction with other fissile materials that are bounded by complete water reflection). Also covered by data in the ANSI/ANS-8.1 standard are simple double parameter limits, expressed as a curve. For example, Figures 21.8 and 21.9 give double parameter limits for slab thickness and volume versus enrichment for uranium–water lattices.

In addition to the tables and figures from ANSI/ANS-8.1, there are several handbooks that contain similar tables and figures that cover other fissile isotopes in other relatively simple parametric combinations. These values are also based on experimental results of critical experiments (although most also utilize validated calculational methods as well). The key to the use of these handbook values is the simplicity of the parametric combinations which, in real situations, translates into the most reactive credible values of the remaining parameters. Use of the ANSI/ANS-8.1 or handbook values usually result in more restrictive limits than might otherwise be achievable from a more detailed analysis of the proposed process.

The most complicated analysis technique—but one that results in the least restrictive controls—involves the particular analysis of the unique situation with complicated geometry models using Monte Carlo analysis tools.

**TABLE 21.4**

Single Parameter Limits for Uniform Aqueous Solutions of Fissile Nuclides

Parameter	Subcritical Limit for Fissile Solute				
	<sup>233</sup> UO <sub>2</sub> F <sub>2</sub> [2]	<sup>233</sup> UO <sub>2</sub> (NO <sub>3</sub> ) <sub>2</sub> [2]	<sup>235</sup> UO <sub>2</sub> F <sub>2</sub> [3]	<sup>235</sup> UO <sub>2</sub> (NO <sub>3</sub> ) <sub>2</sub> [3]	<sup>239</sup> Pu(NO <sub>3</sub> ) <sub>4</sub> [4]
Mass of fissile nuclide, kg	0.54	0.55	0.76	0.78	0.48
Diameter of cylinder of solution, cm	10.5	11.7	13.7	14.4	15.4
Thickness of slab of solution, cm	2.5	3.1	4.4	4.9	5.5
Volume of solution, L	2.8	3.6	5.5	6.2	7.3
Concentration of fissile nuclide, g/L	10.8	10.8	11.6	11.6	7.3
Atomic ratio of hydrogen to fissile nuclide <sup>a</sup>	2390	2390	2250	2250	3630
Areal density of fissile nuclide, g/cm <sup>2</sup>	0.35	0.35	0.40	0.40	0.25

Source: ANS 8.1 Working Group, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors. American Nuclear Society, La Grange Park, IL: 1993. With permission.

<sup>a</sup> Lower limit

**TABLE 21.5**

Single Parameter Limits for Metal Units

Parameter	Subcritical Limit for		
	<sup>233</sup> U [2]	<sup>235</sup> U [3]	<sup>239</sup> Pu [4]
Mass of fissile nuclide, kg	6.0	20.1	5.0
Cylinder diameter, cm	4.5	7.3	4.4
Slab thickness, cm	0.38	7.3	0.65
Uranium enrichment, wt% <sup>235</sup> U	–	5.0	–
Maximum density for which mass and dimension limits are valid, g/cm <sup>3</sup>	18.65	18.81	19.82

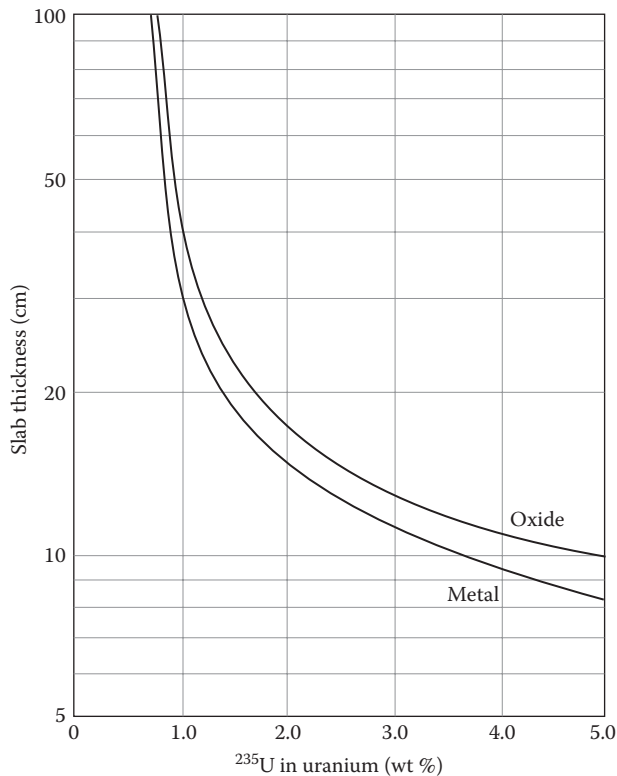
Source: ANS 8.1 Working Group, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors. American Nuclear Society, La Grange Park, IL: 1993. With permission.

### 21.3.2.1 Most Commonly Used Computer Code Systems

The two most common code systems used for the problem-specific criticality analysis are the SCALE code sequence CSAS25 (utilizing KENO-Va) from the Oak Ridge National Laboratory and the MCNP5 computer system developed by the Los Alamos National Laboratory. Each of these allows the user a wide range of capabilities, the most important of which are

1. The ability to model 3D geometries in as complete a geometric description as desired
2. The availability of neutron reaction cross sections from a wide selection of isotopes
3. Resulting *k*-effective values for the modeled system, including an estimate of the statistical standard deviation of the *k*-effective





**FIGURE 21.8**

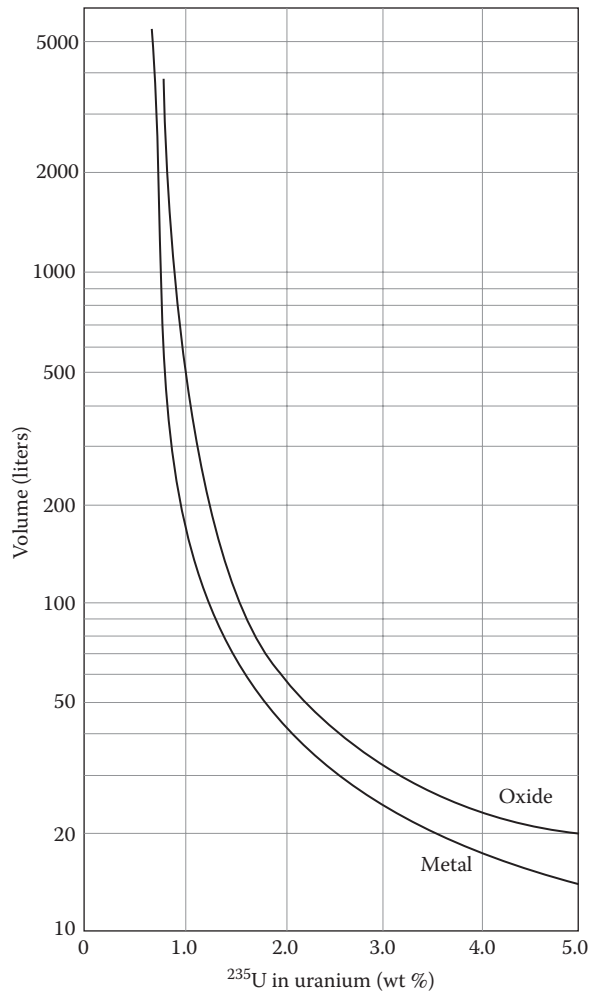
Slab thickness limit for uranium–water lattices. (Data from ANS 8.1 Working Group. 1993. *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*. ANSI/ANS-8.1-1983[R1988]. La Grange Park, IL: American Nuclear Society.)

4. Other statistical measures by which the analyst can judge the appropriate statistical convergence of the result
5. Other physical results from the solutions (e.g., average energy of fission-inducing neutron) that may be useful to the analyst to characterize the system

For more details about the use of these systems, see X-5 Monte Carlo Team 2008 and SCALE 2009.

### 21.3.2.2 Validation of Computational Methods

Because the results of a particular criticality safety calculation are unlikely to be experimentally verifiable, it is important to have a formal validation of the computer methods—a term that formally includes the computer code itself and the data used by the code. The basic approach to this task is to compare the computational tool against as many experimental results that are “similar” to the system being analyzed as is possible; again, the validation effort is consistent with our overall goal of the computation: to assure ourselves that conditions predicted to be subcritical by the method would be actually expected to be subcritical in reality. There are two primary results of the validation process:



**FIGURE 21.9**

Volume limit for uranium–water lattices. (Data from ANS 8.1 Working Group. 1993. *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*. ANSI/ANS-8.1-1983[R1988]. La Grange Park, IL: American Nuclear Society.)

1. An “area of applicability” (AOA) for the methodology which sets out the range of physical situations for which the methodology has been validated
2. A USL which marks the maximum value of a calculated  $k$ -effective using the methodology that can be considered subcritical

The main tasks within the validation process are

1. Selection of appropriate critical benchmark experiments
2. Establishment of the AOA covered by the selected experiments
3. Calculation of  $k$ -effective for each of the selected critical benchmarks using the same methodology to be used in the analysis

4. Determination of the USL through both statistical analysis of the calculated  $k$ -effectives—to determine the basic bias and bias uncertainty of the calculational method—and inclusion of other factors of safety based on engineering judgment of the quality of coverage of the AOA

The first task, the selection of appropriate benchmark experiments, is greatly aided by the International Criticality Safety Benchmark Evaluation Project (ICSBEP). This project is managed through Idaho National Laboratory (INL) and involves nationally known criticality safety experts from various national laboratories in the United States.

From the ICSBEP website, the purpose of the ICSBEP is to

- Identify a comprehensive set of critical benchmark data and, to the extent possible, verify the data by reviewing original and subsequently revised documentation, and by talking with the experimenters or individuals who are familiar with the experimenters or the experimental facility.
- Evaluate the data and quantify overall uncertainties through various types of sensitivity analysis.
- Compile the data into a standardized format.
- Perform calculations of each experiment with standard criticality safety codes.
- Formally document the work into a single source of verified benchmark critical data.

The work of the ICSBEP is documented as the *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, which is distributed by INL as a set of annually updated CD or DVDs. The handbook currently contains 464 evaluations covering more than 4000 critical, near-critical, or subcritical configurations. Table 21.6 gives an overview of the experiments available from the database.

Selection of experiments from the ICSBEP database for a given experiment or set of experiments is greatly facilitated by the use of DICE, which is a Java-based tool that allows users to sort through the thousands of experiments in the database according to specified search parameters.

Once an appropriate set of benchmark experiments has been identified, the next task is to categorize the experiments to determine the AOA. Generally, this is done through determining the range of the experiments in several categories of physical parameters (usually a subset of the MAGICMERV parameter list), including

- The fissile isotope(s) present.
- Other important materials that are present that may affect the spectrum. Principally, this would include important absorbers, moderators, and reflectors. Often, in addition to the qualitative listing of the principal moderators, a quantitative measure of moderation of the system is also given, usually in the form of an H/X ratio—giving the ratio of hydrogen atoms to fissile atoms.
- An indication of the geometric shape of the basic fissile unit.
- An indication of the interaction of multiple fissile units—usually either a single unit, a lattice arrangement (i.e., of fuel pins), or an array of the basic units.
- A quantitative measure of the neutron spectrum energies. This is often quantified through a parameter such as the energy of the average lethargy of neutrons causing fission (EALF) or the average energy group of fission neutrons (for multigroup calculational methods).

**TABLE 21.6**

Distribution of Configurations in the ICSBEP Data Handbook

Plutonium metal configurations	121
Plutonium solution configurations	577
Plutonium compound configurations	36
<b>Total Plutonium Configurations</b>	<b>734</b>
Highly enriched uranium metal configurations	582
Highly enriched uranium solution configurations	536
Highly enriched uranium compound configurations	289
Highly enriched uranium miscellaneous configurations	7
<b>Total Highly Enriched Uranium Configurations</b>	<b>1414</b>
Intermediate/mixed enrichment uranium metal configurations	47
Intermediate/mixed enrichment uranium solution configurations	64
Intermediate/mixed enrichment uranium compound configurations	156
<b>Total Intermediate/Mixed Enrichment Uranium Configurations</b>	<b>269</b>
Low enriched uranium metal configurations	87
Low enriched uranium solution configurations	117
Low enriched uranium compound configurations	1331
Low enriched miscellaneous configurations	60
<b>Total Low Enriched Uranium Configurations</b>	<b>1595</b>
<sup>233</sup> U metal configurations	11
<sup>233</sup> U solution configurations	227
<sup>233</sup> U compound configurations	6
<b>Total <sup>233</sup>U Configurations</b>	<b>244</b>
Mixed uranium–plutonium metal configurations	52
Mixed uranium–plutonium solution configurations	86
Mixed uranium–plutonium compound configurations	301
Mixed uranium–plutonium miscellaneous configurations	85
<b>Total Mixed Uranium–Plutonium Configurations</b>	<b>526</b>
Special isotope metal configurations	20
<b>Total Special Isotope Configurations</b>	<b>20</b>

Source: ICSBEP: International Critically Safety Benchmark Evaluation Project (<http://icsbep.inel.gov/>) (Accessed date 7/26/2014).

The final two tasks involve the calculation of the USL through the standard equation given by ANS 8.1 Working Group 1993

$$\text{USL} = 1 + \text{bias} - \text{bias uncertainty} - \text{MSM} \quad (21.28)$$

where:

bias is the difference between the average calculated  $k$ -effective and the experimentally determined  $k$ -effective of the benchmark (positive bias generally conservatively set to zero in the equation).

bias uncertainty is a measure of uncertainty in the bias based on the standard deviation of the fit to calculated and experimental  $k$ -effective.

MSM is the “minimum subcritical margin,” a fixed value (usually 0.02–0.05) to provide a minimum safety margin for the USL. This value can also be increased, based on engineering judgment, if the set of benchmark experiments is deemed to inadequately represent one or more important physical parameters of the system to be analyzed. Both the bias and bias uncertainty are usually found from a

straightforward statistical analysis of the errors of the method in reproducing the known  $k$ -effective values (usually 1.000) of the benchmarks.

### 21.3.3 Third Step: Translation of Results into Documented Process Controls

Once the analysis limits of the physical parameters have been determined, the final step involves determining and documenting the allowed process conditions and required controls. The controls themselves usually take the form of one of the following:

- Passive design features: Built-in design features (e.g., dividers, holders, spacers) that restrict operations to implement a control
- Engineered safety features: Safety systems that actively monitor a process and provide an automatic action or warning when a control is violated
- Administrative control: A line in a procedure (usually denoted as criticality safety significant), a general building or site rule, required form to fill out during operations, etc., that implement a control

The ordering above is not arbitrary but reflects the preference described in ANSI/ANS-8.1, paragraph 4.2.3. The three approaches are listed in order of their effectiveness but, unfortunately, in opposite order of their usual cost of implementation. Therefore, the exact controls chosen for a particular process follow the familiar trade-off of cost versus safety benefit.

The three steps of the criticality safety evaluation—contingency analysis, limit determination, and control specification—are presented in a document generally referred to as a Nuclear Criticality Safety Evaluation (NCSE) (although some sites separate out the first step into a separate document referred to as a Nuclear Criticality Safety Assessment (NCSA)). Within a given organization or processing site, the structure and format of NCSA/NCSEs are usually strictly proscribed for consistency of development and ease of use.

---

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# 22

## *Heat Transfer, Thermal Hydraulic, and Safety Analysis*

Shripad T. Revankar

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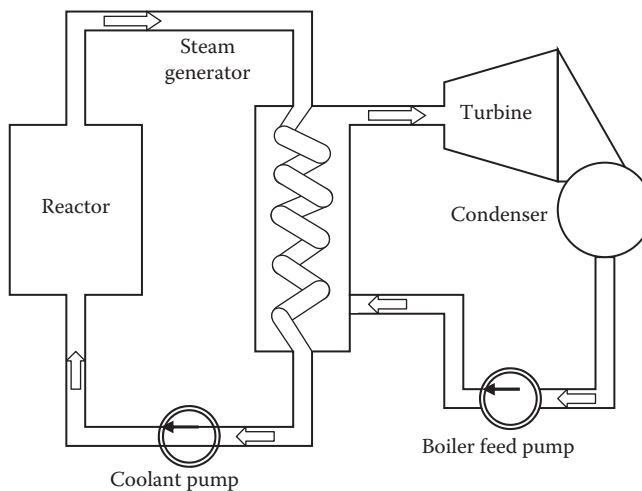


## 22.1 Introduction

The overall objectives of this chapter are to (1) provide a background on heat transfer in reactor systems; (2) describe methods of analysis employed in the reactor thermal-hydraulics and safety with basic analysis processes and tools; and (3) provide analysis examples, sources of information, and computer codes used for detailed reactor thermal-hydraulics and safety analysis.

The nuclear power reactor is a source of thermal energy which is used to produce electricity through the conventional steam heat engine process known as the Rankine cycle. This is shown schematically in Figure 22.1. The thermal energy is generated within the nuclear fuel by fission and is transferred from the fuel by a fluid medium called the reactor coolant. This fluid medium may be boiling water, in which case the steam may be used directly in the turbine such as in a boiling water reactor (BWR) or it may act as an intermediate heat transport medium, giving up its heat to raise steam in external heat exchangers called boilers or steam generators such as in a pressurized water reactor (PWR). In some reactors, such as the high-temperature gas-cooled reactor (HTGCR),  $\text{CO}_2$ , or helium gas is heated to high temperature which is then used to drive turbines to produce electricity in a cycle referred as Brayton cycle.

This chapter deals with heat transfer and thermal-hydraulic analysis of the process and systems that transport heat energy from the nuclear reactor and convert this heat energy into useful work, generally electrical energy. This entails a number of processes and inter-related systems. The basic concept of the nuclear power plant (NPP) heat transport system is set by the overall objective to remove heat from the reactor core at high thermal efficiency and to allow the maximum amount of energy to be extracted from the fuel without surpassing safe limits. The reactor heat transport system must provide continuous coolant flow to remove heat from reactor core during normal or abnormal operation and provide adequate core cooling to remove decay heat produced by the fuel when the reactor is shut down. Even if the reactor is shut down, the decay heat from the fuel must be adequately removed. Failure to remove the fuel decay heat can overheat the core and can lead to accident with release of radioactive material to environment with catastrophic consequences



**FIGURE 22.1**  
Basic power reactor schematic arrangement.

**TABLE 22.1**

Reactor Coolants

Coolant Type	Neutron Economy	Corrosive	Heat Capacity/ Heat Transfer Coefficient	Activation	Vapor Pressure	Cost
He (gas)	Good	Good	Low	Low	High	Higher
CO <sub>2</sub> (gas)	Good	OK—except at high temp.	Low	Low	High	Low
H <sub>2</sub> O	Moderate	OK—transports corrosive materials	High	Yes, but has short half-life	High	Low
D <sub>2</sub> O	Excellent	OK—transports corrosive materials	High	Yes, produces tritium	High	High
Organic <sup>a</sup>	Slightly better than H <sub>2</sub> O	Excellent	High	None	Low	Moderate
Liquid metal <sup>b</sup> (e.g., NaK, Pb)	Not great	OK—need compatible materials	High	High with long half-life	Very low	High

<sup>a</sup> Suffers radiochemical damage.

<sup>b</sup> High melting point, low cross section at high energy; good for fast breeder reactors.

to public and environment. Hence, safety analysis is of paramount importance in reactor power plant design and operation. This chapter outlines safety features of an NPP and discusses safety analysis methods.

Depending on the type of reactor (PWR, BWR, pressurized heavy water reactors [PHWRs], HTGCR, and liquid metal fast breeder reactors [LMFBRs]), the heat removal mechanism involves different coolants, single- or two-phase flow conditions, heat flux level, and system geometry. Typical reactor coolants are CO<sub>2</sub> gas, helium gas, ordinary water, heavy water, organic fluid, and liquid metal. The coolant may be a liquid, two-phase liquid/vapor mixture, or gas. The desirable features of reactor coolants are high heat capacity, good heat transfer properties, low neutron or activation, low operating pressure requirement at high operating temperatures, noncorrosive to fuel cladding and coolant system, and low cost. Table 22.1 gives a summary of the reactor coolants.

Though the fission in the reactor core is analyzed by the nuclear consideration, the heat generated by the reactor through fission and its use in the generation of power for a given reactor core is largely limited by thermal processes and material properties rather than by nuclear considerations. The safety analysis of the reactor during normal and abnormal operational conditions involves detailed thermal-hydraulics analysis.

## 22.2 Reactor Heat Generation

### 22.2.1 Fission Energy in Reactor

The amount of energy released in fission of a nucleus is related to the net decrease in the mass in the fission process. For uranium-235, the average energy released in fission is 200 MeV or 32 pJ. This energy is equivalent to three million times as much energy released with the combustion of carbon of same mass as one uranium-235 atom.

The fission energy released appears as kinetic energy of the fission products; kinetic energy of the newborn neutrons; energy of gamma rays emitted during fission; and later by fission products, neutrino, and beta rays emitted by the fission products following the fission. The energy distribution among these various forms is shown in Table 22.2. About 80% of fission energy is in the form of kinetic energy, which manifests as heat and is deposited in the fuel element. The newborn neutrons carry 2.5% of the energy in the form of kinetic energy, and this energy is deposited in the moderator as heat energy. The gamma rays released instantaneously referred as prompt gamma rays during the fission have about 2.5% of energy, and this energy is transferred to fuel and core structures. The delayed energy from beta rays and gamma rays which amounts to 6% of the total is deposited in fuel and structures. All the neutrino associated with beta decay carry about 5% of energy, and this energy escapes the reactor vessel and is nonrecoverable, but this loss is compensated by the energy released, about 3.5% within a reactor by neutron capture reactions.

In general, the heat generated in the fuel per fission is dependent upon the exact range of particles emitted which in turn depend on the materials used in the reactor and the reactor core configuration. The energy release per fission is slightly dependent on the type of fissile material in the core. The fissile materials uranium-233 and plutonium-239 have slightly different energy release ( $E$ ) than uranium-235 as per Lamarsh (1966):  $E$  (U-233) =  $0.98 E$  (U-235) and  $E$  (Pu-239) =  $1.04E$  (U-235). In the absence of accurate information it can be assumed that approximately 90% of the total energy per fission is produced in the fuel,

TABLE 22.2

Distribution of Fission Energy for U-235

Type		Form	Emitted Energy (MeV)	Recoverable Energy (MeV)
Fission	I Instantaneous energy	Kinetic energy of fission fragments	168	168
		Kinetic energy of newly born fast neutrons	5	5
		Prompt gamma energy	7	7
	II Fission product decay—delayed energy	Beta rays energy	7	7
		Gamma ray energy	6	6
		Neutrinos energy—with beta decay	10	—
		Kinetic energy of delayed neutrons	0.4	0.4
Excess neutron gamma capture reaction	III Instantaneous and delayed energy	Nonfission reactions due to excess neutrons and the proceeding beta, gamma decay energy	—	7

Sources: Glasstone, S. and Sesonske, A., *Nuclear Reactor Engineering*, 3rd edn., Von Nostrand Reinhold, New York, 1981; El-Wakil, M.M., *Nuclear Heat Transport*, American Nuclear Society, La Grange Park, IL, 1978.

about 4% is produced in the moderator, 1% in structural, and 5% is carried away by the neutrinos.

As an example, we can calculate the amount of U-235 consumer per day in a thermal reactor operating at power  $P$  megawatts. Assuming recoverable energy of 200 MeV (32 pJ) per fission, the fission rate for a year producing  $P$  megawatts power is

$$\begin{aligned} \text{Fission rate} &= P(\text{MW}) \frac{10^6 \text{ (J)}}{1 \text{ (MWs)}} \times \frac{\text{Fission (\#)}}{32 \times 10^{-12} \text{ (J)}} \times \frac{86,400 \text{ (s)}}{1 \text{ (day)}} \times \frac{365 \text{ (days)}}{1 \text{ (year)}} \\ &= 9.855 \times 10^{23} P (\#/ \text{year}) \end{aligned} \quad (22.1)$$

$$\begin{aligned} \text{Burnup rate per year} &= 9.855 \times 10^{23} P (\#/ \text{year}) \times \frac{235 \text{ (g/mol)}}{6.022 \times 10^{23} (\#/ \text{mol})} \\ &= 384.58 P (\text{g}) \end{aligned} \quad (22.2)$$

In reality the fissile material is consumed both in fission and in radiative capture. The radiative capture rate is given as

$$\text{Radiative capture rate} = \alpha \times \text{Fission rate} \quad (22.3)$$

where  $\alpha$  is the ratio of microscopic absorption cross section to the microscopic fission cross section and for U-235 its value is 0.169. Thus,

$$\text{Consumption rate of U-235} = 384.58(1 + \alpha)P (\text{g}) = 449.57(\text{g}) \quad (22.4)$$

### 22.2.2 Volumetric Heat Generation in Fuel

In the reactor, the fission generates neutrons with large energies (2–10 MeV), and in a thermal reactor these neutrons are slowed down to thermal energies (~0.025 eV). The thermal neutrons have a Maxwellian energy distribution dependent upon the temperature of the medium. In the low energy range, the neutron absorption cross section for fissile nuclei is inversely proportional to the squarer of the neutron energy or is inversely proportional to neutron speed.

The average volumetric heat generation rate  $q'''$  (W/m<sup>3</sup>) in the reactor core is given as the product of total reaction rate  $R$  (1/s) and the energy per reaction  $G_f$  (J). The reaction rate  $R$  is calculated as

$$R = N_f \int \sigma_f(E) \phi(E) dE \quad (22.5)$$

where:

$N_f$  is the density of the fissionable fuel (nuclei/cm<sup>3</sup>)

$\sigma_f(E)$  is the microscopic cross section for the fissionable fuel (cm<sup>2</sup>) at energy  $E$

$\phi(E)$  is the flux of the neutron (neutron/s cm<sup>2</sup>) having energy between  $E$  and  $E + dE$

The total reaction rate is obtained by taking integral over entire range of neutron energy (0,  $\infty$ ).

By defining flux of all neutrons,  $\phi$ , and effective microscopic cross section,  $\sigma_f$ , as

$$\phi = \int \phi(E) dE \quad \text{and} \quad \sigma_f \phi = \int \sigma_f(E) \phi(E) dE \quad (22.6)$$

The volumetric heat generation rate  $q'''$  is given by

$$\begin{aligned} q''' &= G_f N_f \sigma_f \phi \quad (\text{MeV/s cm}^3) \\ &= 1.602 \times 10^{-10} G_f N_f \sigma_f \phi \quad (\text{kW/m}^3) \end{aligned} \quad (22.7)$$

Since some of the heat is generated in non fuel due to various capture reactions, the overall power generation in the core  $Q$  (W) is related to the  $q'''$  as

$$Q = \frac{q''' V_{\text{fuel}}}{\gamma}, \quad (22.8)$$

where:

$V_{\text{fuel}}$  is the fuel volume

$\gamma$  is the fraction of the power generated in the fuel

The core power density  $Q'''$  (W/m<sup>3</sup>) is related to the  $Q$  and  $q'''$  as

$$Q''' = \frac{Q}{V_{\text{core}}} = q''' \left( \frac{V_{\text{fuel}}}{V_{\text{core}} \gamma} \right) \quad (22.9)$$

As an example, one can calculate the volumetric thermal source strength in a reactor core that has neutron flux  $10^{14}$  neutron/s cm<sup>2</sup> with the effective microscopic cross section for the fissionable fuel of 350 barn and density of the fissionable fuel,  $2 \times 10^{21}$  nuclei/cm<sup>3</sup>.

Given  $N_f = 2 \times 10^{21}$  nuclei/cm<sup>3</sup>,  $\phi = 10^{14}$  neutron/s cm<sup>2</sup>,  $\sigma_f = 350 \times 10^{-24}$  cm<sup>2</sup>.

Therefore,  $q''' = 200 \times 2 \times 10^{22} \times 350 \times 10^{-24} \times 10^{14}$

$$= 14 \times 10^{15} \text{ MeV/s cm}^3$$

$$= 14 \times 10^{15} \times 1.602 \times 10^{-10} = 22.680 \times 10^5 \text{ kW/m}^3$$

$$= 14 \times 10^{15} \times 1.5477 \times 10^{-28} = 21.668 \times 10^7 \text{ Btu/h ft}^3$$

### 22.2.3 Heat Generation during Transient

Up to 50% of the reactor power, the heat generation rate can be assumed to follow proportional to the neutron flux. However, if the reactor is shut down, the reactor core produces heat at 7% of the reactor power due to delayed neutron fission, fission product decay, and activation products from neutron capture. The heat generated from fission product decay due to high-power operation can be temporarily higher than the heat generated from the fission product decay at low-power operation. Because of the decay heat shutdown cooling has to be provided for the reactor core. Immediately after the shutdown the reactor power decreases exponentially from 7% of the core power with a period of 80 s. The heat generation from fission product decay is a result of beta and gamma emission from fission

product with decay half-life ranging from micro seconds to million years. Decay heat is the principal reason of safety concern in light water reactors (LWRs), and it is the source of 60% of radioactive release risk worldwide.

The decay heat power comes mainly from five sources: (1) unstable fission products, which decay via  $\alpha$ ,  $\beta^-$ ,  $\beta^+$ , and  $\gamma$  ray emission to stable isotopes; (2) unstable actinides that are formed by successive neutron capture reactions in the uranium and plutonium isotopes present in the fuel; (3) fissions induced by delayed neutrons; (4) reactions induced by spontaneous fission neutrons; (5) structural and cladding materials in the reactor that may have become radioactive. Heat production due to delayed neutron-induced fission or spontaneous fission is usually neglected. Activation of light elements in structural materials plays a role only in special cases.

### 22.2.3.1 Integrated Beta and Gamma Emission Rates

The heat load from decaying fission products in a fuel assembly is proportional to empirical emission rates of beta and gamma radiation. The rates per U-235 fission, and as a function of decay time  $t_d$  in days, are

$$\beta(t_d) = 1.50 \cdot 10^{-6} \cdot t_d^{-1.2} \text{ MeV/sf} \quad (22.10)$$

$$\gamma(t_d) = 1.67 \cdot 10^{-6} \cdot t_d^{-1.2} \text{ MeV/sf} \quad (22.11)$$

These energy rates are roughly equal for 0.4 MeV mean energy beta particles and 0.7 MeV mean energy gamma rays. Integrating these equations over the reactor operation time gives the rate of decay heat produced from fission products after the reactor is shut down.

For a fuel assembly irradiated continuously for  $t_i$  days at an average fuel assembly power ( $P$ ), the heat ( $H$ ) load power per assembly and  $t_d$  days after irradiation is

$$H = 6.85 \cdot 10^{-3} \cdot P \cdot (t_d^{-0.2} - (t_i + t_d)^{-0.2}) \text{ W} \quad (22.12)$$

A convenient estimate for the average power  $P$  is

$$P = \frac{(G/t_i)}{1.25 \times 10^{-6}} \text{ W} \quad (22.13)$$

where:

$G$  is the mass of U-235 burned in the fuel assembly in grams

The constant is  $g$  <sup>235</sup>U burned per Watt-day

An analytical expression given by El-Wakil (1978), which correlates with the decay heat curves of ANL report (ANL, 1963), estimates heat loads about one-half the heat loads calculated above. This heat load expression is

$$H = 4.95 \cdot 10^{-3} \cdot P \cdot t_d^{-0.06} \cdot (t_d^{-0.2} - (t_i + t_d)^{-0.2}) \text{ (W)} \quad (22.14)$$

Another analytical expression given by Untermeyer and Weills (1952) has been used to fit experimental decay heat data. This heat load expression is

$$H = 0.1 \cdot P \cdot [(t_d + 10)^{-0.2} - (t_i + t_d + 10)^{-0.2}] - 0.0087 \cdot P [(t_d + 2 \cdot 10^7)^{-0.2} - (t_i + t_d + 2 \cdot 10^7)^{-0.2}] \text{ (W)} \quad (22.15)$$

where the irradiation ( $t_i$ ) and decay ( $t_d$ ) times are in seconds.

### 22.2.3.2 ANS Standard for Decay Heat Calculation

Based on several experimental data American Nuclear Society (ANS) has assembled decay heat standard ANS-5.1-1979 that contained a single curve to represent all uranium-fueled reactors (Schrock, 1979). The latest version of the standard is ANS-5.1-1994 (Current Standard, Revision of ANSI/ANS-5.1-1979; R1985). The standard was developed to fulfill a need for evaluations of fission reactor performance dependent upon knowledge of decay heat power in the fuel elements. The ANS-5.1 standard provides bases for determining the shutdown decay heat power and its uncertainty following shutdown of LWRs.

The ANS-5.1 standard for decay heat generation in NPPs provides a simplified mean of estimating nuclear fuel cooling requirements that can be readily programmed into computer codes used to predict plant performance. The ANS-5.1 standard models the energy release from the fission products of  $^{235}\text{U}$ ,  $^{238}\text{U}$ , and  $^{239}\text{Pu}$  using a summation of exponential terms with empirical constants. Corrections are provided to account for energy release from the decay of  $^{239}\text{U}$  and  $^{239}\text{Np}$  and for the neutron activation of stable fission products. Although the empirical constants are built into the standards, certain data inputs are left to the discretion of the user. These options permit accounting for differences in power history, initial fuel enrichment, and neutron flux level. Decay heat power from other actinides and activation products in structural materials, and fission power from delayed neutron-induced fission, are not included in this standard.

## 22.2.4 Heat Generation in Other Reactor Components

### 22.2.4.1 Heat Generation in the Moderator, Other Core Elements, and Core Structure

Radiation from the core fissioning process interacts with the core structure or moderator. The most significant energy sources are fission g-rays which have high energy, fast neutrons, fission product gamma rays, and capture gamma rays from neutron absorption. The fission gamma rays have very high energy. There are three types of interactions gamma rays can have with matter; they are

- *Photoelectric effect*—( $0.8 \leq E_\gamma \leq 3 \text{ MeV}$ ) Here gamma ray excites an orbital electron which loses its additional kinetic energy by Bremsstrahlung radiation and the gamma ray disappears.
- *Compton scattering*—( $0.1 \leq E_\gamma \leq 20 \text{ MeV}$ ) Here the gamma ray scatters electrons and loses only some of its energy by the scattering event. The electrons lose their kinetic energy by Bremsstrahlung radiations and the gamma ray emerges at a lower energy value.

- *Pair production*—( $1.0 \leq E_\gamma \leq 1000$  MeV) The incoming gamma ray is absorbed and forms an electron/positive pair with the balance of the gamma energy as kinetic energy to the pair.

#### 22.2.4.2 Heat Generation from Radiation within the Core

A portion of the total energy produced is absorbed outside the fuel elements. In the metallic structure (fuel cladding, fuel assembly cans and grids, control element guide thimbles, etc.), the energy is almost entirely due to the absorption of gamma rays emitted by the fuel. Although the gamma flux is attenuated as it passes through the metal, the structural elements in the core are quite thin and the attenuation is small. Gamma ray absorption accounts for about 10.5% of the total heat production, and absorption is roughly proportional to mass. Heat generated within the moderator is from neutron thermalization through elastic scattering.

#### 22.2.4.3 Heat Generation due to Radiation External to the Core

Radiation absorption into the pressure vessel is an important design problem for PWR vendor and is correspondingly less important for BWR, LMFBR, and HTGR reactor types. Design pressure of a PWR is high and the reactor vessel is stainless steel-coated carbon steel which loses its ductility in a radiation field. The concern is accident conditions, large steam line break, or small loss of coolant accident (LOCA), where the hot brittle pressure vessel is repressurized by cold water from the charging pumps. While neutrons can lead to structural damage, it is the gamma rays which contribute to the heating of the shield. The number of particles absorbed at a particular location is proportional to the number of particles at that location.

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## 22.3 Heat Transport in the Fuel Element

### 22.3.1 Introduction

In nature there are three modes of heat transfer: conduction, convection, and radiation.

Conduction requires a media wherein adjacent molecules in a solid or fluid collide passing the molecular thermal (translational, rotational, and vibrational) energy from molecular layer to molecular layer. It is as if the thermal energy is *diffusing* through the material in an existing temperature gradient. The convection heat transfer is solely due to the bulk motion of the fluid. The fluid gains thermal energy from a hot surface and is carried to cold region with fluid flow. The flow in can be induced either by forced flow pumping or by buoyancy force due to temperature difference in the fluid. The heat transfer with forced flow is referred as forced convection, and if the heat transfer is primarily due to gravitational force, it is referred as natural convection or free convection. The convection heat transfer can occur in single phase as well as in two-phase flow such as with phase change heat transfer in boiling and condensation. The flow can occur external to the surface or wall or internal to a channel or piping. The radiation heat transfer does not require a material medium. Radiation is explained through wave mechanics as electromagnetic waves and in quantum mechanics as particles with speed and a wave frequency.



### 22.3.2 General Heat Conduction Equation

The goal of the conduction heat transfer is to determine the temperature field in a medium (such as fuel rod) and the rate of heat transfer to and from the medium. Typically, the media is subjected to nonuniform temperature distribution which is a result of either a heat source within the medium or heat flux from the boundary of the medium. In this section various forms of the heat conduction equation that govern the temperature field in a medium and its associated boundary conditions are given. Some examples of the heat conduction in nuclear fuel rod and other components are presented.

The general heat conduction can be derived by using the differential analysis for heat diffusion in the Cartesian coordinate system as shown in Figure 22.2. Writing heat balance, the rate of accumulation of thermal energy in an element (of size  $dx dy dz$ ) is equal to the rate of input to the element by conduction, minus the rate of output of thermal energy by conduction, plus the net rate of generation of thermal energy from nuclear reaction, electrical work, friction, and in some cases chemical reaction (if handled in a certain way):

$$\rho c_p \frac{dT}{dt} dx dy dz = q''_x|_{x=0} dy dz - q''_x|_{x=dx} dy dz + q''_y|_{y=0} dx dz - q''_y|_{y=dy} dx dz + q''_z|_{z=0} dx dy - q''_z|_{z=dz} dx dy + \dot{q} dx dy dz \tag{22.16}$$

where:

$T$  is the temperature

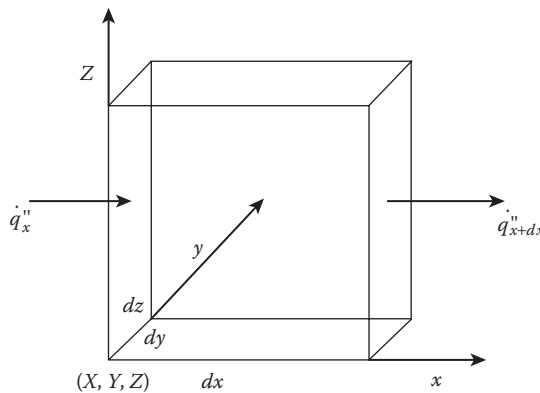
$t$  is the time

$q''$  is the heat flux

The heat generation term is expressed as the product of the generation per unit volume,  $\dot{q}$

$\rho$  is the density and  $c_p$  is the specific heat of the medium and both are general functions of temperature and position ( $\vec{r}$ ). Dividing the equation by  $dx dy dz$  gives

$$\rho c_p \frac{dT}{dt} = \frac{q''_x|_{x=0} - q''_x|_{x=dx}}{dx} + \frac{q''_y|_{y=0} - q''_y|_{y=dy}}{dy} + \frac{q''_z|_{z=0} - q''_z|_{z=dz}}{dz} + \dot{q} \tag{22.17}$$



**FIGURE 22.2**  
Heat diffusion in a control volume.

The heat diffusion equation becomes

$$\rho c_p \frac{dT}{dt} = -\frac{dq_x''}{dx} - \frac{dq_y''}{dy} - \frac{dq_z''}{dz} + \dot{q} \quad (22.18)$$

Using Fourier's law, the heat fluxes ( $q_x''$ ,  $q_y''$ ,  $q_z''$ ) are related to the temperature gradients in the respective directions as

$$q_x'' = -k \frac{\partial T}{\partial x}, \quad q_y'' = -k \frac{\partial T}{\partial y}, \quad q_z'' = -k \frac{\partial T}{\partial z} \quad (22.19)$$

Substituting Fourier's law into the thermal heat balance gives

$$\rho c_p \frac{\partial T}{\partial t} = \frac{\partial}{\partial x} \left( k \frac{\partial T}{\partial x} \right) + \frac{\partial}{\partial y} \left( k \frac{\partial T}{\partial y} \right) + \frac{\partial}{\partial z} \left( k \frac{\partial T}{\partial z} \right) + \dot{q} \quad (22.20)$$

In general, the heat conduction equation is written as

$$\frac{1}{\alpha} \frac{\partial T}{\partial t} = \nabla^2 T + \frac{\dot{q}}{k} \quad (22.21)$$

where  $\alpha = k/\rho c_p$  which is called thermal diffusivity. It indicates how fast the heat diffuses in the medium. The Laplacian operator  $\nabla^2$  for Cartesian, spherical, and cylindrical coordinates as shown in Figure 22.3 are given in Table 22.3.

### 22.3.2.1 The Initial and Boundary Conditions

In each equation the dependent variable,  $T$ , is a function of four independent variables,  $(x, y, z, t)$ ;  $(r, \phi, z, t)$ ;  $(r, \theta, \phi, t)$  and is a second-order, partial differential equation. In order to solve the differential conduction equation for specific problem, two boundary conditions and one initial condition are required.

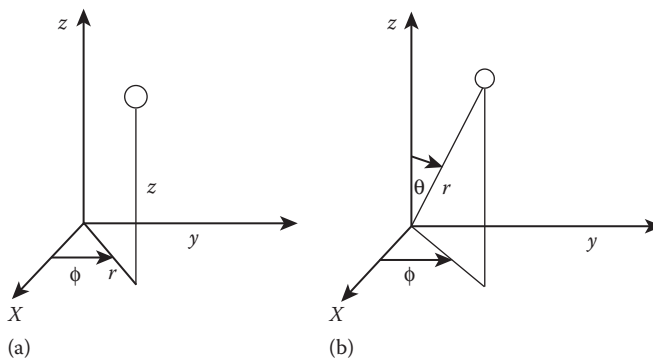


FIGURE 22.3

(a) Cylindrical coordinates  $(r, \phi, z)$  and (b) spherical coordinates  $(r, \theta, \phi)$ .

**TABLE 22.3**

Laplacian for Cartesian, Spherical, and Cylindrical Coordinates

$\nabla^2 T = \frac{\partial^2 T}{\partial x^2} + \frac{\partial^2 T}{\partial y^2} + \frac{\partial^2 T}{\partial z^2}$	Cartesian
$\nabla^2 T = \frac{1}{r} \frac{\partial}{\partial r} \left( r \cdot \frac{\partial T}{\partial r} \right) + \frac{1}{r^2} \cdot \frac{\partial^2 T}{\partial \phi^2} + \frac{\partial^2 T}{\partial z^2}$	Cylindrical
$\nabla^2 T = \frac{1}{r^2} \frac{\partial}{\partial r} \left( r^2 \cdot \frac{\partial T}{\partial r} \right) + \frac{1}{r^2 \cdot \sin^2 \theta} \cdot \frac{\partial^2 T}{\partial \phi^2} + \frac{1}{r^2 \sin \theta} \cdot \frac{\partial}{\partial \theta} \left( \sin \theta \cdot \frac{\partial T}{\partial \theta} \right)$	Spherical

**22.3.2.2 Boundary Conditions**

Consider a plane wall normal to the  $x$ -axis with a thickness  $L$  as shown in Figure 22.4.

The left-hand surface is located at  $x = 0$ . The temperature at any point in the wall may be represented by  $T(x, y, z, t)$ . The common boundary conditions are as follows:

1. *Prescribed surface temperature.* At the surface  $x = 0$ , the temperature is a constant and equals to  $T_0$ . The boundary condition is  $T(0, y, z, t) = T_0 = \text{constant}$ . The prescribed surface temperature is referred to as Dirichlet boundary condition.
2. *Prescribed surface heat flux.* At the surface  $x = 0$ , the heat transfer rate per unit area is equal to  $q''$  which is a constant,

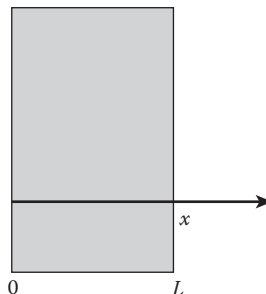
$$\left( -k \frac{\partial T}{\partial x} \right)_{x=0} = q'' \quad (22.22)$$

The prescribed heat flux boundary condition (BC) is referred to as Neumann boundary condition.

If there is no heat flow normal to the surface  $x = 0$ , the temperature gradient at the surface must be equal to zero.

$$\left( \frac{\partial T}{\partial x} \right)_{x=0} = 0 \quad (22.23)$$

This is adiabatic or insulated surface boundary condition



**FIGURE 22.4**  
Plane wall geometry.

3. *Convection boundary condition.* The heat conducted outward from the interior point of the wall to the surface  $x = L$  must be equal to that dissipated from the surface by convection

$$\left(-k \frac{\partial T}{\partial x}\right)_{x=L} = h(T_L - T_\infty) \quad (22.24)$$

where:

$h$  is convection heat transfer coefficient

$T_L$  is the surface temperature at  $x = L$

$T_\infty$  is the ambient temperature of the fluid

4. *Symmetric boundary condition.* If the temperature profile is symmetric about a certain plane, the temperature distribution at this plane must be a maximum or a minimum point and its gradient must be equal to zero.

$$\left(\frac{\partial T}{\partial x}\right)_{\text{symmetric}} = 0. \quad (22.25)$$

There is not heat flow across the symmetric plane.

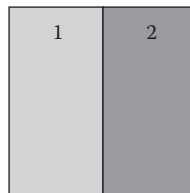
5. *Combined heat transfer boundary condition.* In combined mode heat transfer, the heat conducted from the interior point of the medium to the surface must be equal to that dissipated by convection and radiation

$$\left(-k \frac{\partial T}{\partial x}\right)_{x=L} = h(T_L - T_\infty) + h_r(T_L - T_{\text{surr}}) \quad (22.26)$$

6. *Interface boundary condition.* If two solid plates 1 and 2 are brought together in perfect contact (Figure 22.5), the heat flux conducted from plate 1 to the interface must be equal to that conducted to plate 2 from the interface and temperatures at the interface of the two plates must be equal

$$-\left(k \frac{dT}{dx}\right)_{1i} = -\left(k \frac{dT}{dx}\right)_{2i} \quad (22.27)$$

$$T_{1i} = T_{2i}$$



**FIGURE 22.5**  
Interface boundary condition.

### 22.3.2.3 Initial Condition

Initial condition is required to find a solution for problems with transient conduction. Initial condition is used to specify the temperature field at initial time. It may specify the initial temperature field at a particular point, surface, or region. If the time zero distribution is given as  $T_0(x)$ , the solution must satisfy  $T(x, t = 0) = T_0(x)$ .

## 22.3.3 Examples of One-Dimensional Heat Conduction Equation

### 22.3.3.1 Large Slab with Internal Heat Generation

Consider one-dimensional heat conduction in a large slab of width  $L$  with constant temperatures  $T_1$  and  $T_2$  at surfaces  $x = 0$  and  $x = L$  and internal heat generation,  $\dot{q}$ , as shown in Figure 22.6. Assume constant thermal conductivity and heat conduction process is steady.

The conduction equation is

$$\frac{d^2T}{dx^2} + \frac{\dot{q}}{k} = 0. \quad (22.28)$$

The boundary conditions are

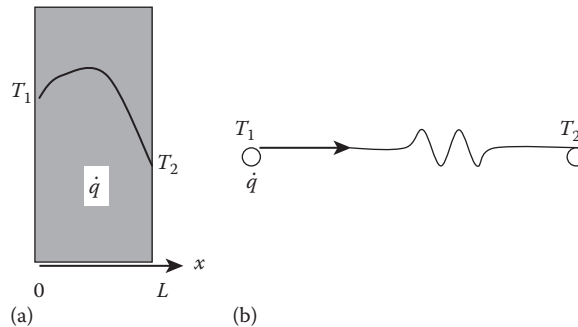
$$T(0) = T_1 \quad \text{and} \quad T(L) = T_2. \quad (22.29)$$

Integrating equation twice and applying the boundary conditions we have the solution,

$$T = \frac{\dot{q}}{2k} x(L-x) + (T_2 - T_1) \frac{x}{L} + T_1. \quad (22.30)$$

By taking derivative of  $T$  with respect to  $x$  and setting equal to zero, the location of the maximum temperature point is

$$x = \frac{L}{2} + \frac{k}{\dot{q}} \frac{T_2 - T_1}{L} \quad (22.31)$$



**FIGURE 22.6**

(a) Plane slab and (b) conduction resistance equivalent.

### 22.3.3.2 Heat Conduction in a Solid Circular Cylinder with Heat Generation—A Fuel Rod

Consider one-dimensional steady-state heat conduction in a cylinder with internal heat generation and convection boundary condition.

The heat conduction equation in cylindrical coordinates is simplified to

$$\frac{1}{r} \frac{\partial}{\partial r} \left( kr \cdot \frac{\partial T}{\partial r} \right) + \dot{q} = 0 \quad (22.32)$$

The boundary conditions are as discussed below.

At center of the cylinder  $r = 0$ ,  $dT/dr = 0$  (maximum temperature condition) and at surface  $r = r_o$ , the convection boundary condition  $\dot{q}(\pi r_o^2 L) = h(2\pi r_o L)(T_o - T_a)$  gives

$$T = T_o = T_a + \frac{\dot{q}r_o}{2h}. \quad (22.33)$$

In general, the thermal conductivity of the substance is a function of temperature.

Integrating the heat conduction equation twice,

$$\int_{T_s}^T k(T') dT' = - \int_{r_o}^r \frac{1}{r} \int_0^r r' \dot{q}(r') dr' dr \quad (22.34)$$

Defining linear heat rate  $q'$ , which is the rate of heat generation per unit length of fuel rod,  $q' = \pi r_o^2 \dot{q}$ , we have the heat conduction equation

$$\int_{T_s}^T k(T') dT' = \frac{q'}{4\pi} \left( 1 - \frac{r^2}{r_o^2} \right). \quad (22.35)$$

By assuming that the thermal conductivity is independent of temperature and for uniform heat generation rate  $\dot{q}$ , the fuel rod temperature profile is

$$T - T_o = \frac{\dot{q}(r_o^2 - r^2)}{4k} = \frac{q'(r_o^2 - r^2)}{4\pi r_o^2 k}. \quad (22.36)$$

The maximum temperature across the fuel rod occurs at center,  $r = 0$

$$T_{\max} - T_o = \frac{\dot{q}r_o^2}{4k} = \frac{q'}{4\pi k}. \quad (22.37)$$

In terms of the temperature dependent thermal conductivity, the fuel rod temperature is

$$\int_{T_s}^{T_{\max}} k(T') dT' = \frac{q'}{4\pi}. \quad (22.38)$$

### 22.3.4 Thermal Properties

In Table 22.4, the thermal properties of fuel and cladding materials are given.

There is considerable body of data and correlations in the open literature covering thermal properties of  $\text{UO}_2$ . Recent reviews (Fink, 2000; Carbajo et al., 2001) present the best correlations for  $\text{UO}_2$ . The work of Fink (2000) is also available at the International Nuclear Safety Center (INSC) database of Argonne National Laboratory (<http://www.insc.anl.gov/matprop>).

The expression for the thermal conductivity for 95% TD  $\text{UO}_2$  as suggested by Fink (2000) is

$$k(T) = \frac{100}{7.5408 + 17.692t + 3.6142t^2} + \frac{6400}{t^{5/2}} e^{-16.35/t} \text{ (W/m K)}, \quad (22.39)$$

where  $t = T/1000$ .

Table 22.5 lists the formulations for temperature-dependent thermal conductivity of  $\text{UO}_2$  at 95% theoretical density used by reactor vendors.

The measured thermal conductivities of the irradiated  $\text{UO}_2$  in a commercial reactor (maximum burnup: 60 MWd/t) up to 2000 K have been reported by Amaya et al. (2002).

The effects of dissolved and precipitated fission products, the effects of the porosity, and the effects of the irradiation are taken into account through factors

$$K \text{ (irradiated)} = k \text{ (fresh)} \times FD \times FP \times FM \times FR \quad (22.40)$$

$$\text{Dissolved fission products factor: } FD = \omega[\arctan(1/\omega)] \quad (22.41)$$

**TABLE 22.4**

Thermal Properties of Fuel and Cladding Materials

Property	U	$\text{UO}_2$	UC	UN	Zircaloy 2	SS 316
Density <sup>a</sup> (kg/m <sup>3</sup> )	$19.04 \times 10^3$	$10.97 \times 10^3$	$13.63 \times 10^3$	$14.32 \times 10^3$	$6.5 \times 10^3$	$7.8 \times 10^3$
Metal density (kg/m <sup>3</sup> )	$19.04 \times 10^3$	$9.67 \times 10^3$	$12.97 \times 10^3$	$13.06 \times 10^3$	$6.5 \times 10^3$	$7.8 \times 10^3$
Melting point (°C)	133	2800	2390	2800	1850	1400
Thermal conductivity <sup>b</sup> (W/m °C)	32	3.6	24	21	13(400°C)	23(400°C)
Specific heat (J/kg °C)	116 (100°C)	247 (100°C)	146 (100°C)	—	330 (400°C)	580 (400°C)
Linear thermal expansion coefficient (/°C)	—	$10.1 \times 10^{-6}$	$11.1 \times 10^{-6}$	$9.4 \times 10^{-6}$	$5.9 \times 10^{-6}$	$18 \times 10^{-6}$
Crystal structure	Below 655°C: $\alpha$ orthorhombic, above 770°C: $\gamma$ , body centered cubic	Face centered cubic	Face centered cubic	Face centered cubic		
Tensile strength (MPa)	344–1380	110	62	—		

Source: Todreas, N.E. and Kazimi, M.S., *Nuclear System. I: Thermal Hydraulic Fundamentals*, Hemisphere, New York, 1990.

<sup>a</sup> Theoretical density for fuels.

<sup>b</sup> Average for fuel materials for range of temperature (200°C–1000°C).

**TABLE 22.5**

UO<sub>2</sub> Thermal Conductivity at 95% Theoretical Density

(1) Used by combustion engineering—Derived from Lyon's  $\int kdT$

$$k \text{ (W/cm } ^\circ\text{C)} = \frac{38.24}{402.4 + T} + 6.1256 \times 10^{-13} (T + 271)^3$$

$T$  in  $^\circ\text{C}$

(2) Used by Westinghouse

$$k \text{ (W/cm } ^\circ\text{C)} = \frac{1}{11.8 + 0.0238T} + 8.775 \times 10^{-13} T^3$$

$T$  in  $^\circ\text{C}$

(3) Used by Babcock and Wilcox—Polynomial representation of Lyon's  $\int kdT$

$$\int_{32}^T kdT \text{ (Btu/h ft)} = -170.9124 + 5.597256T - 3.368695 \times 10^{-3} T^2 \\ + 1.96278 \times 10^{-6} T^3 - 8.391225 \times 10^{-10} T^4 + 2.404192 \times 10^{-13} T^5 \\ - 4.275284 \times 10^{-17} T^6 + 4.249043 \times 10^{-21} T^7 - 1797017 \times 10^{-25} T^8$$

$T$  in  $^\circ\text{F}$

Source: Todreas, N.E. and Kazimi, M.S., *Nuclear System. I: Thermal Hydraulic Fundamentals*, Hemisphere, New York, 1990.

where:

$$\omega = 1.09/B^{3.265} + 0.0643(T/B)^{0.5} \quad (22.42)$$

$T$  in  $^\circ\text{K}$

$B$  is burnup in %

$$\text{Precipitated fission products factor: } FD = 1 + \frac{0.019B}{(3 - 0.019B)[1 + \exp\{-(T - 1200)/100\}]} \quad (22.43)$$

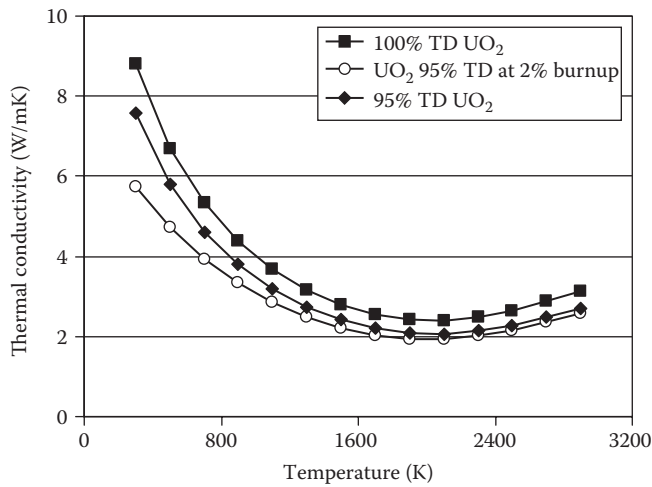
$$\text{Porosity factor: } FM = \frac{1 - \varepsilon}{1 + 2\varepsilon} \quad (22.44)$$

where  $\varepsilon$  is porosity in the fuel.

$$\text{Radiation effect factor: } FM = 1 - \frac{0.2}{1 + \exp(T - 900)/80} \quad (22.45)$$

In Figure 22.7 the thermal conductivity of 95% TD fresh fuel, 95% TD at 2% burnup, and 100% TD fresh fuel are shown.





**FIGURE 22.7**  
Thermal conductivity of UO<sub>2</sub>, 95% TD fresh fuel, 95% TD at 2% burnup, and 100% TD fresh fuel.

## 22.4 Heat Transport to the Coolant-Single Phase

### 22.4.1 Introduction

Heat removal from the reactor core is one of the important issues in reactor design and operation. The main objective of heat transfer analysis for single-phase flow in reactor is to determine the temperature field in a coolant channel such that the reactor operating temperatures are within the specified limits and rate of heat transfer to and from surface or object. Since reactor power densities are typically orders of magnitude higher than other conventional heat sources, the heat removal rate from any given reactor core coolant channel is quite large. The heat transfer to the coolant in single phase thus requires coolant with large heat capacity. The coolants should ideally have the following properties:

- Low absorption cross section
- Abundant and inexpensive
- Low melting point
- High boiling point
- Noncorrosive properties
- High moderating ratio (for thermal reactors)
- Radiation and thermal stability
- Low induced radioactivity
- No reaction with turbine working fluid
- High heat transport and transfer coefficient
- Low pumping power

The heat transfer from coolant to any solid surface is given through Newton's cooling law in terms of heat flux

$$q'' = h(T_s - T_b), \quad (22.46)$$

where:

$T_s$  is the surface temperature

$T_b$  is the bulk temperature of the coolant

$h$  (W/m<sup>2</sup> °K) is the heat transfer coefficient

The heat transfer coefficient is dependent on coolant properties; flow parameters such as velocity field, temperature field in solid surface, and coolant; solid surface condition; and geometry.

The heat transfer coefficient  $h$  is defined through nondimensional Nusselt number as

$$\text{Nu} = \frac{hD_h}{k} = f\left(\text{Re}, \text{Pr}, \text{Gr}, \frac{\mu_w}{\mu_b}\right) \quad (22.47)$$

where:

$D_h$  is the hydraulics diameter

$\mu$  is the viscosity

Subscripts  $w$  and  $b$  refer to wall and bulk, respectively

The nondimensional numbers Re, Pr, and Gr are defined as

$$\text{Re} = \frac{\rho VD}{\mu}, \text{ Reynolds number}$$

$$\text{Pr} = \frac{\mu c_p}{k}, \text{ Prandtl number}$$

$$\text{Gr} = \frac{g\beta\rho^2(T_w - T_b)^3}{\mu^2}, \text{ Grashof number}$$

where:

$\rho$  is the density

$\beta$  is the temperature coefficient of expansion

Typical values of the heat transfer coefficient for various fluids and processes are shown in Table 22.6.

In most of the commercial reactors today, water is used as the coolant. The water also serves as a moderator in these reactors. Coolants that have been used, in test or commercial applications, include

- Light (H<sub>2</sub>O) or heavy water (D<sub>2</sub>O)
- Liquid metal (sodium, potassium, or NaK—an alloy combination of sodium and potassium)
- Liquid organics (e.g., ethanol, propane, pentane, benzene, heptane)
- Air, helium, or carbon dioxide gases

**TABLE 22.6**

Typical Values and Ranges of Heat Transfer for Various Processes and Materials

Mode and Material	Heat Transfer Coefficient ( <i>h</i> )	
	W/m <sup>2</sup> °C	Btu/h ft <sup>2</sup> °F
<i>Natural Convection</i>		
Air-vertical plate 0.3 m (1 ft) high	4.5	0.75
Air-horizontal cylinder, 5 cm dia	6.5	1.14
Low pressure gas	6–28	1–5
Water-horizontal cylinder, 2 cm dia	890	157
Liquids	60–1000	10–175
<i>Forced Convection</i>		
Airflow at 2 m/s over 0.2 m square plate	12	2.1
Airflow at 35 m/s over 0.75 m square plate	75	13.2
Air at 2 atm flowing in 2.5cm dia tube at 10 m/s	65	11.4
Low pressure gas flow in pipe	6–600	1–100
Water at 0.5 kg/s flowing in 2.5cm dia tube	3,500	616
Water flow in pipes	250–12,000	44–2,000
Sodium flow in pipes	2,500–25,000	440–4,400
<i>Boiling Water</i>		
In a pool or container	2,500–35,000	440–6,200
Flowing in a tube	5,000–100,000	880–17,600
<i>Condensation of Water, 1 atm</i>		
Vertical surfaces	4,000–11,300	700–2,000
Outside horizontal tubes	9,500–25,000	1,700–4,400

Sources: Holman, J.P., *Heat Transfer*, 7th edn., McGraw Hill, 1990; Todreas, N.E. and Kazimi, M.S., *Nuclear System. I: Thermal Hydraulic Fundamentals*, Hemisphere, New York, 1990.

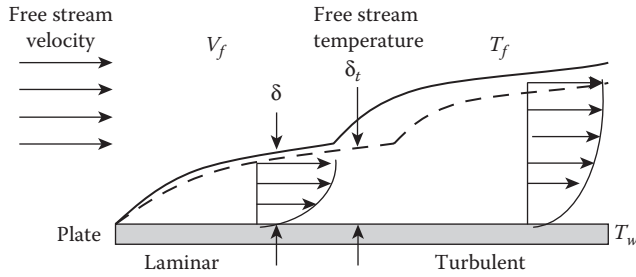
## 22.4.2 Laminar Heat Transfer

### 22.4.2.1 External Flow

For external flow such as flow over a flat stationary plate whose surface temperature is different from the bulk fluid temperature, both hydrodynamic and thermal boundary layers develop along the direction of the flow. Inside the hydrodynamic boundary layer viscous forces are dominant resulting in velocity profile. Similarly as thermal boundary layer develops along the flow direction and heat is being transferred to or from the surface results in a temperature profile. In Figure 22.8 the hydrodynamic and thermal boundary layers are shown for flow over a heated flat plate. Both velocity and temperature inside the boundary layer reach 99% of the free stream velocity ( $V_f$ ) and temperature ( $T_f$ ), respectively, at the edge of the boundary layer.

For simple case of laminar flow over flat plate, it is possible to obtain velocity and temperature profiles with simplifying assumptions.

Assuming isothermal plate and constant fluid properties and negligible conduction heat transfer in the  $x$ -direction, the integral  $x$ -momentum equation for the hydrodynamic boundary layer is



**FIGURE 22.8** Hydrodynamic and thermal boundary layers on a flat plate with wall temperature higher than free stream temperature.

$$\delta \frac{dp}{dx} - \tau_w = \frac{\partial}{\partial x} \int_0^{\delta} u \rho u \, dy - V_f \frac{\partial}{\partial x} \int_0^{\delta} \rho u \, dy. \tag{22.48}$$

Here  $\tau_w$  is the shear stress at the wall. If the pressure is constant throughout the flow then  $dp/dx = 0$ . By assuming  $V_f$  constant, the momentum equation becomes

$$\tau_w = \rho V_f^2 \frac{\partial}{\partial x} \int_0^{\delta} \frac{u}{V_f} \left( 1 - \frac{u}{V_f} \right) dy, \tag{22.49}$$

where now shear stress at the wall is

$$\tau_w = \mu \left. \frac{\partial u}{\partial y} \right)_{y=0} \tag{22.50}$$

The boundary conditions are

$$\text{At the plate surface } y = 0, \quad u = 0, \quad \text{and} \quad \frac{\partial^2 u}{\partial y^2} = 0 \tag{22.51}$$

$$\text{At the edge of boundary layer } y = \delta, \quad u = V_f, \quad \text{and} \quad \frac{\partial u}{\partial y} = 0 \tag{22.52}$$

By assuming a third order polynomial for the velocity profile

$$u = C_1 + C_2 y + C_3 y^2 + C_4 y^3 \tag{22.53}$$

and using the BCs the solution for the velocity profile is obtained as

$$\frac{u}{V_f} = \frac{3}{2} \left( \frac{y}{\delta} \right) - \frac{1}{2} \left( \frac{y}{\delta} \right)^3 \tag{22.54}$$

Using the velocity profile expression and integrating momentum equation, the boundary layer thickness  $\delta$  is given as

$$\frac{\delta}{x} = \frac{4.64}{\sqrt{\text{Re}_x}} \quad (22.55)$$

where  $\text{Re}_x = \rho V_f x / \mu$ . The exact solution of the momentum equation yield the constant as 5 instead on 5.48 in the above equation.

The boundary layer integral energy equation is

$$q_w'' = \frac{d}{dx} \int_0^{\delta_t} (T_f - T) u dy + \frac{\mu}{\rho c_p} \int_0^{\delta} \left( \frac{\partial u}{\partial y} \right)^2 dy \quad (22.56)$$

where the wall heat flux is given as

$$q_w'' = \alpha \left. \frac{\partial T}{\partial y} \right|_{y=0} \quad (22.57)$$

$$\text{At the plate surface } y=0, \quad T = T_w, \quad \text{and} \quad \frac{\partial^2 T}{\partial y^2} = 0 \quad (\text{no viscous heating}) \quad (22.58)$$

$$\text{At the edge of boundary layer } y = \delta_t, \quad T = T_f, \quad \text{and} \quad \frac{\partial T}{\partial y} = 0. \quad (22.59)$$

Similar to the velocity profile if a cubic polynomial is assumed for the temperature profile, we have

$$\frac{\theta}{\theta_w} = \frac{T - T_w}{T_f - T_w} = \frac{3}{2} \left( \frac{y}{\delta_t} \right) - \frac{1}{2} \left( \frac{y}{\delta_t} \right)^3 \quad (22.60)$$

where  $\theta = T - T_w$ .

Again using the velocity and temperature profiles and integrating energy equation, the thermal boundary layer thickness is given as

$$\frac{\delta_t}{\delta} = \frac{1}{1.026} \text{Pr}^{-1/3}, \quad (22.61)$$

where the Prandtl number  $\text{Pr} = c_p \mu / k$ .

The heat transfer coefficient,  $h$ , is

$$h = \frac{q_w''}{(T_w - T_f)} = \frac{-k(\partial T / \partial y)_w}{(T_w - T_f)} = \frac{3k}{2\delta_t} \quad (22.62)$$

Thus, the local heat transfer coefficient for the plate is

$$h_x = 0.332kPr^{1/3} \left( \frac{\rho V_f}{\mu x} \right)^{1/2} \tag{22.63}$$

In terms of the Nusselt number  $Nu_x = h_x x/k$ , we have

$$Nu_x = 0.332Pr^{1/3} Re_x^{1/2}. \tag{22.64}$$

**22.4.2.2 Internal Flow**

For internal flows both the hydrodynamic boundary layer and the thermal boundary layer developing on the wall merge (at the center for symmetric channels) downstream of the entrance and provide thereafter a fully developed boundary layers.

*22.4.2.2.1 Pipe Flow*

In contrast to external flows, the momentum and thermal boundary layer developing on the wall in internal flows such as in pipe merge (at the center of symmetric channels) and provide thereafter the developed region where axial variation do not occur further downstream.

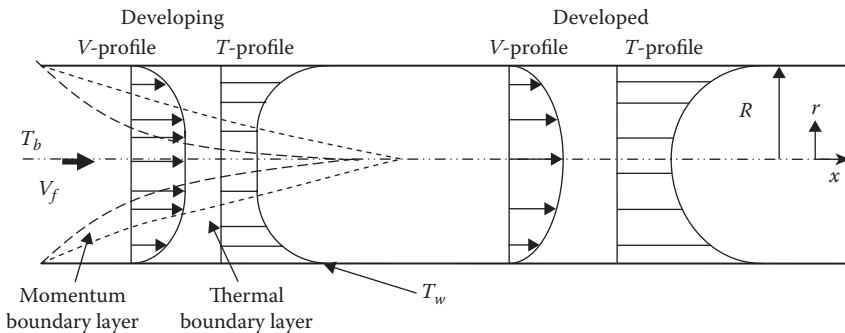
For pipe flow, velocity and temperature profiles in a developing and fully developed region are illustrated in Figure 22.9. The entrance length where the merging of the momentum boundary layer access is given by

$$\frac{L_c}{D} = 0.0288 \left( \frac{U_m D \ell}{\mu} \right) = 0.0288 Re_m, \tag{22.65}$$

where:

$U_m$  is the average neon fluid velocity in the pipe

Re is the Reynolds number based upon pipe diameter  $D$



**FIGURE 22.9**  
Momentum and thermal boundary layers internal flow in pipe in developing and developed regions.

The mean fluid velocity for the fully developed flow is given as

$$V_m = \frac{\int_0^R \rho u 2\pi r \cdot dr}{\pi R^2}. \quad (22.66)$$

Similarly the fluid mean temperature is defined as

$$T_m = \frac{\int_0^R \rho u T 2\pi r \cdot dr}{\int_0^R \rho u 2\pi r \cdot dr} = \frac{\int_0^R \rho u T 2\pi r \cdot dr}{V_m \pi R^2} \quad (22.67)$$

The velocity profile for fully developed pipe flow is a parabolic profile and is given as

$$u = 2V_m \left( 1 - \frac{r^2}{R^2} \right) \quad (22.68)$$

This is obtained from momentum equation and boundary conditions given as

$$\frac{\mu}{r} \frac{d}{dr} \left( r \frac{du}{dr} \right) = \frac{dp}{dx} \quad (22.69)$$

$$r = 0 \quad \frac{du}{dr} = 0 \text{ (maximum } u \text{ at symmetry)} \quad (22.70)$$

$$r = R \quad u = 0 \text{ (at the wall)} \quad (22.71)$$

The mean velocity is now given as

$$V_m = \frac{R^2}{2\mu} \left( -\frac{dp}{dx} \right) \quad (22.72)$$

The shear stress at the wall is given as

$$\tau_w = \mu \left. \frac{du}{dr} \right|_{r=R} = -4\mu \frac{V_m}{R} = \frac{R}{2} \left( \frac{dp}{dx} \right) \quad (22.73)$$

Defining the pressure drop in the form proposed by Darcy

$$-\frac{dp}{dx} = \frac{f}{D} \frac{\rho v^2}{2} \quad (22.74)$$

where  $f$  is the friction factors, we have

$$f = -\frac{2D}{\rho V_m^2} \left( \frac{dp}{dx} \right) = \frac{64}{(\rho D v_m / \mu)} = \frac{64}{\text{Re}} \quad (22.75)$$

The friction pressure drop inside a pipe of length  $L$  is given as

$$\Delta p_{\text{friction}} = \frac{fL}{D} \frac{\rho V_m^2}{2} \quad (22.76)$$

The expression for the mean temperature of the fluid can be given for two important boundary conditions, namely constant wall temperature and constant heat flux conditions.

For constant heat flux, the mean fluid temperature is

$$T_m - T_w = \frac{11}{24} \frac{R}{K} q_w'' \quad (22.77)$$

The heat transfer coefficient is given as

$$h = \frac{q_w''}{T_m - T_w} = \frac{24}{11} \frac{k}{R} \quad (22.78)$$

The Nusselt number for constant heat flux boundary conditions for pipe is

$$\text{Nu} = \frac{h(2R)}{k} = \frac{48}{11} = 4.364 \quad (22.79)$$

The laminar heat transfer coefficient in terms of Nusselt number is independent of flow velocity or Prandtl numbers for constant heat flux and constant temperature boundary conditions.

For geometries other than round pipe, a Nusselt number is explained using equivalent heated diameter, which is defined as

$$D_H = \frac{4 \times \text{Flow area}}{\text{Heated perimeter}} \quad (22.80)$$

In Table 22.7 hydraulic diameter and Nusselt number for Laminar fully developed flow in channel of various geometry are listed.

## 22.4.3 Turbulent Heat Transfer


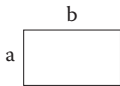

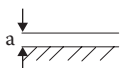

### 22.4.3.1 External Flow

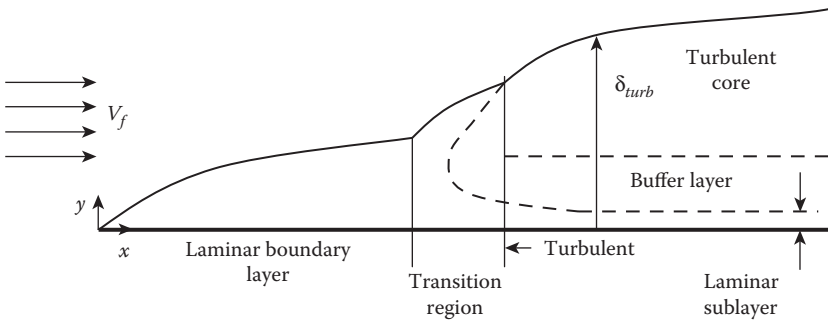
In an external flow such as flow along a flat plate, at same distance along the plate, the flow becomes unstable and the laminar boundary layer becomes turbulent. Figure 22.10 illustrates regions of various types of boundary layers in external flow. At low velocities, the flow within the boundary layer is laminar. However, immediately downstream from laminar region disturbance within flow appears leading to the intermittency between laminar and turbulent called transition region. Downstream from transition region, the flow is completely turbulent. Within turbulent flow eddies are formed that have random velocity. However, adjacent to the surface in the turbulent boundary layer is a thin sublayer in which the flow is laminar.



**TABLE 22.7**

Hydraulic Diameter and Nusselt Number for Fully Developed Flow in Various Geometry Channels

Channel	$b/a$	$D_H$	NU Constant Heat Flux	NU Constant Wall Temperature
	—	$D$	4.364	366
	1	$a$	3.63	
	2	$4/3a$	4.11	2.98
	3	$3/2a$	4.77	3.39
	4	$8/5a$	5.35	4.44
	8	$16/9a$	6.60	5.95
		$2a$	8.235	7.54
		$2a$	5.385	4.86
	—	$\frac{a}{\sqrt{3}}$	3	2.35



**FIGURE 22.10**  
Boundary-layer transition.

There is a critical Reynolds number which determines the change from laminar to turbulent boundary layers. The range of critical Reynolds number is typically large and depends upon the plate surface conditions, shape of leading edge, and nature of the upstream flow. The range of critical Reynolds number for transition to turbulent flow is

$$8 \times 10^4 \leq Re_x \leq 5 \times 10^6 \tag{22.81}$$

For design purposes the transition from laminar to turbulent boundary is assumed at  $Re_x = 3 \times 10^5$ .

For a free stream without pressure gradient, the turbulent boundary layer thickness is given as

$$\delta_t = \frac{0.374x}{Re_x^{0.2}} \tag{22.82}$$

The laminar sublayer thickness is given as

$$\delta_b = 29 \frac{x}{\text{Re}^{0.9}} \quad (22.83)$$

Velocity profiles in turbulent boundary layer are determined experimentally since solutions to the flow equations are not available. Based on experimental data, universal velocity profile is proposed in which the boundary layer is divided into four regions—laminar sublayer, buffer zone, turbulent core, and turbulent wake. The turbulent velocity profile is given in terms of dimensionless velocity defined as

$$u^+ = \frac{u}{\sqrt{\tau_w/\rho}} \quad (22.84)$$

and dimensional distance from wall given as

$$y^+ = \frac{y\sqrt{\tau_w/\rho}}{\nu} \quad (22.85)$$

The turbulent layer velocity profile without a pressure gradient is given as

$$\text{Laminar sublayer} \quad 0 < y^+ < 5 \quad U^+ = y^+ \quad (22.86)$$

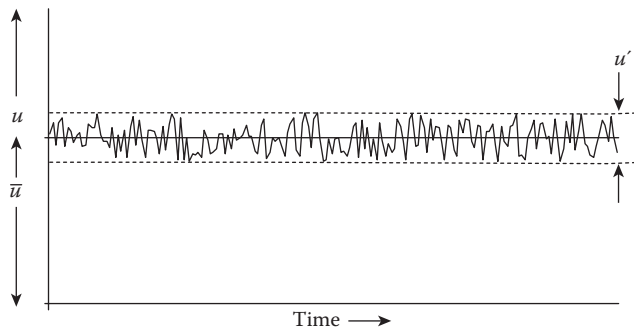
$$\text{Buffer zone} \quad 5 < y^+ < 18.2 \quad U^+ = 3.05 + 5.0 \ln y^+ \quad (22.87)$$

$$\text{Turbulent core} \quad 18.2 < y^+ < 360 \quad U^+ = 4.4 + 2.43 \ln y^+ \quad (22.88)$$

$$\text{Turbulent wake} \quad Y^+ > 360 \quad U^+ = -4.4 + 3.96 y^+ \quad (22.89)$$

As shown in Figure 22.11 in turbulent flow, the instantaneous velocity is composed of two components: the mean velocity  $\bar{u}$  and the fluctuation  $u'$  from the mean. The instantaneous velocity

$$u = \bar{u} + u' \quad (22.90)$$



**FIGURE 22.11**  
Instantaneous turbulent velocity.

The fluctuations are isotropic. The fluctuations give rise to shear stress in addition to the mean shear stress called turbulent shear stress. Thus, in turbulent flow the effective shear stress  $\tau_{\text{eff}}$  in flow direction is

$$\tau_{\text{eff}} = \tau_L + \tau_t \quad (22.91)$$

where  $\tau_t = -\rho \overline{v'u'}$ .

The first shear stress  $\tau_L$  is due to laminar flow,  $\tau_t$  is due to turbulence, and  $v'$  is the velocity component in  $y$  direction. The turbulent shear stress is given in terms of eddy viscosity or eddy diffusivity for momentum  $\epsilon_M$ :

$$\tau_t = \rho \epsilon_M \frac{du}{dy} \quad (22.92)$$

Prandtl postulated that the turbulent fluctuation  $u'$  and  $v'$  are proportional to the velocity gradient given by

$$v', u' \approx \ell \frac{du}{dy} \quad (22.93)$$

where the distance  $\ell$  is called the Prandtl mixing length.

Thus,

$$\tau_t = -\rho \overline{u'v'} = \rho \ell^2 \left( \frac{du}{dy} \right) \left( \frac{du}{dy} \right) = \rho \epsilon_M \left( \frac{du}{dy} \right) \quad (22.94)$$

where

$$\epsilon_M = \ell^2 \left( \frac{du}{dy} \right) \quad (22.95)$$

is the eddy viscosity.

The shear stress in turbulent flow is thus given by

$$\frac{\tau}{\rho} = (v + \epsilon_M) \frac{du}{dy} \quad (22.96)$$

Turbulent heat transfer is similar to turbulent momentum transfer. The heat flux in turbulent flow is thus comprised of molecular and turbulent components.

$$q'' = -\rho c_p \left( \alpha + \epsilon_H \frac{dT}{dy} \right) \quad (22.97)$$

where  $\epsilon_H$  is the eddy diffusivity.

The turbulent boundary layer heat transfer relations are given using Colburn analogy between fluid friction and heat transfer. Colburn analogy is exposed as

**TABLE 22.8**

Fluid Friction Analogy-Based Turbulent Heat Transfer Relations

*Local Relations:*

$$\text{St}_x \text{Pr}^{2/3} = 0.0296 \text{Re}_x^{-1/5} \quad 5 \times 10^5 \leq \text{Re}_x \leq 10^7$$

$$\text{St}_x \text{Pr}^{2/3} = 0.185 (\log \text{Re}_x)^{-2.586} \quad 10^7 \leq \text{Re}_x \leq 10^9$$

*Average:*

$$\overline{\text{St}} \text{Pr}^{2/3} = 0.037 \text{Re}_L^{-1/5} - 850 \text{Re}_L \quad \begin{array}{l} \text{Re}_{\text{air}} = 5 \times 10^5, \\ \text{Re}_L \leq 10^7 \end{array}$$

$$\text{St}_x \text{Pr}^{2/3} = \frac{Cfx}{2} \quad (22.98)$$

$$\text{where Stanton number } \text{St}_x = \frac{h_x}{\rho c_p v_f} = \frac{\text{Nu}_x}{\text{Re}_x \text{Pr}} \quad (22.99)$$

and

$$\text{Plate friction coefficient } Cfx = \frac{\tau_x}{\rho V_m^2 / 2} \quad (22.100)$$

Table 22.8 giving the level and plate average heat transfer relation for turbulent flow over a flat plate.

### 22.4.3.2 Internal Flow

For isothermal fully developed flow, the transition from laminar to turbulent flow occurs at critical Reynolds number,

$$\text{Re}_c = \frac{DV_m \rho}{\mu} \quad (22.101)$$

Though carefully conducted experiments have succeeded in obtaining critical Reynolds number as high as  $5 \times 10^4$ , for ordinary design purposes critical Reynolds number 2300 is satisfactory for flow inside channel.

For heated flows, the onset of turbulence occurs at lower critical Reynolds number as given below.

$$\text{For horizontal tubes } 600 \leq \text{Re}_c \leq 800$$

and

$$\text{for vertical tubes } 400 \leq \text{Re}_c \leq 600$$

The turbulent boundary layer in pipe flow is subdivided into three layers and are given below (Martinelli, 1947):

Laminar sublayer

$$0 \leq y^+ \leq 5 \quad u^+ = y^+ \quad (22.102)$$

Buffer sublayer

$$5 \leq y^+ \leq 30 \quad u^+ = -3.05 + 5.00 \ln y^+ \quad (22.103)$$

Fully turbulent sublayer

$$y^+ > 30 \quad u^+ = 5.5 + 2.5 \ln y^+ \quad (22.104)$$

A fully developed turbulent velocity profile in a pipe is approximately given by 1/7 power law:

$$\frac{u}{V_c} = \left( \frac{R-r}{R} \right)^{1/7} \quad (22.105)$$

where  $V_c$  is center line velocity. The average velocity is given by  $V_m = 0.817 V_c$ .

The wall shear stress for turbulent channel flow is given by

$$\tau_w = \frac{f}{4} \frac{\rho V_m^2}{2}. \quad (22.106)$$

Here  $f$  is the friction factor for turbulent flow.

The expressions for turbulent friction  $f$  in isothermal smooth pipe are listed below.

$$\text{(Karman)} \frac{1}{\sqrt{f}} = 2 \log_{10}(0.398 \text{Re} \sqrt{f}) \quad (\text{Valid for all Re}) \quad (22.107)$$

$$\text{(Karman } \pm 5\% \text{ fit)} f = 0.0056 + 0.5 \text{Re}^{-0.32} \quad 4 \times 10^3 \leq \text{Re} \leq 3 \times 10^6 \quad (22.108)$$

$$\text{McAdams } f = 0.184 \text{Re}^{-0.2} \quad 3 \times 10^4 \leq \text{Re} \leq 2 \times 10^6 \quad (22.109)$$

$$\text{Blasius } f = 0.316 \text{Re}^{-0.25} \quad \text{Re} < 3 \times 10^4 \quad (22.110)$$

Colebrook proposed empirical equation for friction factor for both smooth and rough pipe is

$$\frac{1}{\sqrt{f}} = -2 \log_{10} \left[ \frac{e/D}{3.70} + \frac{2.51}{\text{Re} \sqrt{f}} \right] \quad (22.111)$$

where  $e/D$  is the relative roughness of the pipe inner surface. A graphical representation of the Colebrook equation is the well-known Moody chart.

In the analysis of flow in nuclear reactor cores, the geometry usually consists of plates with narrow rectangular channels, rods having square or triangular subchannels with rods located at the vertices. The plates and rods have wire or grid spacers to maintain separation between plates and rods. For these geometries various expressions for friction factors are available in the literature that enables one to calculate the pressure drop in the reactor core (Rust, 1979; Todreas and Kazimi, 1990).

The fluid temperature profile for fully turbulent flow in a pipe can be obtained by Reynolds analogy for momentum and heat transfer. Reynolds assumed that in turbulent flow molecular diffusivity can be neglected, that is,  $\epsilon_M \gg \nu$ ,  $\epsilon_H \gg \alpha$ .

Then using expressions for turbulent shear stress and heat flux we have

$$\frac{\tau}{q''} = -\frac{f \epsilon_M}{\rho c_p \epsilon_H} \frac{du}{dT}. \quad (22.112)$$

By assuming  $\epsilon_M = \epsilon_H$  and that  $\tau/q''$  is constant the mean fluid temperature is obtained by integrating the above equations

$$T_m = T_w - \frac{q''_w V_m}{\tau_w c_p}. \quad (22.113)$$

With definition of heat transfer coefficient  $h = q''/(T_w - T_m)$  and wall shear (stress)  $\tau_w = f/4 \times \rho V_m^2/2$ , we have Reynolds analogy:

$$St = \frac{h}{\rho c_p V_m} = \frac{f}{8}. \quad (22.114)$$

Prandtl analogy (1925) provided further refinement to this model and is given as

$$St = \frac{f}{8} \frac{1}{1 + 5\sqrt{f/8}(\text{Pr} - 1)}. \quad (22.115)$$

For given velocity profile and relation between  $\epsilon_M$  and  $\epsilon_H$  the temperature profile in various geometries can be obtained by Martinelli (1947) equations

$$\frac{q''_w}{\rho c_p} \left(1 - \frac{y}{R}\right) = (\alpha + \epsilon_H) \frac{dT}{dy} \quad (22.116)$$

$$\frac{\tau_w}{\rho} \left(1 - \frac{y}{R}\right) = (\nu + \epsilon_M) \frac{du}{dy} \quad (22.117)$$

### 22.4.3.3 Heat Transfer Correlations

A considerable amount of experimental and theoretical work is available on the heat transfer coefficient for turbulent flow in channels. Through most of the correlations were developed based on circular pipe, with appropriate corrective factors, the correlations can be applied to various geometries. The heat transfer coefficients at the entrance region of the channel are higher. For nonmetallic fluids ( $\text{Pr} > 1$ ) the laminar layer is very thin compared to the turbulent region, and the heat transfer coefficient is less sensitive to boundary conditions. For liquid metals ( $\text{Pr} < 0.4$ ), conduction heat transfer is important and the boundary conditions have an impact on the heat transfer coefficient. For fully developed turbulent flow of nonmetallic fluid the heat transfer coefficient is expressed as:

$$\text{Nu} = C \text{Re}^a \text{Pr}^b \left(\frac{\mu_w}{\mu}\right)^d, \quad (22.118)$$

where:

$\mu_w$  is the fluid viscosity at the wall temperature

$a, b, c, d$  are constants which are dependent on fluid and channel geometry

Table 22.9 lists the heat transfer correlations for turbulent non-metallic fluid flow in various geometries and fluids. Table 22.10 gives the heat transfer relations for turbulent metallic fluid flow. For metallic fluids the turbulent heat transfer correlations are of the form

$$\text{Nu} = A + B(\text{Pe})^c = A + B(\text{Re} \times \text{Pr})^c \quad (22.119)$$

The constants  $A, B, C$  depend on geometry and boundary conditions. The constant  $A$  represents the conduction heat transfer contribution.

**TABLE 22.9**

Turbulent Heat Transfer Correlations for Nonmetallic Fluids

Fluid	Geometry	Correlation
Ordinary fluids	Circular tube	Seider and Tate (1936) <sup>a</sup> $\text{Nu} = 0.023 \text{Re}^{0.8} \text{Pr}^{0.4} \left(\frac{\mu_w}{\mu}\right)^{0.14}$
Ordinary fluids	Circular tube	Dittus and Boelter (1930) <sup>a</sup> $\text{Nu} = 0.023 \text{Re}^{0.8} \text{Pr}^n$ , $n = 0.4$ for heating, 0.3 for cooling
Ordinary fluids	Circular tube	Colburn (1933) <sup>b</sup> $\text{St} = \frac{h}{\rho c_p u} = 0.023 \text{Re}^{-0.2} \text{Pr}^{-0.666}$
Organic liquids	Circular tube	Silberberg and Huber (1959) <sup>a</sup> $\text{Nu} = 0.015 \text{Re}^{0.85} \text{Pr}^{0.3}$
Ordinary fluids	Annulus	Monrad and Pelton (1942) <sup>a</sup> $\text{Nu} = 0.02 \text{Re}^{0.8} \text{Pr}^{1/3} \left(\frac{D_2}{D_1}\right)^{0.53}$
Ordinary fluids	Thin rectangular	Hausen (1943) <sup>c</sup> $\text{Nu} = 0.116[\text{Re}^{2/3} - 125]\text{Pr}^{1/3} \left[1 + \frac{1}{3} \left(\frac{De}{2}\right)^{2/3}\right] \left(\frac{\mu}{\mu_w}\right)^{0.14}$
Ordinary fluids	Rod bundles— infinite array	Presser (1967) <sup>a</sup> $\text{Nu} = F \text{Re}^{0.8} \text{Pr}^{0.4}$ $F = 0.9090 + 0.0783P/D - 0.1283e^{-2.4(P/D-1)}$ $1.05 \leq P/D \leq 2.2$ —triangular array $F = 0.9217 + 0.1478P/D - 0.1130e^{-7(P/D-1)}$ $1.05 \leq P/D \leq 1.9$ —square array
Water	Rod bundles— infinite array	Weisman (1959) <sup>a</sup> $\text{Nu} = F \text{Re}^{0.8} \text{Pr}^{0.333}$ $F = 1.730P/D - 0.2609$ $1.1 \leq P/D \leq 1.5$ —triangular array $F = 1.826P/D - 1.0430$ $1.1 \leq P/D \leq 1.3$ —square array
Ordinary fluids	Rod bundles— finite array	Markoczy (1972) <sup>d</sup> $F = 1 + 0.9120 \text{Re}^{-0.1} \text{Pr}^{0.4} (1 - 2.0043e^{-B})$ $B = \frac{2\sqrt{3}}{\pi} (P/D)^2 - 1$ $1.0 \leq P/D \leq 2.0$ —triangular array $B = \frac{4}{\pi} (P/D)^2 - 1$ $1.0 \leq P/D \leq 1.8$ —square array

<sup>a</sup>  $\mu_w$  evaluated at  $T_w$ , and all other fluid properties evaluated at arithmetic mean temperature, applicable for  $0.7 < \text{Pr} < 120$ ,  $\text{Re} > 10,000$ ,  $L/D > 60$ .

<sup>b</sup>  $c_p, \text{St}$  evaluated at bulk fluid temperature, all fluid properties at film temperature  $T_f = (T_b + T_w)/2$ .

<sup>c</sup>  $De$ —equivalent diameter, applicable for  $T_f \geq 15^\circ\text{C}$ ,  $Z/De > 30$ .

<sup>d</sup>  $B$  evaluated for interior rods,  $3 \times 10^3 \leq \text{Re} \leq 10^6$ ,  $0.66 \leq \text{Pr} \leq 3.0$ .

**TABLE 22.10**

Turbulent Heat Transfer Correlations for Nonmetallic Fluids

Geometry	Correlation
Circular tube	$Nu = 7 + 0.025(Pe)^{0.8}$ Constant heat flux around tube $Nu = 5.0 + 0.025(Pe)^{0.8}$ Uniform axial wall temperature and uniform radial heat flux
Parallel plate	$Nu = 5.8 + 0.02(Pe)^{0.8}$ for fully developed flow constant heat flux through one wall only
Concentric annulus	$Nu = 5.25 + 0.0188(Pe)^{0.8} \left(\frac{D_2}{D_1}\right)^{0.3}$ uniform heat flux in the inner wall $\frac{D_2}{D_1} > 1.4$
Rod bundles	Westinghouse (Kazimi and Carelli, 1976): $Nu = 4.0 + 0.33(P/D)^{3.8} (Pe/100)^{0.86} + 0.16(P/D)^{5.0}$ For $1.1 \leq P/D \leq 1.4$ , $10 \leq Pe \leq 5000$ Schad-Kazimi (Kazimi and Carelli, 1976): $Nu = [-16.15 + 24.96(P/D) - 8.55(P/D)^2] Pe^{0.3}$ For $1.1 \leq P/D \leq 1.5$ , $150 \leq Pe \leq 1000$ $Nu = 4.496[-16.5 + 24.96(P/D) - 8.55(P/D)^2]$ For $Pe \leq 150$ Graber and Rieger (1973): $Nu = [0.25 + 6.2(P/D) + 0.32(P/D) - 0.007] Pe^{0.8-0.024(P/D)}$ For $1.25 \leq P/D \leq 1.95$ , $150 \leq Pe \leq 3000$ Borishanskii et al. (1969): $Nu = 24.15 \log[-8.12 + 12.76(P/D) - 3.65(P/D)^2]$ $+ 0.0174[1 - \exp(6 - 6P/D)][Pe - 200]^{0.9}$ For $1.1 \leq P/D \leq 1.5$ , $200 \leq Pe \leq 2000$ $Nu = 24.15 \log[-8.12 + 12.76(P/D) - 3.65(P/D)^2]$ For $1.1 < P/D \leq 1.5$ , $Pe \leq 200$

## 22.5 Two-Phase Flow

### 22.5.1 Introduction

A simultaneous flow of a gas and a liquid, a gas and a solid, two different liquids, or a liquid and a solid is described as a two-phase flow. Among these types of two-phase flow, gas-liquid flow is the most complex flow due to the deformability and the compressibility of the phases. The analysis of the two-phase flow is very important for liquid-cooled reactors. Two-phase flow occurs in the BWR core and in the steam generator of the PWRs. In order to analyze reactor systems with liquid-vapor mixtures, it is necessary to predict liquid-vapor density, pressure drop across a given channel length, flow stability, maximum flow rates, and heat transfer rates. As the liquid is vaporized, the mixture of vapor and liquid flow gives rise to interesting flow and heat transfer challenges.

Two-phase flows have all the complexity of single-phase turbulent flow, but in addition, there are now interfaces which have to be considered. Through the interfaces the interfacial tension forces act and the exchange of mass, momentum, and heat between the liquid and gas phases occur. Since the flow conditions in a flow channel vary along its length, over its cross section, and with time, the gas-liquid two-phase flow is an extremely complex three-dimensional (3D) transient problem. Often simplified descriptions of the



problem are sought which are both capable of analysis and retain important features of the flow. For two-phase flows, the respective distribution of the liquid and vapor phases in the flow channel expressed through some commonly observed flow structures is an important aspect of their description. The flow structures, which are defined as two-phase flow patterns, have particular identifying characteristics. The flow patterns have distinct heat transfer coefficients and pressure drops, and thus two-phase flow pattern prediction is an important aspect in reactor thermal hydraulics.

Here some of key two-phase parameters are defined. In two-phase flow, it is customary to describe flow conditions in terms of cross-sectional averages taken across the channel flow cross section.

The total mass flow rate through a flow channel is the sum of the mass rates of the two phases with subscript  $g$  for gas and  $f$  for liquid,

$$\dot{m} = \dot{m}_g + \dot{m}_f. \quad (22.120)$$

The gas and liquid mass fluxes (or mass velocities) are given as

$$G_g = \frac{\dot{m}_g}{A}, \quad G_f = \frac{\dot{m}_f}{A}, \quad (22.121)$$

where the total cross section,  $A$ , is the sum of the cross sections occupied by the gas and liquid phases  $A = A_g + A_f$ .

The volume flow rates of gas ( $Q_g$ ) and liquid ( $Q_f$ ) are defined as

$$Q_g = A_g u_g, \quad Q_f = A_f u_f, \quad (22.122)$$

where  $u_f$  and  $u_g$  are liquid and gas velocities and total volume flow rate is

$$Q = Q_g + Q_f. \quad (22.123)$$

The two-phase void fraction is defined as the volumetric fraction of gas phase in a two-phase control volume

$$\alpha = \frac{A_g}{A_g + A_f}. \quad (22.124)$$

The volumetric flux or superficial velocity is the phase volumetric flow divided by the total flow area as

$$j_f = \frac{Q_f}{A_g + A_f} \quad \text{and} \quad j_g = \frac{Q_g}{A_g + A_f}. \quad (22.125)$$

Using the definition of the void fraction and the volume flow rate, the superficial velocities are

$$j_f = u_f(1 - \alpha) \quad \text{and} \quad j_g = u_g \alpha \quad (22.126)$$

The superficial velocities are always equal to or less than the phase velocities.

The relative velocity is the difference between the phase velocities

$$u_r = u_g - u_f. \quad (22.127)$$

The ratio of the phase velocities or the velocity ratio or slip ratio is

$$S = \frac{u_g}{u_f}. \quad (22.128)$$

It is usually greater than unity which means that  $u_g$  is usually greater than  $u_f$ .

The two-phase system has three distinct qualities, equilibrium, real, and static.

Equilibrium quality  $x_e$  comes from an energy balance and is defined in terms of the enthalpy as

$$x_e = \frac{h - h_f}{h_{fg}}. \quad (22.129)$$

The *real* quality is the fraction of gas flow relative to the total flow and is given as

$$x = \frac{\dot{m}_g}{\dot{m}}. \quad (22.130)$$

The real quality is between 0 and 1.

The static quality is the mass fraction of gas to total mass

$$x = \frac{m_g}{m}. \quad (22.131)$$

The two-phase mixture density is given as

$$\rho_m = \alpha \rho_g + (1 - \alpha) \rho_f, \quad (22.132)$$

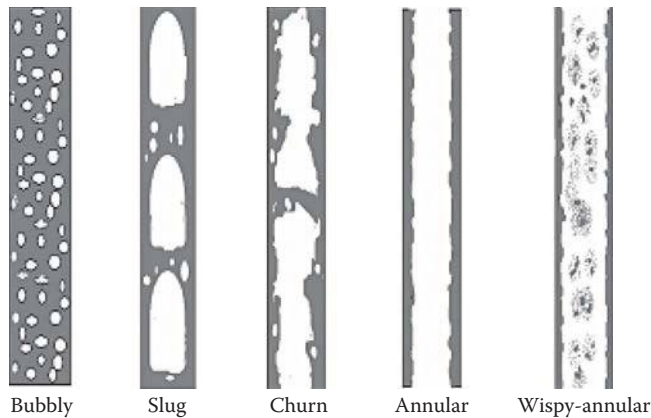
where  $\rho_f$  and  $\rho_g$  are liquid and gas densities, respectively.

Both quality and void fraction are unique to two-phase flows and are important for heat transfer. They are related to each other as

$$\alpha = \frac{1}{1 + \left(\frac{1-x}{x}\right) \frac{\rho_g}{\rho_f} S} \quad \text{and} \quad x = \frac{1}{1 + \left(\frac{1-\alpha}{\alpha}\right) \frac{\rho_f}{\rho_g} \frac{1}{S}} \quad (22.133)$$

### 22.5.2 Flow Patterns

The unique characteristic of the two-phase flow structure defined in terms of the flow pattern or flow regime have different void fraction and thus have large impact on pressure drop and heat transfer. The rate of exchange of mass, momentum, and energy between gas and liquid phases as well as between any multiphase mixture and the external boundaries depend on these internal flow geometries and interfacial area, and hence is dependent on flow pattern. The heat transfer models for predicting two-phase flow such as in boiling and condensation are based on the local flow pattern. This requires reliable flow pattern maps to identify what type of flow pattern exists at the local flow conditions. Analogous to predicting the transition from laminar to turbulent flow in single-phase flows, two-phase



**FIGURE 22.12**  
Two-phase flow patterns in vertical upward in a pipe.

flow pattern maps are used for predicting the transition from one type of flow pattern to another.

Figure 22.12 illustrates the common flow patterns for vertical upward flow. The order of the regimes is in terms of increasing gas flow for a fixed liquid flow.

*Bubbly flow:* The gas (or vapor) is dispersed in the form of discrete bubbles in the continuous liquid phase. The bubbles may vary in size and shape but they are typically nearly spherical and are much smaller than the diameter of the pipe itself.

*Slug flow:* The bubble dimensions are close to tube pipe diameter and the bubbles have a characteristic shape similar to a bullet with a hemispherical nose with a blunt tail end. They are commonly referred to as Taylor bubbles after the instability of that name. Taylor bubbles are separated from one another by slugs of liquid, which may include small bubbles.

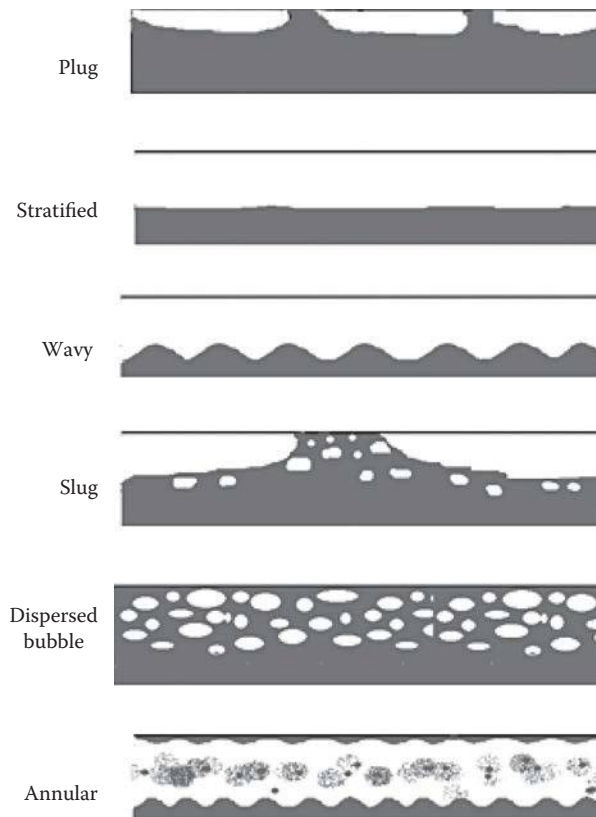
*Churn flow:* The flow is highly unstable with the fluid traveling up and down in an oscillatory fashion but with a net upward flow. This flow pattern is in fact an intermediate regime between the slug flow and annular flow regimes.

*Annular flow:* The liquid travels partly as an annular film on the wall of the tube and partly as small drops distributed in the gas which flows in the center of the tube. The interface is disturbed by high frequency waves and ripples.

*Wispy annular flow:* As the liquid flow rate is increased in annular flow, the concentration of drops in the gas core increases; ultimately, droplet coalescence in the core leads to large lumps or streaks (wisp) of liquid in the gas core.

*Mist flow:* At very high gas flow rates, the annular film is thinned by the shear of the gas core on the interface until it becomes unstable and is destroyed, such that all the liquid is entrained as droplets in the continuous gas phase analogous to the inverse of the bubbly flow regime. Impinging liquid droplets intermittently wet the tube wall locally. The droplets in the mist are often too small to be seen without special lighting and/or magnification.

The common flow patterns for horizontal and slightly upward inclined flows in a pipe are illustrated in Figure 22.13. Though the two-phase flow patterns in horizontal pipes are



**FIGURE 22.13**

Two-phase flow patterns in horizontal concurrent flow in pipe.

similar to those in vertical upward pipe flows, the influence of the gravity on the liquid distribution tends to stratify the liquid to the bottom of the tube and the gas to the top.

*Plug flow:* This flow regime has liquid plugs that are separated by elongated gas bubbles. The diameters of the elongated bubbles are smaller than the pipe such that the liquid phase is continuous along the bottom of the pipe below the elongated bubbles. Plug flow is also sometimes referred to as elongated bubble flow.

*Stratified flow:* In this flow complete separation of the two phases occurs with the gas flow near top and the liquid flow near bottom of the pipe. The gas and liquid are separated by an undisturbed smooth interface.

*Stratified-wavy flow:* Increasing the gas velocity in a stratified flow creates waves on the interface which travel in the direction of flow. The wave crests do not reach the top of the pipe.

*Intermittent flow:* In this flow interfacial waves have large amplitude waves intermittently washing the top of the pipe with smaller amplitude waves in between. Large amplitude waves often contain entrained bubbles. The top wall is nearly continuously wetted by the large amplitude waves and the thin liquid films left behind. Intermittent flow has subcategories, plug flow and slug flow, which are characterized as follows:

*Plug flow:* This flow regime has liquid plugs that are separated by elongated gas bubbles on top of the pipe and the liquid phase is continuous along the bottom of the pipe below the elongated bubbles. Plug flow is also sometimes referred to as elongated bubble flow.

*Slug flow:* With increased gas velocities the diameters of elongated bubbles become similar in size to the channel height.

*Dispersed bubble flow:* In this flow regime many small gas bubbles are distributed uniformly across the entire tube cross section when the gas and liquid velocities are high.

*Annular flow:* This flow regime is similar to vertical annular flow except that the liquid film is much thicker at the bottom of the tube than at the top.

*Mist flow:* Similar to vertical flow, at very high gas velocities, all the liquid may be stripped from the wall and entrained as small droplets in the now continuous gas phase.

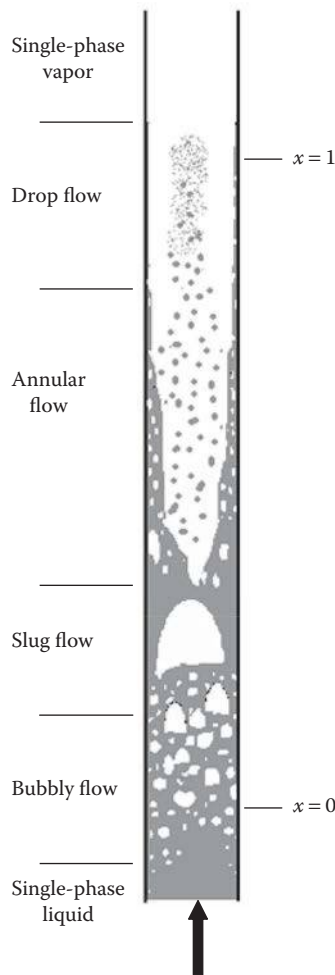
In nuclear reactors such as in BWR, two-phase flow in the core is more complicated. As the liquid is heated the void fraction continually increases along the channel. Figure 22.14 illustrates various flow regimes in a heated vertical channel. The flow pattern typically begins in the bubbly flow regime at the inlet at the onset of nucleate boiling in the tube. After bubbly flow the slug flow regime is entered and then the annular flow regime with its characteristic annular film of liquid. This film eventually dries out or the film is entrained by the interfacial vapor shear, taking the flow into the mist flow regime. The entrained liquid droplets may persist in the flow past the point of the vapor quality equal to 1.0.

### 22.5.3 Flow Pattern Maps

Flow pattern map is an attempt, on a two-dimensional graph, to separate the space into areas corresponding to the various flow patterns and is typically plotted on log-log axes using dimensionless parameters to represent the liquid and gas velocities. The commonly recommended map for gas-liquid upward vertical flow is the Hewitt and Roberts (1969) map. The map presents the flow regimes in terms of transition boundaries for slug, bubbly-slug, bubbly, churn, annular, and wispy annular flows with each coordinate as the superficial momentum fluxes for the respective phases. The Hewitt and Roberts (1969) map works reasonably well for air-water and steam-water systems. However, the transitions between the neighbor flow regimes appear as lines, which actually occur over a range of given coordinate terms.

The most widely quoted flow pattern maps for predicting the transition between two-phase flow regimes for adiabatic flow in horizontal tubes are those of Baker (1954) and Taitel and Dukler (1976). The Baker (1954) map for horizontal two-phase flow in tubes is presented with coordinates  $\dot{m}_g/\lambda$  and  $\dot{m}_f\psi$ . In order to use the map first the mass velocities of the liquid ( $\dot{m}_f$ ) and vapor ( $\dot{m}_g$ ) must be determined. Then the parameters lambda  $\lambda$  and  $\psi$  are calculated using the following equations:

$$\lambda = \left( \frac{\rho_g}{\rho_{\text{air}}} \frac{\rho_f}{\rho_{\text{water}}} \right)^{1/2}, \quad \psi = \left( \frac{\sigma_{\text{water}}}{\sigma} \right) \left[ \left( \frac{\mu_f}{\mu_{\text{water}}} \right) \left( \frac{\rho_{\text{water}}}{\rho_f} \right)^2 \right]^{1/3} \quad (22.134)$$



**FIGURE 22.14**  
Flow patterns in a heated channel.

where  $\rho$ ,  $\sigma$ , and  $\mu$  are fluid density, surface tension, and viscosity. The reference fluid properties are for air and water and are given as  $\rho_{\text{water}} = 1000 \text{ kg/m}^3$ ,  $\rho_{\text{air}} = 1.23 \text{ kg/m}^3$ ,  $\sigma_{\text{water}} = 0.072 \text{ N/m}$ , and  $\mu_{\text{water}} = 0.001 \text{ N s/m}^2$ . Taitel and Dukler (1976) introduced theoretical models for determining transition boundaries of five flow regimes in horizontal and near horizontal two-phase gas-liquid flow. The theory was developed in dimensionless form, and the flow regime boundaries were introduced as a function of four dimensionless parameters. These parameters are the Martinelli parameter  $X$ , the gas Froude number  $Fr_g$ , and the parameters  $T$  and  $K$  and is composed of three graphs (Martinelli and Nelson, 1948; Lockhart and Martinelli, 1949).

The Martinelli parameter is

$$X = \left( \frac{(dp/dz)_f}{(dp/dz)_g} \right)^{1/2} \quad (22.135)$$

The gas-phase Froude number is

$$Fr_g = \frac{\dot{m}_g}{[\rho_g(\rho_f - \rho_g)dg]^{1/2}}. \quad (22.136)$$

The plot of gas-phase Froude number  $Fr_g$  versus  $X$  shows the transitions between annular and other flow regimes.

The  $T$  parameter is

$$T = \left[ \frac{|(dp/dz)_f|}{g(\rho_f - \rho_g)} \right]^{1/2}, \quad (22.137)$$

where  $g$  is gravitational acceleration. The pressure gradient is given as

$$\left( \frac{dp}{dz} \right)_f = \frac{2f_f \dot{m}_f^2}{\rho_f d}. \quad (22.138)$$

The plot of  $T$  versus  $X$  shows the transitions between bubbly and intermittent flow regimes.

Here the  $d$  is the pipe diameter. The friction factor is given as

$$f_f = \frac{16}{Re_f} \quad \text{for laminar flow } Re_f < 2000 \quad (22.139)$$

$$f_f = \frac{0.079}{Re_f^{1/4}} \quad \text{for turbulent flow } Re > 2000. \quad (22.140)$$

The Reynolds number for  $k$  phase is defined for ( $k = \text{gas or liquid}$ )

$$Re_k = \frac{\dot{m}_k d}{\mu_k}. \quad (22.141)$$

The  $K$  parameter is defined as

$$K = Fr_g Re_f^{1/2}. \quad (22.142)$$

The plot of  $K$  versus  $X$  shows the transitions between stratified and stratified-wavy flow regimes.

### 22.5.4 Countercurrent Flow and Flooding

There are a number of situations where countercurrent two-phase flow can exist in nuclear reactor coolant channels. For example, during emergency core cooling of the BWR rod bundles at low flow has steam and water countercurrent flow. The water flow rate can continue for certain ranges of water and steam flow rates. However, the relative velocity between the steam and water creates waves on the liquid surface for large gas velocities. And as the steam velocity increases the waves reach the channel walls and block the downward flow of the water. This transition point is called flooding or countercurrent flow limit. Further increase in steam velocity leads another transition where water is carried upward and thus flow reversal occurs. The transitions are associated with large pressure drop in the pipe.

The transition conditions are given in terms of correlations with two nondimensional numbers—Wallis number which is the ratio of inertia force to hydrostatic force on a bubble or drop of diameter  $d$ , and Kutateladze number which is the ratio of the inertia force to hydrostatic force on a bubble with dimension given by Laplace constant  $\{g\sigma(\rho_f - \rho_g)\}^{1/2}$  (Wallis, 1969). The Wallis number is defined as

$$j_k^* = j_k \left[ \frac{\rho_k}{gd(\rho_f - \rho_g)} \right]^{1/2} \quad (22.143)$$

where  $k = f$  for liquid and  $g$  for vapor or gas.

Kutateladze number is defined as

$$Ku_k = j_k \left[ \frac{\rho_k}{\{g\sigma(\rho_f - \rho_g)\}^{1/2}} \right]^{1/2} \quad (22.144)$$

Wallis has provided an empirical correlation for flooding point as

$$j_g^{*1/2} + j_g^{*1/2} = c. \quad (22.145)$$

The value of  $c$  depends on the pipe entrance conditions. Typically, its value is 0.88 for round-edged pipes and 0.725 for sharp-edged pipes. The flow reversal point was given by a correlation

$$j_g^* = c_1, \quad (22.146)$$

where  $c_1$  is constant with 0.9 for round-edged pipes and 0.5 for sharp-edged pipes.

Pushkin and Sorokin (1969) studied flooding in a wide range of vertical pipes having diameter range from 6 to 309 mm. They provided with a flooding correlation using Kutateladze number which is independent of pipe diameter as

$$Ku = 3.2. \quad (22.147)$$

## 22.5.5 Flow Models

There are three commonly used two-phase flow models: the homogeneous flow model, separated flow model, and two fluid model. The thermal nonequilibrium allows one or both of the phases to have temperatures other than the saturation temperature and unequal velocities. Separated flow allows not only potential higher gas/vapor velocity but also the possibility of countercurrent flows. The homogeneous flow model does not allow countercurrent flows.

### 22.5.5.1 Homogeneous Equilibrium Model

In homogeneous equilibrium model (HEM) the velocity, temperature, and pressure between the phases or components are assumed equal. The two phases exit at the saturation temperature for the prevailing pressure. The mixture is treated as a single fluid. This model is particularly useful for high pressure and high flow rate conditions. The HEM governing equations, such as mass, momentum, and energy, resemble those for a pseudo-fluid with mixture properties and an equation of state which links the phases



to obtain these mixture thermodynamic properties. For HEM, the slip ration is 1 and the relation between void to quality is given as

$$\alpha = \frac{1}{1 + \left(\frac{1-x}{x}\right) \frac{\rho_g}{\rho_f}} \quad (22.148)$$

### 22.5.5.2 Separated Flow Model—Drift Flux Model

In the separated flow model, the restriction on equal phase velocities is relaxed. So it is essentially a thermal equilibrium mixture model with an algebraic relation between the velocities (or a slip ratio) of the two phases. The unequal phase velocities become important when the densities between the phases are quite different in the presence of a gravitational potential field or large pressure gradients. Since the liquid density is higher, buoyancy effects tend to induce a drift velocity of the gas phase in the liquid phase. This model is different from the HEM model only in allowing the two phases to have different velocities that are related via a predetermined relation. There are now two momentum equations in the separate flow model. In each equation there appears a term which represents the friction force at the phase interface caused by the relative velocity between the phases. For the equations to be solved separately a constitutive relation model for this momentum transfer term is required. In the separate flow model the properties are not averaged exclusively using the void fraction and density of the phases and require a separate constitutive relation that relates the volume fraction to the flowing mass fraction. A method, introduced by Zuber and Findley (1969), and also described by Wallis (1969) gives general approach to obtain the slip ratio by considering the average velocity of the vapor in the channel. A particular separated flow model in which the average void fraction is specified in terms of drift velocity is called drift flux model. The basic continuity equation of the drift flux model is given as

$$\bar{\alpha} = \frac{j_g}{C_0(j_f + j_g) + u_{gj}} \quad (22.149)$$

where:

$\bar{\alpha}$  is the pipe averaged void fraction

$C_0$  is a distribution parameter correcting the basic one-dimensional model to allow for concentration and velocity profiles across the channel

$u_{gj}$  is a void weighted mean drift velocity

The values of  $C_0$  and  $u_{gj}$  are dependent upon the flow regime and pressure range.

### 22.5.5.3 Two-Fluid Model

The two-fluid model allows the phases to have thermal nonequilibrium as well as unequal velocities. In this model, each phase or component is treated as a separate fluid with its own velocity, temperature, and pressure. Thus, each phase has three independent set of governing balance equations for mass momentum and energy. The velocity difference as in the separated flow is induced by density differences and the temperature differences between the phases is fundamentally induced by the time lag of energy transfer between the phases at the interface as thermal equilibrium is reached. The two-fluid model

is needed for very fast transients and nonequilibrium conditions where the time lag for reaching thermal equilibrium between the phases may become significant in comparison to the characteristic time it takes for flow conditions to change. Due to the independent velocity, temperature, and pressure for each phase, the two-fluid model interfacial transport coefficients are required for mass, momentum, heat, and pressure transfer between the interface.

### 22.5.6 Pressure Drop and Pressure Gradient

The local total pressure drop in a two-phase flow channel is given as sum of three separate components: a frictional term, an acceleration term, and a static head term.

$$\Delta p_{\text{total}} = \Delta p_{\text{acceleration}} + \Delta p_{\text{friction}} + \Delta p_{\text{gravity}} \quad (22.150)$$

The pressure gradient for one-dimensional flow along  $z$  direction is thus given as

$$-\frac{dp}{dz} = \left(\frac{dp}{dz}\right)_{\text{acceleration}} + \left(\frac{dp}{dz}\right)_{\text{friction}} + \left(\frac{dp}{dz}\right)_{\text{gravity}} \quad (22.151)$$

where:

$$\left(\frac{dp}{dz}\right)_{\text{acceleration}} = \frac{d}{dz} \left(\frac{G_m}{\rho'_m}\right), \quad \left(\frac{dp}{dz}\right)_{\text{friction}} = \frac{\tau_w P_w}{A_z}, \quad \left(\frac{dp}{dz}\right)_{\text{gravity}} = \rho_m g \cos \theta, \quad (22.152)$$

$\tau_w$  is the circumferential averaged wall shear stress, average

$\rho_m$  is the two-phase mixture density

$A_z$  is the flow cross sectional area

$P_w$  is the wetted perimeter

$G_m$  is the mixture mass flux

$\rho'_m$  is the dynamic density or mixing cup density and is given as

$$\rho'_m = \frac{G_m^2}{\rho_g \alpha u_g + \rho_f (1 - \alpha) u_f}. \quad (22.153)$$

If velocities of both phases are uniform across the channel, using the relations for  $u_g$  and  $u_f$  from  $u_g = G_g / \alpha \rho_g = xG / \alpha \rho_g$  and  $u_f = (1 - x)G / (1 - \alpha) \rho_f$  the mixing cup density is given in terms of flow quality as

$$\frac{1}{\rho'_m} = \frac{x^2}{\rho_g \alpha} + \frac{(1 - x)^2}{\rho_f (1 - \alpha)}. \quad (22.154)$$

The friction gradient for two-phase flow is generally expressed similar to the single-phase flow as

$$\left(\frac{dp}{dz}\right)_{\text{friction}} = \frac{\tau_w P_z}{A_z} = \frac{f_{TP}}{D_e} \left(\frac{G_m^2}{2\rho'_m}\right), \quad (22.155)$$

where the hydraulic equivalent diameter is given as  $D_e = 4A_z / P_w$ .

The pressure drop due to friction for two-phase flow is related to friction factors  $f$  and multipliers  $\phi$ , defined for single phase (gas or liquid) flowing at the same mass flux as the total two-phase mass flux through the following relations.

$$\Delta p_{\text{friction}}^{\text{TP}} = \frac{f_{\text{TP}}}{D_e} \left( \frac{G_m^2}{2\rho_m'} \right) = \phi^2 \Delta p_{\text{friction}}^{\text{SP}} = \phi^2 \frac{f_{\text{SP}}}{D_e} \left( \frac{G_m^2}{2\rho_{\text{SP}}} \right), \quad (22.156)$$

where scripts TP and SP refer to two phase and single phase, respectively. The multiplier is given as

$$\phi^2 = \frac{f_{\text{TP}}}{f_{\text{SP}}} \left( \frac{\rho_{\text{SP}}}{\rho_m'} \right). \quad (22.157)$$

### 22.5.6.1 HEM Model Pressure Gradient

Several simplifications are possible with HEM model since phase velocities are equal. For HEM model  $\rho_m' = \rho_m = \alpha\rho_g + (1-\alpha)\rho_f$ . The two-phase friction multiplier for liquid only becomes

$$\phi_{\text{fo}}^2 = \frac{f_{\text{TP}}}{f_{\text{SP}}} \left( \frac{\rho_{\text{SP}}}{\rho_m} \right) = \frac{f_{\text{TP}}}{f_{\text{SP}}} \left( 1 + x \frac{v_f - v_g}{v_f} \right), \quad (22.158)$$

where  $v_f$  and  $v_g$  are specific volumes of liquid and gas, respectively.

The expression for acceleration pressure gradient for constant area channel assuming  $v_f$  and  $v_g$  are assumed independent of  $z$ , and the liquid is incompressible, given as

$$\left( \frac{dp}{dz} \right)_{\text{acceleration}} = G_m^2 \left( x \frac{\partial v_g}{\partial p} \frac{dp}{dz} + (v_f - v_g) \frac{dx}{dz} \right). \quad (22.159)$$

The total pressure gradient for HEM model is thus

$$-\left( \frac{dp}{dz} \right)_{\text{HEM}} = \frac{(f_{\text{TP}}/D_e)(G_m^2/2\rho_m) + G_m^2(v_f - v_g)(dx/dz) + \rho_m g \cos \theta}{1 + G_m^2 x (\partial v_g / \partial p)} \quad (22.160)$$

To evaluate the two-phase friction multiplier in HEM, the following approximation is used for two-phase friction factor:

$$\frac{f_{\text{TP}}}{f_{\text{fo}}} = \frac{C_1 / \text{Re}_{\text{TP}}^n}{C_1 / \text{Re}_{\text{TP}}^n} = \left( \frac{\mu_{\text{TP}}}{\mu_f} \right)^n, \quad (22.161)$$

where the  $C_1 = 0.318$  and  $n = 0.25$  or  $C_1 = 0.184$  and  $n = 0.2$  for turbulent flow. The two-phase viscosity  $\mu_{\text{TP}}$  is given by the following formulas:

$$\frac{\mu_{\text{TP}}}{\mu_f} = \left[ 1 + x \left( \frac{\mu_f}{\mu_g} - 1 \right) \right]^{-1} \quad (\text{McAdams}); \quad (22.162)$$

$$\frac{\mu_{\text{TP}}}{\mu_f} = \left[ 1 + x \left( \frac{\mu_g}{\mu_f} - 1 \right) \right] \quad (\text{Cichitti});$$

$$\frac{\mu_{TP}}{\mu_f} = \left[ 1 + \alpha \left( \frac{\mu_g}{\mu_f} - 1 \right) \right] \quad (\text{Dukler}) \quad (22.163)$$

### 22.5.6.2 Two-Phase Friction Multipliers

The most widely used correlations for predicting two-phase pressure drop are those of Lockhart and Martinelli (1949) for adiabatic flows and Martinelli and Nelson (1948) for heated flows. Both correlations assume that there is no flow regime change along the pipe length. For adiabatic flow in horizontal pipe flow, the experimental data is classified into four types of flows: (1) flows of both liquid and gas are turbulent ( $t-t$ ); (2) flow of liquid is viscous and that of gas is turbulent ( $v-t$ ); (3) flow of liquid is turbulent and that of gas is viscous ( $t-v$ ); and (4) flow of both flows liquid and gas are viscous ( $v-v$ ).

The friction pressure gradient along the channel is expressed in terms of the pressure gradient due to flow of each phase alone in the channel in terms of the following parameters:

$$\phi_f^2 = \frac{(dp/dz)_{friction}^{TP}}{(dp/dz)_{friction}^f}, \quad \phi_g^2 = \frac{(dp/dz)_{friction}^{TP}}{(dp/dz)_{friction}^g}, \quad (22.164)$$

In this analysis Lockhart and Nelson postulated that  $\phi_f^2$  and  $\phi_g^2$  are related to each other through parameter  $X$  as

$$X^2 = \frac{(dp/dz)_{friction}^f}{(dp/dz)_{friction}^g} \quad (22.165)$$

By substituting the single-phase pressure gradient in terms of Reynolds number and friction factors we have

$$X^2 = \frac{\text{Re}_g^m C_f}{\text{Re}_f^n C_g} \left( \frac{G_g}{G_f} \right)^2 \frac{\rho_g}{\rho_f} \quad (22.166)$$

The exponents  $m$  and  $n$  and coefficients  $C_f$  and  $C_g$  are determined from appropriate single-phase expressions for laminar and turbulent flow pressure loss and are tabulated in Table 22.11 for four flow types of gas and liquid.

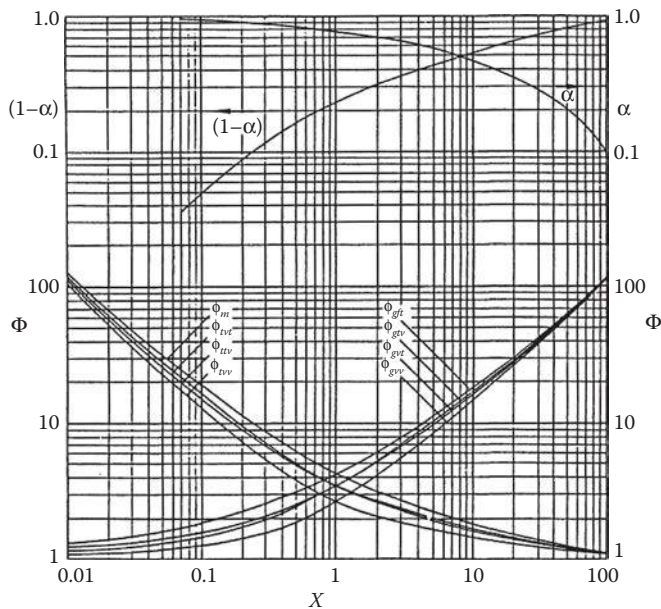
Lockhart and Martinelli (1949) suggested that smooth curve correlation can be obtained for values of  $\phi_f^2$  and  $\phi_g^2$  as function of  $X$  as shown in Figure 22.15.

**TABLE 22.11**

Value of Exponents  $m$  and  $n$  and Constants  $C_f$  and  $C_g$  for Various Flow Types

	$t-t$	$v-t$	$t-v$	$v-v$
$n$	0.2	1	0.2	1.0
$m$	0.2	0.2	1.0	1.0
$C_f$	0.184 <sup>a</sup>	64	0.184	64
$C_g$	0.184 <sup>a</sup>	0.184 <sup>a</sup>	64	64

<sup>a</sup> For smooth pipes.



**FIGURE 22.15**

The two-phase friction pressure drop ratio as function of Lockhart and Martinelli parameter  $X$  for different flow combinations. (Data from Todreas, N.E. and Kazimi, M.S., *Nuclear System. I: Thermal Hydraulic Fundamentals*, Hemisphere, New York, 1990.)

These curves are represented with the following relations:

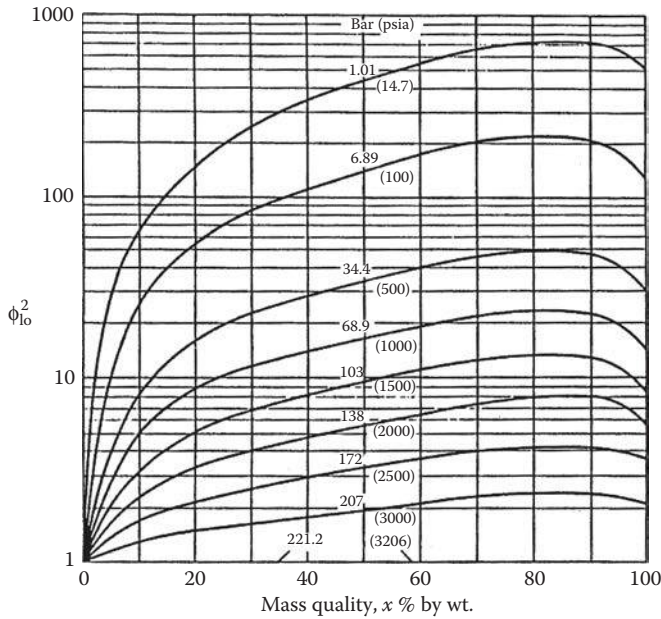
$$\phi_f^2 = \frac{(dp/dz)_{\text{friction}}^{\text{TP}}}{(dp/dz)_{\text{friction}}^f}. \quad (22.167)$$

For heated horizontal channels, Martinelli and Nelson (1948) assumed the total pressure loss is given by the sum of a pressure loss due to friction and a pressure loss due to increased fluid momentum as vaporization occurs. The local two-phase friction pressure gradients in boiling water were calculated using isothermal flow pressure loss data assuming turbulent-turbulent flow.

The ratio of two-phase friction pressure loss to single-phase pressure loss was computed by integrating local pressure gradients at a constant pressure and heat flux from saturated liquid up to any quality  $x$ . Figure 22.16 illustrates two-phase friction pressure losses as a function of exit quality and system pressure. The friction pressure loss for steam-water mixtures at any pressure and exit quality, where quality varies linearly in the channel, is obtained by multiplying the friction loss for total flow rate as liquid by the ratio obtained from the appropriate curve in Figure 22.16.

### 22.5.7 Choked Flow

Choked flow also called critical flow is defined in single-phase flow as the flow when the fluid Mach number which is the ratio between the local fluid velocity and the local sound speed in the fluid approaches unity. For compressible single-phase flow or for gas-liquid two-phase flow when Mach number equal to one, the pressure gradient asymptotically

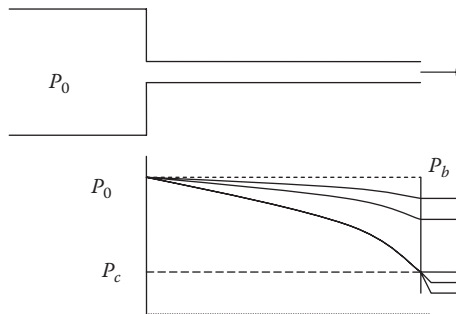


**FIGURE 22.16**

Ratio of local two-phase pressure gradient to pressure gradient for liquid flow only as a function of quality and pressure. (Data from Todreas, N.E. and Kazimi, M.S., *Nuclear System. I: Thermal Hydraulic Fundamentals*, Hemisphere, New York, 1990.)

approaches an infinite gradient (shock wave). Similar to the critical flow of compressible single-phase fluid, flow of a two-phase mixture in a channel may also become choked. However, for two-phase flows the sound velocity is frequently smaller than in the flow of either phase separately. The reason is that the mixture has a density of the order of the heavy liquid phase and compressibility close to that of the vapor phase. In reactors during abnormal transients, such as LOCAs, choked flow rates limit the flow from reactor vessel. Thus, choked flow is important in the analysis of nuclear reactor safety as it determines the coolant inventory loss in case of leakage from primary coolant system of the reactor.

As shown in Figure 22.17, the flow from a tank containing compressible fluid at stagnation pressure  $P_0$  to the backpressure  $P_b$ . As the  $P_b$  is decreased, the flow rate increases until  $P_b$  reaches the critical value  $P_c$  and then any further decrease in the  $P_b$  will not increase the



**FIGURE 22.17**

Critical flow configuration.

flow rate. Then this maximum flow rate is said to be choked flow. For adiabatic flow, both the critical flow, and the critical pressure are functions only of the stagnation state and the channel geometry. Mathematically, maximum flow rate is given as  $dG/dP = 0$ , where  $P$  is pressure, mass flux  $G = m/A$ ,  $m$  is mass flow rate, and  $A$  is channel cross section. In terms of specific volume  $v$ , the critical mass flux  $G_c$  is given as

$$G_c = \left( \frac{dP}{dv} \right)^{1/2} . \quad (22.168)$$

The sound speed for isentropic flow is defined as

$$c = \left( \frac{dP}{d\rho} \right)_s^{1/2} . \quad (22.169)$$

For an ideal gas, the sound speed is given as  $c = \sqrt{kRT}$ , where  $k$  is the ratio of specific heats,  $R$  is the gas constant, and  $T$  is absolute temperature. The critical pressure ratio for ideal gas is given by

$$\frac{P_c}{P_0} = \left( \frac{2}{k+1} \right)^{k/(k-1)} . \quad (22.170)$$

The theoretical treatments in two-phase critical flows are divided into two broad categories: (1) thermodynamic equilibrium models and (2) nonequilibrium models which account for non-equilibrium effects between phases. The effect of flow regime is incorporated in these models by either assuming homogeneous flow (slip ratio of unity) or accounting for slip between phases and determining expressions for the slip ratio. Nonequilibrium effects appear to be important for critical flows in pipes of short length ( $L/D < 6$ ) and phase equilibrium may be a reasonable assumption for flows through pipes of greater lengths ( $L/D > 12$ ). In the HEM, the critical mass velocity ( $G_c$ ) occurs for compressible fluids when all the energy obtained from a differential pressure loss is transformed into a corresponding increase in kinetic energy. The critical mass flux for HEM and separated flow model are respectively given as

$$\text{HEM: } G_c = \left( \frac{1}{x} \frac{dP}{dv} \right)^{1/2} , \quad (22.171)$$

$$\text{Separated flow model: } G_c = \left( \frac{\alpha}{x} \frac{dP}{dv} \right)^{1/2} . \quad (22.172)$$

For isentropic expansion of two-phase mixture in thermal equilibrium, the expression for critical mass flux is obtained from enthalpy as

$$G_c = \frac{\sqrt{2(h_0 - h)}}{[S(1-x)v_f + xv_g][x + (1-x)/S^2]^{1/2}} , \quad (22.173)$$

where:

$h_0$  is the stagnation enthalpy

$x$  is the flow quality

$v_f$  and  $v_g$  are specific enthalpy of liquid and gas phases

The enthalpy of the two-phase mixture is given by

$$h = h_f + (1-x)h_{fg}. \quad (22.174)$$

The relations for the velocity slip ratio  $S$  are as follows:

$$\text{HEM model: } S = 1.0 \quad (22.175)$$

$$\text{Moody model: } S = \left( \frac{v_g}{v_f} \right)^{1/3} \quad (22.176)$$

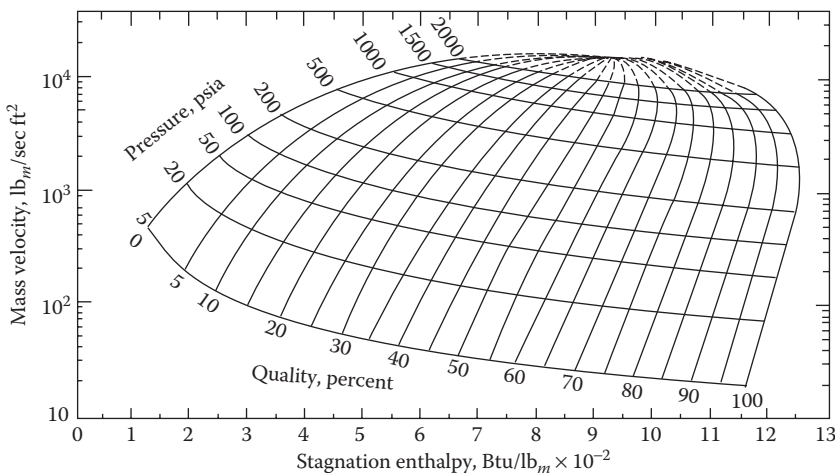
$$\text{Fauske model: } S = \left( \frac{v_g}{v_f} \right)^{1/2} \quad (22.177)$$

The critical mass flux is thus  $G_c = G_c(h_0, P_0, P_c, S)$ . The Moody model (1965) is based on maximizing specific kinetic energy of the mixture with respect to the slip ratio whereas the Fauske model (1961, 1965) is based on the flow momentum with respect to the slip ratio. In Figure 22.18, the critical discharge rate of water at various stagnation pressure and enthalpy with Fauske slip model is shown.

The thermal nonequilibrium effects become important when the length of time for a fluid particle to be expelled through an opening is small ( $<1$  ms), since a finite time is required for the bubble growth resulting from rapid pressure change in choking flow. In Figure 22.19, the experimental data on critical pressure ratio dependence on the length to diameter ratio is shown. The following relations for critical mass flux are given depending on the  $L/D$ :

$$\text{For } L/D = 0 \text{ (orifices): } G_c = 0.61\sqrt{2\rho_f(P_0 - P_b)} \quad (22.178)$$

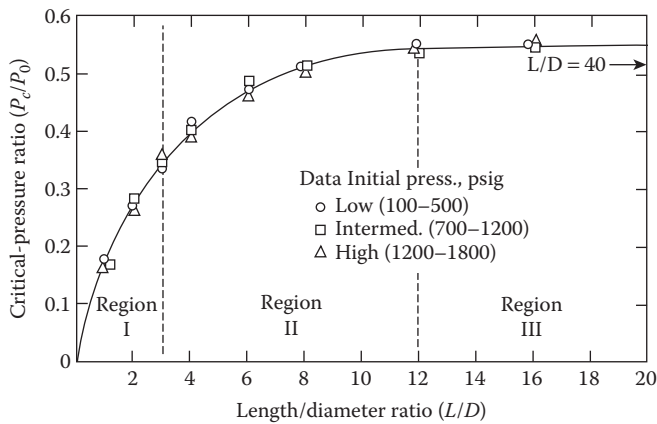
$$\text{For } 0 < L/D \text{ (regions I, II, and III): } G_c = 0.61\sqrt{2\rho_f(P_0 - P_c)} \quad (22.179)$$



**FIGURE 22.18**

Choked mass flow rates as function of stagnation enthalpy and pressure for steam water with Fauske slip model. (Data from Todreas, N.E. and Kazimi, M.S., *Nuclear System. I: Thermal Hydraulic Fundamentals*, Hemisphere, New York, 1990.)





**FIGURE 22.19**

Experimental critical pressure ratio data as a function of length. (Data from Todreas, N.E. and Kazimi, M.S., *Nuclear System. I: Thermal Hydraulic Fundamentals*, Hemisphere, New York, 1990.)

where  $P_c$  is obtained from Figure 22.19.

It should be noted that for region II ( $3 < L/D < 12$ ) the flow is less than that predicted by above equation.

### 22.5.8 Flow Instability

A fluid flow in heated channel with subcooled inlet condition undergoes large volume change in a nonuniform manner. As the thermal and hydraulic properties of the flow change continuously along the channel, the flow at any axial point in the channel can never be fully developed either thermally or hydrodynamically. As the flow is not in equilibrium, the flow properties fluctuate both upstream and downstream of the point considered often leading to instability (Collier, 1981). Flow instabilities are of different types depending on the system configuration and operating conditions. In reactor the oscillations in the flow dramatically reduce the ability of the coolant to remove heat from the core. On the basis of primary features such as oscillation periods, amplitudes, and relationships between pressure drop and flow rate, flow instabilities have been classified into several types (Boure et al., 1973). The flow instabilities are caused by self-oscillation of two-phase flow heated channel due to density wave effects. These instabilities are primarily classified as static and dynamic. Some static and dynamic-type instabilities occur particularly during startup conditions. In addition to the flow instability, BWRs are susceptible for the following two types of instabilities: (a) control system instabilities—these are due to the malfunction of reactor hardware. Suitable control mechanisms are provided to deal with this type of instability. (b) Coupled neutronic and thermal-hydraulic instabilities, also called reactivity instabilities, are due to the void and power feedback effects on neutron kinetics and thermal hydraulics, respectively.

#### 22.5.8.1 Static Instabilities

A static instability occurs when a small perturbation from original steady-state flow leads to a new stable operating condition which is not in the vicinity of the original state. The mechanism and the threshold conditions are predicted using steady-state characteristics of the system. Pressure drop characteristics of a flow channel, nucleation properties, and

flow regime transitions play an important role in the characterization of static instabilities. Critical heat flux (CHF), which limits the heat transfer capability of boiling systems, is influenced by the static instabilities. Ledinegg instability, flow pattern transition instability, geysering, chugging, and vapor burst are categorized as static instabilities (Boure et al., 1973).

*Ledinegg instability or flow excursive instability* is characterized by a sudden change in the flow rate to a lower value or a flow reversal. This happens when the slope of the channel demand pressure drop versus flow rate curve (internal characteristics of the channel) becomes algebraically smaller than that of the loop supply pressure drop vs. flow rate curve (external characteristics of the channel). Physically, this behavior exists when the pressure drop decreases with increasing flow. The criterion or condition for Ledinegg instability to occur is expressed by the inequality (Boure et al., 1973)

$$\left(\frac{\partial \Delta p}{\partial G}\right)_{\text{int}} \leq \left(\frac{\partial \Delta p}{\partial G}\right)_{\text{ext}} \quad (22.180)$$

Ledinegg instability can be avoided by making the slope of the external characteristics steeper than that of the internal characteristics by providing inlet throttle valve (Boure et al., 1973).

The onset of Ledinegg instability may sometimes cause wall dryout, but often lies very close to the dryout point because the onset may be followed by limit cycle fluctuations in the flow.

*Flow pattern transition instability* is mainly caused by different pressure drop characteristics of various flow patterns. For example, when the bubbly flow changes to annular flow due to large void generation, the channel pressure drop decreases. For constant pressure drop in a heated channel, when a bubbly flow changes to annular flow pattern the flow rate increases leading to decreased vapor generation. This leads the flow pattern to revert back to either bubbly or slug flow pattern and the cycle is repeated. Flow pattern transition instability and Ledinegg instability have similar characteristics, and the former is of greater concern because it triggers CHF due to drastic flow changes.

*Geysering* occurs at low powers and low circulation flow rates. It occurs due to bubble formation, growth, and its subsequent collapse during start-up. In vertical channels during start-up, with high inlet subcooling, voids are generated and a large slug of bubbles is formed, which grows due to decrease in the hydrostatic pressure head as it moves toward the exit. The vapor then mixes with the liquid in the subcooled riser and is condensed there. The condensed liquid re-enters the channel and restores nonboiling conditions. This process repeats periodically and causes flow oscillations.

*Vapor burst instability* occurs due to sudden vaporization of the liquid phase with rapid decrease in mixture density. For example, a very clean and smooth heated surface may require high wall superheat for nucleation. The fluid adjacent to the surface is highly superheated and vapor generation is rapid when nucleation starts. This in turn ejects liquid from the heated channel. Rapid vaporization cools the surface, and cooler liquid keeps the vaporization suppressed until wall temperature reaches required nucleation superheat and the process repeats. Vapor burst instabilities are observed during reflood phase of the reemergence core cooling of reactor.

### 22.5.8.2 Dynamic Instabilities

Dynamic instability is caused by the dynamic interaction between the flow rate, pressure drop, void fraction, and so on. The mechanism involves the propagation of disturbances

by pressure and void or density waves. Density wave oscillations (DWOs), parallel channel instability, pressure drop oscillations (PDOs), and so on fall under this class (Boure et al., 1973).

DWOs are due to multiple regenerative feedback between the flow rate, vapor generation rate, and pressure drop. Under certain conditions, the feedback can become 180° out-of-phase with the perturbation and result in self-sustained oscillations. DWOs are classified into two types: Type I and Type II (Fukuda and Kobori, 1979).

Type I instability occurs at very low steam quality condition (at low power and high inlet subcooling) and is of specific importance in natural circulation BWRs. At low pressure and low flow quality, the hydrostatic head (as imposed by the core and riser void fraction) is very sensitive to flow rate fluctuations. These are low frequency oscillations dominated by gravity effects in unheated riser section. Type II instability occurs at relatively high power and low inlet subcooling and is due to the interaction between the two-phase flow pressure loss, mass flux, and void formation and its propagation in the core. These are high frequency oscillations dominated by two-phase frictional pressure drop. This phenomenon is important both in forced and natural circulation reactors. Most two-phase heat exchangers consist of multiple parallel boiling channels where the channel-to-channel interactions significantly influence the density wave instability phenomenon.

PDOs are categorized as compound dynamic instability and occur as secondary phenomenon triggered by static instability. PDO occurs in a system having compressible volume upstream or within the heated section and when the system operates in the negative slope region of the pressure drop versus flow rate curve.

### 22.5.9 Start-Up Transients

Natural circulation systems may undergo thermal-hydraulic instabilities under low-power and low-pressure conditions, which occur during start-up. The void reactivity feedback and void fraction fluctuations in the reactor core would create power oscillations during start-up. Three kinds of thermal-hydraulic instabilities may occur during start-up in natural circulation BWRs, which are as follows: (1) geysering induced by condensation, (2) natural circulation instability induced by hydrostatic head fluctuation in steam separators, and (3) density wave instabilities.

During start-up conditions, inlet subcooling is the main parameter, which causes geysering. High inlet subcooling results in subcooled riser section and triggers geysering. At high inlet subcooling, geysering occurs at very low velocities. Flow reversal rate strongly influences the occurrence of geysering, but it is independent of inlet subcooling in both natural and forced circulation. It is observed that geysering is more pronounced in boiling channels with longer risers. The rate of flow reversals also increases as the riser length increases. Inlet throttling aggravates geysering as the natural circulation rate is reduced.

As the heat input is increased, geysering is suppressed and another instability called *natural circulation instability* is induced due to hydrostatic head fluctuation (caused by PDO), which varies the natural circulation force. As the heat flux is further increased, density wave instabilities appear. The period of natural circulation oscillations is much longer than that of density wave instabilities, and reduces with an increase in heat flux and with a decrease in inlet subcooling.

Flashing occurs when hot liquid flows from a higher to a lower pressure region, the saturation temperature decreases considerably, due to which a part of the liquid is converted into steam. This phenomenon is termed as flashing. At start-up conditions, both system pressure and heating power are low, which implies large differences in saturation

temperature between the inlet and the outlet of the adiabatic section. At low powers, the coolant, which is heated up in the heated section of the natural circulation loop, may not reach saturation conditions in the core itself. However, due to the strong variation of saturation temperature along the system, flashing can occur in the adiabatic section, leading to an enhancement of the natural circulation flow rate.

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## 22.6 Heat Transport with Phase Change

### 22.6.1 Introduction

Heat transport with phase change such as in boiling or condensation is an efficient method to transfer heat because latent heat per unit mass is very large compared to the sensible heat. For single component fluid, the interface temperature difference involved for heat transfer in evaporation and condensation is relatively small. However, when more than one component is present in a system the temperature difference can be higher. An example is condensation of vapors in the presence of noncondensable gases. The two-phase heat transfer relevant to reactors includes pool boiling, evaporation in a vertical channels, and condensation inside or outside the tubes.

### 22.6.2 Boiling Heat Transfer

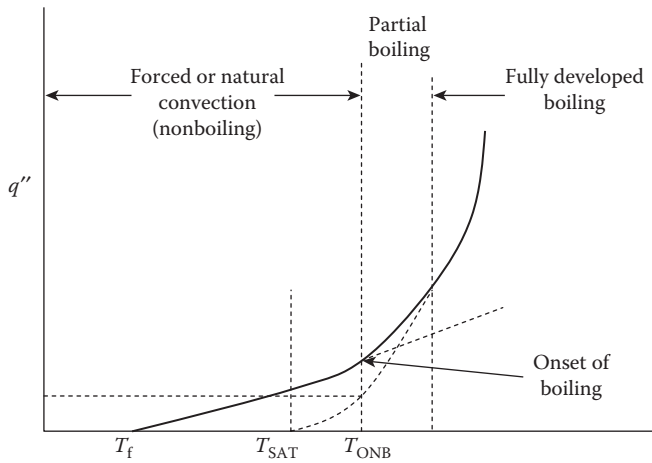
#### 22.6.2.1 Onset of Nucleate Boiling

Boiling or nucleate boiling is defined as the process of evaporation at a surface when the surface temperature exceeds the local saturation and the liquid immediately adjacent to the surface is at a superheated temperature, relative to its pressure. Based on the relative bulk motion of the body of a liquid to the heating surface, the boiling is divided into two categories: pool boiling and convective boiling. Pool boiling is the process in which the heating surface is submerged in a large body of stagnant liquid. The relative motion of the vapor produced and the surrounding liquid near the heating surface is due primarily to the buoyancy effect of the vapor. The vapor bubbles are generated at specific sites on the surface called nucleation site. In Figure 22.20 the flow regimes in boiling heat transfer and their relationship with wall temperature and heat flux are shown. The fluid temperature  $T_f$  increases with the heat flux  $q''$ , and initially the heat transfer to the fluid by natural or forced convection. The boiling does not start when the surface reaches saturation temperature  $T_{SAT}$ . The boiling starts when the surface temperature reaches a superheat with respect to the  $T_{SAT}$  for the onset of nucleate boiling  $T_{ONB}$ . At first, there are only few nucleation sites on the surface, but eventually a rapid increase in heat flux occurs with small rise in wall temperature. This region is referred to as fully developed boiling region which is preceded by partial boiling region.

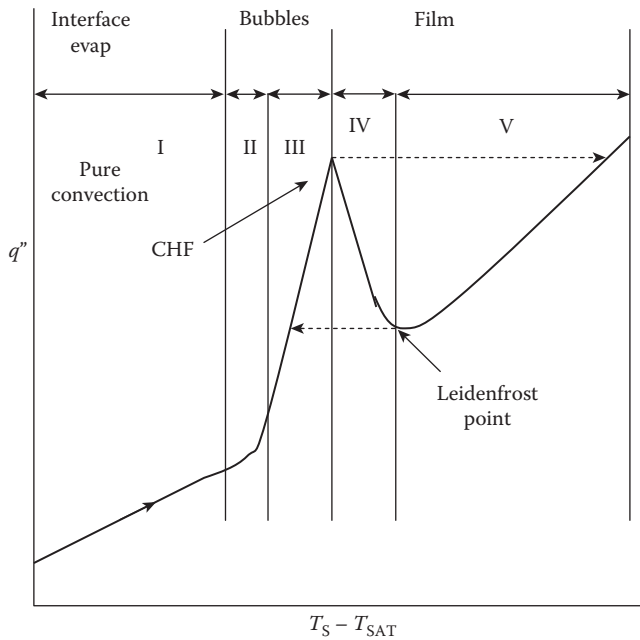
##### 22.6.2.1.1 Pool Boiling

In Figure 22.21 a pool boiling curve shown where the applied heat flux is plotted versus the temperature difference ( $\Delta T = T_s - T_{SAT}$ ) between the surface temperature and the liquid saturation temperature.

Five different boiling regimes can be observed in pool boiling: natural convection boiling, subcooled nucleate boiling, saturated nucleate boiling, and transition boiling and film boiling as shown in Figure 22.21. In region I,  $\Delta T$  is so small that no bubbles are observed and



**FIGURE 22.20**  
Regions of boiling heat transfer and wall temperature.



**FIGURE 22.21**  
Pool boiling curve.

the fluid motion in this mode of boiling is governed by natural convection currents. Vapor is produced by the evaporation of the liquid into gas nuclei on the exposed surface of the liquid. In region II,  $\Delta T$  is large enough that small isolated bubbles are produced along the heating surface but later condense in the region above the superheated liquid. In region III,  $\Delta T$  is enough to sustain *nucleate boiling*, with the creation of the bubbles such that they are continuously generated then depart and rise through the liquid forming columns of vapor. At large values of  $\Delta T$ , a large fraction of heater surface is covered with by bubbles making

it difficult for the liquid to reach the heater surface and wet it. The heat flux reaches a maximum called CHF  $q_{\max}$  at which the slope of the boiling curve changes sign with further increase in  $\Delta T$ . Further increase in  $\Delta T$  results in larger fraction of the heater surfaces covered by a vapor film. This region IV is called transition region or unstable film boiling where partial nucleate boiling and film boiling occur. For the region V called film boiling the heater surface is covered completely with stable vapor blanket. The minimum heat flux is called Leidenfrost point. In the film boiling region, the heat transfer is primarily governed by convection in the vapor blanket and radiation as heater surface temperature is high.

The regions described above are observed if the temperature of the heater or  $\Delta T$  is a controlling parameter. If the power or heat flux to the heater is the controlled variable then the increase in the power (or heat flux,  $q''$ ) in region III results in a jump in the surface temperature from critical heat flux point to a point in film boiling region V. This occurs due to hydrodynamic fluid instabilities. If the power is decreased from region V, the vapor film remains stable up to the minimum point, the Leidenfrost point. At this point, the vapor film becomes unstable and it collapses, with *nucleate boiling* becoming the mode of energy transfer. Thus, there is sudden jump from region V to III resulting in lower surface temperature. This *hysteresis* behavior is always seen when the power (or heat flux) is the controlled parameter.

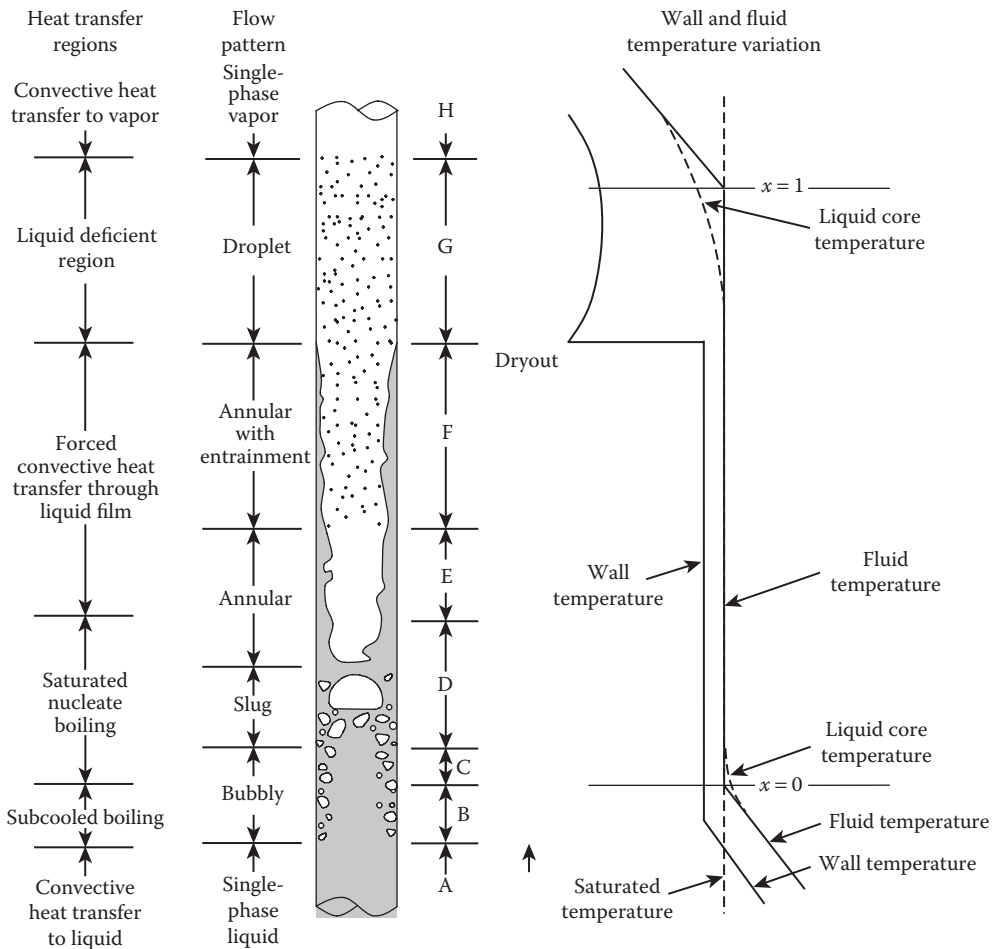
#### 22.6.2.1.2 Flow Boiling

Figure 22.22 illustrates various regimes of flow and heat transfer in a tube with vertical upflow forced convective boiling subjected to axially uniform heat. This figure also shows how the wall and fluid temperatures might vary along the tube. As the subcooled fluid enters the tube, the heat transfer regime changes along the length of the tube starting with single-phase convective heat transfer (A), then bubbly flow with subcooled boiling (B, C), and followed by saturated nucleate boiling with slug flow (D) and annular flow (E). The annular flow thins out as droplet entrainment increases (F) and is followed by droplet flow (G) where the tube wall is liquid deficient and is referred as dryout. This is then followed by single-phase vapor flow (H). The tube wall temperature increases suddenly at the dryout point (transition from F to G). The dryout is a mechanism for the occurrence of critical heat flux. Figure 22.23 shows the corresponding regimes of heat transfer A to H as they occur in terms of heat flux and quality. Higher heat flux has lower thermodynamic quality at which the boiling inception and the critical heat flux occur. For regions having enough liquid on the tube wall, high heat flux can form vapor blanket on the wall surface. This leads to departure from nucleate boiling (DNB) at the wall similar to the CHF condition in pool boiling leading to increase in wall temperature.

**22.6.2.1.2.1 Critical Heat Flux** In general, CHF occurs with high heat flux and is accompanied by large surface temperature rise.

CHF is also referred as boiling crisis and burnout. DNB is generally used to describe the CHF condition in pool boiling and also in flow boiling where rapid bubble formation causes vapor blanket on the heater surface. Thus, CHF at low quality is referred as DNB such as in PWR core. Dryout is used to describe liquid film dryout in annular flow. At high flow quality, dryout is expected with increase in heat flux such as in BWR core.

**22.6.2.1.2.2 Correlations for Boiling Heat Transfer** Nucleate boiling is a complex phenomenon. Several mechanisms have been proposed to explain the boiling process, such as latent heat transport, microconvection, vapor-liquid exchange, wake flow, enhanced convection, and microlayer evaporation, details of which can be found in Ginoux (1978).



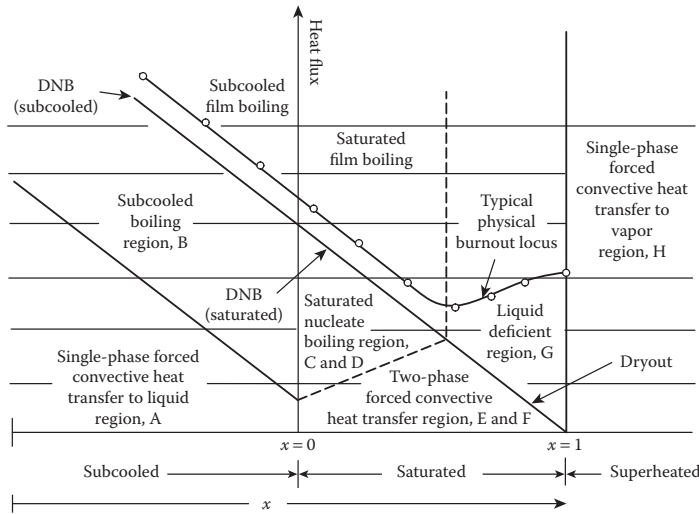
**FIGURE 22.22**

Regimes of flow and heat transfer in a tube with vertical upflow forced convective boiling subjected to axially uniform heat. (Data from Jones, O.C., Jr., *Nuclear Reactor Safety Heat Transfer*, Hemisphere Publishing Corp., New York, 1981.)

The heat transfer relations for various flow regimes of boiling heat transfer are obtained either by mechanistic model, empirical correlations, or by semi-theoretical correlations. Heat transfer relations for pool boiling heat transfer and for flow boiling heat transfer regimes are given in Tables 22.12 and 22.13, respectively. The CHF correlation by US reactor vendors and general CHF correlations are given in Tables 22.14 and 22.15, respectively.

### 22.6.3 Condensation Heat Transfer

Condensation occurs in many industrial applications including in nuclear reactor where condensation occurs in the suppression pool when steam is injected and in turbine condensers. Condensation is classified into the following classes: (1) surface filmwise condensation, where the vapor condenses in drops which grow by further condensation and



**FIGURE 22.23** Regimes of heat transfer corresponding to Figure 22.22 as function of heat flux and quality. (Data from Jones, O.C., Jr., *Nuclear Reactor Safety Heat Transfer*, Hemisphere Publishing Corp., New York, 1981.)

**TABLE 22.12**  
Correlations for Pool Boiling Heat Transfer

Regimes	Heat Transfer Relation	Remarks
Saturated nucleate boiling	Foster and Zuber (1955): $h_{NB} = \frac{0.00122\Delta T^{-0.24} \Delta p^{0.75} c_p^{0.45} \rho_f^{0.49} k_f^{0.79}}{\sigma^{0.5} h_{fg}^{0.24} \mu_f^{0.29} \rho_g^{0.24}},$ $q'' = h_{NB}\Delta T$	$\Delta p = p(T_w) - p(T_{SAT}), \Delta T = T_w - T_{SAT}$ this
	Thom et al. (1966): $\Delta T = \frac{22.7q''^{0.5}}{\exp(p/87)}, q'' = h_{NB}\Delta T$	For boiling water. $\Delta T$ in °C, $q''$ in MW/m <sup>2</sup> , $p$ system pressure (bars)
	Rohsenow (1952): $\frac{c_p \Delta T}{h_{fg}} = C_{SF} \left[ \frac{q''}{\mu_f h_{fg}} \left( \frac{\sigma}{g(\rho_f - \rho_g)} \right)^{1/2} \right] \left[ \frac{c_p \mu_f}{k_f} \right]^n$	$n = 1.0$ for water, $n = 1.7$ for other fluids, $C_{SF}$ is constant which depends on fluid surface combination: Water-SS 0.013 Water-platinum 0.020 Water-brass 0.006
Critical or maximum heat flux	Kutateladze (1948): $q''_{max} = C_1 h_{fg} \rho_g \left[ \frac{g\sigma(\rho_f - \rho_g)}{\rho_g^2} \right]^{1/4}$	Zuber (1958): $C_1 = \pi/24$ , Lienhard et al. (1973): $C_1 = 0.149$
Minimum heat flux	Zuber (1958): $q''_{max} = 0.09 h_{fg} \rho_g \left[ \frac{g\sigma(\rho_f - \rho_g)}{(\rho_f + \rho_g)^2} \right]^{1/4}$	For horizontal plate
Film boiling	Bromley (1950): $h_{conv} = C \left[ \frac{g(\rho_f - \rho_g) h_{fg}^3 k_g^3}{D v_g (\rho_f + \rho_g)^2} \right]^{1/4}$ $q'' = h_{conv} \Delta T$	$h'_{fg} = h_{fg} + 0.8c_{pg}\Delta T$ , $D$ is diameter of cylinder or sphere $C = 0.62$ for cylinder $C = 0.67$ for sphere



TABLE 22.13

Correlations for Flow Boiling Heat Transfer

Regime	Heat Transfer Relation	Remarks
Subcooled flow boiling-nucleation criteria	Bergles and Rohsenow (1964): $q_i'' = 15.6p^{1.156}(T_w - T_{SAT})_i^{2.3}/p^{0.023}$	For water at pressure between 0.1 to 13.6 MPa $p$ in psi, temperatures in °F, $q''$ in BTU/h ft <sup>2</sup>
Subcooled flow boiling-net vapor generation (NVP)	Saha and Zuber (1974): $T_{SAT} - T_{bulk} = 0.0022 \frac{q'' D_e}{k_f}$ for $Pe < 7 \times 10^4$ $T_{SAT} - T_{bulk} = 154 \frac{q''}{Gc_p}$ for $Pe > 7 \times 10^4$ $Pe = \frac{GD_e c_p}{k_f}$	For water, ranges: $p = 0.1$ – $13.8$ MPa $G = 95$ – $2760$ kg/m <sup>2</sup> s $q'' = 0.28$ – $1.89$ MW/m <sup>2</sup>
Saturated flow boiling	Chen (1963, 1966): $h_{2\phi} = h_{NB} + h_c$ $h_c = 0.023 \left( \frac{G(1-x)D_e}{\mu_f} \right)^{0.8} Pr_f^{0.4} \frac{k_f}{D_e} F$ $F = 1.0$ for $\frac{1}{X_{tt}} < 0.1$ $F = 2.35 \left( 0.213 + \frac{1}{X_{tt}} \right)^{0.736}$ for $\frac{1}{X_{tt}} > 0.1$ $h_{NB} = \frac{S(0.00122)\Delta T^{0.24} \Delta p^{0.75} c_p^{0.45} \rho_f^{0.49} k_f^{0.79}}{\sigma^{0.5} \mu_{fg}^{0.24} \mu_f^{0.29} \rho_g^{0.24}}$ $S = \frac{1}{1 + 2.53 \times 10^{-6} Re^{1.17}}$ , $Re = Re_f F^{1.25}$	For water, ranges: $P = 0.17$ – $3.5$ MPa Liquid inlet velocity = $0.06$ – $4.5$ m/s $q''$ up to $2.4$ MW/m <sup>2</sup> Correlation tested for methanol, pentane, cyclohexane, and benzene
Post CHF-boiling-liquid deficient	Groeneveld (1973): $\frac{hD_e}{k_g} = a \left\{ Re_g \left[ x + (\rho_g/\rho_f)(1-x) \right] \right\}^b Pr_g^c Y^d$ $Y = 1 - 0.1 \left( \frac{\rho_f}{\rho_g} - 1 \right)^{0.4} (1-x)$ , $Re_g = GD_e/\mu_g$ Tubes: $a = 1.09 \times 10^{-3}$ , $b = 0.989$ , $c = 1.41$ , $d = -1.15$ Annuli: $a = 5.20 \times 10^{-2}$ , $b = 0.688$ , $c = 1.26$ , $d = -1.06$	Steam water flow, range: Tubes: $D_e$ (mm) = $2.5$ – $25$ $P$ (MPa) = $6.8$ – $21.5$ $G$ (kg/m <sup>2</sup> s) = $700$ – $5300$ $q''$ (kW/m <sup>2</sup> ) = $120$ – $2100$ $x = 0.1$ – $0.9$ Annuli: $D_e$ (mm) = $1.5$ – $6.3$ $P$ (MPa) = $3.4$ – $10$ $G$ (kg/m <sup>2</sup> s) = $800$ – $4100$ $q''$ (kW/m <sup>2</sup> ) = $450$ – $2250$ $x = 0.1$ – $0.9$
Transition boiling	Tong (1972b): $h_{tb} = 39.75 e^{-0.0144\Delta T} + 2.3 \times 10^{-5} \frac{k_g}{D_e} e^{-105/\Delta T} Re_f^{0.8} Pr_f^{0.4}$	$h_{tb}$ (kW/m <sup>2</sup> K) = heat transfer coefficient for transition region
Film boiling	Tong (1972b): $h_{tb} = 39.75 e^{-0.0144\Delta T} + 2.3 \times 10^{-5} \frac{k_g}{D_e} e^{-105/\Delta T} Re_f^{0.8} Pr_f^{0.4}$	$h_{tb}$ (kW/m <sup>2</sup> K) = heat transfer coefficient for transition region

coalesce to form a continuous liquid film and usually flows downward by gravity; (2) surface dropwise, if the surface is nonwetting rivulets of condensate flow away and new drops then begin to form resulting in dropwise condensation; (3) homogeneous condensation, where vapor condenses out as droplets suspended in gas phase to form fog; and (4) direct contact condensation, which occurs when vapor is brought into contact with cold liquid.

TABLE 22.14

CHF Correlation by US Reactor Vendors

Regime	Correlation	Remarks
DNB for PWR	Westinghouse W-3, Tong (1967, 1972a) $q_{cr,n}'' = q_{cr}''/F, F = \frac{C \int_0^l q''(z') e^{-C(l-z')} dz'}{q''(l)[1 - e^{-Cl}]}$ $C = \frac{4.23 \times 10^6 [1 - x_e(l)]^{7.9}}{G^{1.72}} \text{ m}^{-1}$ $q_{cr}'' = [(2.022 - 0.06238p) + (0.1722 - 0.001427p) e^{18.177 - 0.5987p} x_e] [(0.1484 - 1.596x_e + 0.1729x_e  x_e ) 2.326G + 3271] [1.157 - 0.869x_e] [0.2664 + 0.837 e^{-124D_h}] [0.8258 + 0.0003413(h_f - h_{in})]$ $q_{cr,n}'' = q'' \text{ (kW/m}^2\text{) local at DNB position for axially non uniform heat flux, } l \text{ (m) = distance to DNB, } x_e \text{ = local steam thermodynamic quality}$	For circular, rectangle, and rod bundle geometry. Range: $p = 5.5\text{--}13.8 \text{ MPa}$ $G = 1350\text{--}6789 \text{ kg/m}^2 \text{ s}$ $D_h = 0.005\text{--}0.0178 \text{ m}$ $x_e = -0.15 \text{ to } 0.15$ $L = 0.254\text{--}3.658 \text{ m}$
Dryout for BWR	GE, Janssen-Levy limit (Janssen and Levy, 1962) For $p = 1000 \text{ psia}$ : $q_{cr}''/10^6 = 1.0$ for $x_e \leq (x_e)_1$ ; $q_{cr}''/10^6 = 1.9 - 3.3x_e - 0.7 \tanh^2(3G/10^6) \text{ Btu/h ft}^2$ for $(x_e)_1 < x_e < (x_e)_2$ ; and $q_{cr}''/10^6 = 0.6 - 0.7x_e - 0.09 \tanh^2(2G/10^6)$ for $x_e \geq (x_e)_2$ $(x_e)_1 = 0.273 - 0.212 \tanh^2(3G/10^6)$ $(x_e)_2 = 0.5 - 0.269 \tanh^2(3G/10^6) + 0.0346 \tanh^2(2G/10^6)$ Pressure correction for system pressure other than 1000 psia $q_{cr}''(p) = q_{cr}''(1000) \left[ 1.1 - 0.1 \left( \frac{p - 600}{400} \right)^{1.25} \right]$	For rod bundles Range: $G/10^6 \text{ (lb/h ft}^2\text{)} = 0.2\text{--}1.6$ $D_h \text{ (in.)} = 0.324\text{--}0.485$ $p \text{ (psia)} = 600\text{--}1450$ Rod spacing $> 0.06 \text{ in.}$

TABLE 22.15

General CHF Correlations

Gaspari et al. (1975)	$x_{crit} = \frac{D_h}{D_e} \left( a \frac{L_{crit}}{L_{crit} + b} \right), \text{ where } a = \frac{1 - p/p_c}{(G/1000)^{1/3}} \text{ if } G \geq G^*$ $\text{and } a = \frac{1}{1 + 1.481 \times 10^{-4} (1 - p/p_c)^{-3} G} \text{ if } G \leq G^*$ $b = 0.199(p/p_c - 1)^{0.4} GD^{1.4}$ $G^* = 3375(1 - p/p_c)^3, p_c = \text{critical pressure (MPa), } L_{crit} = \text{boiling length to CHF (m)}$	BWR rod bundle Range: $D = 0.0102\text{--}0.0198 \text{ m,}$ $L = 0.76\text{--}3.66 \text{ m,}$ $p = 4.96\text{--}6.89 \text{ MPa,}$ $G = 1085\text{--}4069 \text{ kg/m}^2 \text{ s}$
Biasi et al. (1967)	$q_{crit}'' = \frac{2.764 \times 10^7}{G^{1/6} (100D)^{-n}} \left[ \frac{1.468F(p_{bar})}{G^{1/6}} - x \right] \text{ (W/m}^2\text{) for low quality}$ $q_{crit}'' = \frac{15.048 \times 10^7 H(p_{bar})}{G^{0.6} (100D)^{-n}} [1 - x] \text{ (W/m}^2\text{) for high quality}$ $(G < 300 \text{ kg/m}^2 \text{ s)}$ $F(p_{bar}) = 0.7249 + 0.099p_{bar} e^{-0.032p_{bar}}$ $H(p_{bar}) = -1.159 + 0.149p_{bar} e^{-0.019p_{bar}} + \frac{9p_{bar}}{10 + p_{bar}^2}$ $n = 0.4 \text{ for } D \geq 9.91 \text{ m and } n = 0.6 \text{ for } D < 0.01 \text{ m}$	Round tube Range: $D = 0.0030\text{--}0.0375 \text{ m,}$ $L = 0.2\text{--}6.0 \text{ m,}$ $p = 0.27\text{--}14 \text{ MPa,}$ $G = 100\text{--}6000 \text{ kg/m}^2 \text{ s}$ $x = 1/(1 + \rho_l/\rho_g) \text{ to } 1$

(Continued)

TABLE 22.15 (Continued)

General CHF Correlations

Bowring (1972)	$q_{\text{crit}}^* = \frac{A - Bh_{fg}x}{C}, \quad A = \frac{A - Bh_{fg}x}{C}, \quad B = DG/4$ $C = \frac{0.077F_3DG}{1 + 0.347F_4(G/1356)^n}, \quad p_R = 0.145p$ <p>where <math>p</math> is MPa, <math>n = 2.0 - 0.5p_R</math>            For <math>p_R &lt; 1</math> MPa:  <math>F_1 = \{p_R^{18.942} \exp[20.89(1 - p_R)] + 0.917\} / 1.917</math>  <math>F_2 = F_1 / (\{p_R^{1.316} \exp[2.444(1 - p_R)] + 0.309\} / 1.309)</math>  <math>F_3 = \{p_R^{17.023} \exp[16.658(1 - p_R)] + 0.667\} / 1.667</math>  <math>F_4 = F_3 p_R^{1.649}</math>            For <math>p_R &gt; 1</math> MPa:  <math>F_1 = p_R^{-0.368} \exp[0.648(1 - p_R)]</math>  <math>F_2 = F_1 / \{p_R^{-0.448} \exp[0.245(1 - p_R)]\}</math>  <math>F_3 = p_R^{0.219}</math>  <math>F_4 = F_3 p_R^{1.649}</math></p>	PWR, BWR rod bundle Range: $D = 0.002 - 0.045$ m, $L = 0.15 - 3.7$ m, $p = 0.2 - 19.0$ MPa, $G = 136 - 18,600$ kg/m <sup>2</sup> s
Barnett (1966)	$q_{\text{crit}}^* = 3.1546 \times 10^6 \frac{3.458Ah_{fg} + 4.3 \times 10^{-4}B(h_f - h_i)}{C + 39.37L}$ <p>Where, <math>L =</math> heated length (m)  <math>A = 230.7D_h^{0.68}G^{0.192} [1 - 0.744 \exp(-0.3477D_eG)]</math>  <math>B = 01206D_h^{1.415}G^{0.817}, D = 8249D_e^{1.415}G^{0.212}</math>            For annuli wetted and heated equivalent diameters <math>D_e</math>            and <math>D_h</math> are: <math>D_e = (D_s - D_i)</math> and <math>D_h = (D_s^2 - D_i^2) / D_i</math>, where  <math>D_s =</math> diameter of the shroud and <math>D_i =</math> diameter of the inner rod</p>	Annuli, rod bundle, Range: $D = 0.0095 - 0.0960$ m, $D_s = 0.014 - 0.102$ m, $L = 0.61 - 2.74$ m, $p = 6.9$ MPa, $G = 190 - 8409$ kg/m <sup>2</sup> s

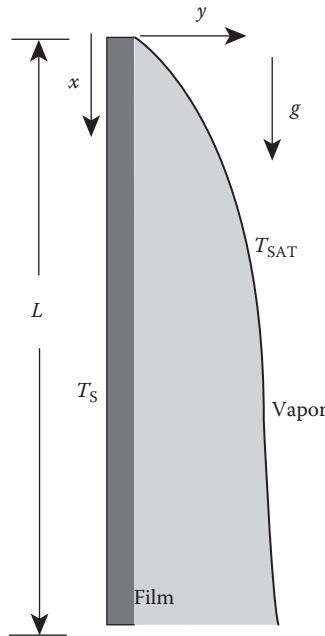
The dominant form of condensation is film condensation and most industrial systems employ this form of condensation. The local heat transfer coefficients for dropwise condensation are often an order of magnitude greater than those for filmwise condensation. However, it is difficult to maintain the surface to have dropwise condensation. Rates of heat transfer for film condensation can be predicted as a function of bulk and surface temperatures, total bulk pressure, surface and liquid film characteristics, bulk velocity, and the presence of noncondensable gases.

### 22.6.3.1 Film Condensation

Theoretical analysis of filmwise condensation of a stationary pure saturated vapor was originally presented by Nusselt (1916) for vertical surface (Figure 22.24). This analysis assumed laminar flow and constant properties for liquid film, no shear stress at the liquid-vapor interface, vapor at saturation temperature, and heat transfer through the film by conduction only.

The condensate mass flow rate per unit width  $\Gamma(x)$  is given by

$$\Gamma(x) = \frac{g\rho_f(\rho_f - \rho_g)\delta^3}{3\mu_f}. \quad (22.181)$$



**FIGURE 22.24**  
Laminar condensation over vertical surface.

The film condensate thickness at distance  $x$  is given as

$$\delta(x) = \left[ \frac{4k_f \mu_f (T_{SAT} - T_s) x}{g \rho_f (\rho_f - \rho_g) h_{fg}} \right]^{1/4}. \quad (22.182)$$

The average heat transfer coefficient for a vertical surface of height  $L$  was given by

$$\bar{h} = 0.943 \left[ \frac{g \rho_f (\rho_f - \rho_g) h_{fg} k_f^3}{\mu_f L (T_{SAT} - T_s)} \right]^{1/4}. \quad (22.183)$$

For horizontal tube,  $L$  is replaced by the tube diameter,  $D$ , and constant 0.943 becomes 0.725. An improvement to the Nusselt model was made by Rohsenow (1956) who considered the effects of subcooling within the liquid film and also allowed for a nonlinear distribution of temperature through the film due to energy convection. The latent heat of vaporization,  $h_{fg}$ , was replaced by a modified form  $h'_{fg} = h_{fg} + 0.68c_{pf}(T_{SAT} - T_s)$  in the above equation.

For film condensation inside horizontal tube, the Chato (1962) correlation is recommended for low vapor inlet Reynolds number ( $<35,000$ )

$$\bar{h} = 0.555 \left[ \frac{g \rho_f (\rho_f - \rho_g) h''_{fg} k_f^3}{\mu_f L (T_{SAT} - T_s)} \right]^{1/4}, \quad (22.184)$$

where the modified latent heat is

$$h''_{fg} = h_{fg} + 0.375c_{pf}(T_{SAT} - T_s). \quad (22.185)$$

For sufficient long vertical surface, the condensate film may become turbulent. The transition criterion may be expressed in terms of Reynolds number defined as

$$\text{Re}_\delta = \frac{4\Gamma}{\mu_f} \quad (22.186)$$

In terms of film thickness of  $\delta$ , the Reynolds number is

$$\text{Re}_\delta = \frac{4g\rho_f(\rho_f - \rho_g)\delta^3}{3\mu_f^2} \quad (22.187)$$

Three flow regimes and the corresponding modified Nusselt number in terms of the  $\text{Re}_\delta$  are given as,

Wave free laminar region  $\text{Re}_\delta \leq 30$ ,

$$\frac{\bar{h}(v_f^2/g)^{1/3}}{k_f} = 1.47 \text{Re}_\delta^{-1/3} \quad (22.188)$$

Laminar wavy region  $30 \leq \text{Re}_\delta \leq 1800$  (Kutateladze, 1963),

$$\frac{\bar{h}(v_f^2/g)^{1/3}}{k_f} = \frac{1.47 \text{Re}_\delta}{1.08 \text{Re}_\delta^{1.22} - 5.2} \quad (22.189)$$

Turbulent region  $\text{Re}_\delta \geq 1800$  (Labuntsov, 1957),

$$\frac{\bar{h}(v_f^2/g)^{1/3}}{k_f} = \frac{\text{Re}_\delta}{8750 + 58 \text{Pr}^{-0.5} (\text{Re}_\delta^{0.75} - 253)} \quad (22.190)$$

### 22.6.3.2 Effect of Noncondensable Gas

Presence of even a small amount of noncondensable gas in the condensing vapor leads to a significant reduction in heat transfer during condensation. The buildup of noncondensable gases near the film vapor interface inhibits the diffusion of vapor from the bulk mixture to the liquid film. The net effect is to reduce the effective driving force for heat and mass transfer. Figure 22.25 shows the temperature and noncondensable gas boundary layer near the film and vapor interface.

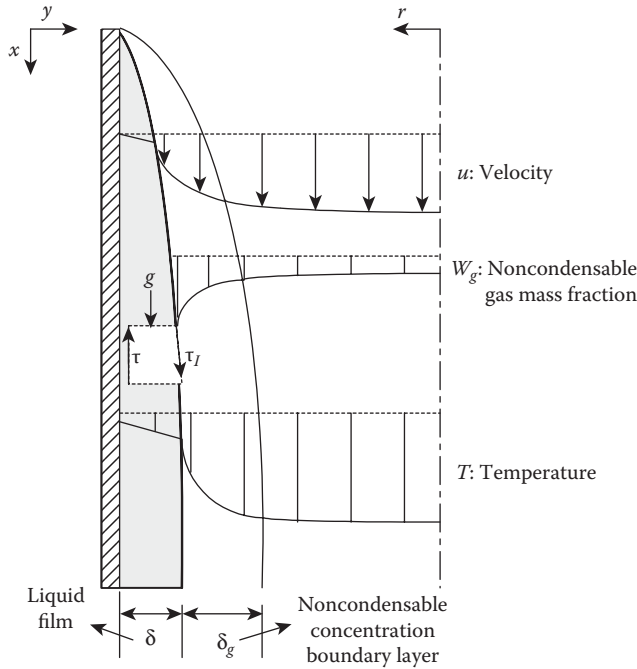
For the condensation of stagnant vapor/noncondensable gas mixture on the vertical surface, the following correlations are available.

The correlation developed by Uchida et al. (1965)

$$h_{\text{tot}} \cong 380 \left( \frac{W_{\text{nc}}}{1 - W_{\text{nc}}} \right)^{-0.7} \quad (\text{W/m}^2 \text{K}) \quad (22.191)$$

from the correlations for the long sections of structural walls. The Tagami correlation (1965)

$$h_{\text{tot}} = 11.4 - 284 \left( \frac{W_{\text{nc}}}{1 - W_{\text{nc}}} \right) \quad (\text{W/m}^2 \text{K}) \quad (22.192)$$



**FIGURE 22.25**  
Boundary layers due to the presence of noncondensable in vapor condensation on a plate or inside a tube.

where  $W_{nc}$  is defined as the mass fraction of noncondensable gas. The geometrical aspects and the effect of velocity field were ignored. Therefore, caution should be used to extrapolate results.

The Dehbi correlation (1991)

$$h_{tot} = \frac{L^{0.05} \left[ (3.7 + 28.7 P_{tot}) - (2438 + 458.3 P_{tot}) \log_{10} W_{nc} \right]}{(T_b - T_w)^{0.25}} \text{ (W/m}^2 \text{ K)} \quad (22.193)$$

This correlation is applicable for the following conditions:

$$0.3 \text{ m} < L < 3.5 \text{ m}, \quad 1.5 \text{ atm.} < P_{tot} < 4.5 \text{ atm.}, \quad \text{and} \quad 10^\circ\text{C} < T_b - T_w < 50^\circ\text{C}$$

There are two types of correlations for estimating the heat transfer coefficient for condensation inside vertical tube. In the first type of correlations, the local heat transfer coefficient is expressed in the form of a degradation factor defined as the ratio of the experimental heat transfer coefficient (when noncondensable gas is present) and pure steam heat transfer coefficient (Kuhn et al. [1997]). The correlations in general are the functions of local noncondensable gas mass fraction and mixture Reynolds number (or condensate Reynolds number). In the other type of correlations, the local heat transfer coefficient is expressed in the form of dimensionless numbers. In these correlations, local Nusselt number is expressed as a function of mixed Reynolds number, Jakob number, noncondensable gas mass fraction, condensate Reynolds number, and so on.

The degradation factor is defined as

$$f = \frac{h_{\text{tot}}}{h_{\text{film}}}, \quad (22.194)$$

where  $f$  may be the function of film Reynolds number ( $Re_f$ ), vapor/noncondensable gas mixture ( $Re_g$ ), noncondensable gas mass fraction ( $W_a$ ), Jakob number ( $Ja_g$ ), and Prandtl number ( $Pr_g$ ).

For air

$$f = f_1 f_2 \quad (22.195)$$

$$f_1 = f_{1\text{shear}} \left( 1 + 7.32 \times 10^{-4} Re_f \right) \quad (22.196)$$

$$f_2 = \left( 1 - 2.601 W_a^{0.708} \right) \quad \text{for } M_a < 0.1 \quad (22.197)$$

$$f_2 = \left( 1 - W_a^{0.292} \right) \quad \text{for } M_a > 0.1. \quad (22.198)$$

where:

$W_a$  is the air mass fraction

$Re_f$  is the film Reynolds number

For helium

$$f_1 = f_{1\text{shear}} \left( 1 + 7.32 \times 10^{-4} Re_f \right) \quad (22.199)$$

$$f_2 = \left( 1 - 35.81 W_{\text{He}}^{1.04} \right) \quad \text{for } 0.003 < W_{\text{He}} < 0.01 \quad (22.200)$$

$$f_2 = \left( 1 - 2.09 W_{\text{He}}^{0.457} \right) \quad \text{for } 0.01 < W_{\text{He}} < 0.1 \quad (22.201)$$

$$f_2 = \left( 1 - M_{\text{He}}^{0.137} \right) \quad \text{for } 0.01 < W_{\text{He}} < 0.1 \quad (22.202)$$

The parameter  $f_{1\text{shear}}$  is the ratio of liquid film thickness with interfacial shear to film thickness without interfacial shear.

The correlations for tube condensation heat transfer in presence of noncondensable gas by Revankar and Oh (2006) are given as

$$Nu_c = \left\{ \frac{1}{1.2} \frac{\delta k_g}{d k_f} + \frac{1}{160 W_{\text{nc}}^{-0.9} \exp(-42.5 Ja_g) + 4.364} \right\}^{-1} \quad \text{for laminar flow} \quad (22.203)$$

$$Nu_c = \left\{ \frac{1}{1.2} \frac{\delta k_g}{d k_f} + \frac{1}{0.08 W_{\text{nc}}^{-0.9} Re_g^{1.1} \exp(-42.5 Ja_g) + 0.021 Re_g^{0.8} Pr_g^{0.5}} \right\}^{-1} \quad \text{for turbulent flow} \quad (22.204)$$

These correlations are valid for the ranges:  $W_{\text{nc}}$  (0.01–0.8),  $Re_g$  (20–45,000),  $Ja_g$  (0.002–0.057).

## 22.7 Core Thermal–Hydraulics Analysis

### 22.7.1 Introduction

The amount of reactor power generation in the reactor core is determined by core thermal-hydraulics. The reactor core is operated such that the core heat is adequately removed to maintain the prescribed limits on the temperatures of fuel and cladding material at all locations of the core. The thermal-hydraulics limit for the core designs is usually the maximum cladding temperature. The temperature limits are determined by the maximum allowable heat fluxes at the fuel elements and the coolant interface. The heat flux from the fuel element is in turn limited by the fuel burnup and is also based on economics and refueling considerations. Generally, the core thermal-hydraulics analysis involves the following steps: (1) determine the core heat sources and their time and spatial distribution, (2) determine temperature distribution in the fuel elements using heat conduction analysis, (3) determine fuel element surface temperature and heat transfer rate between coolant and the fuel element using convection and/or boiling heat transfer analysis, (4) analyze the heat transfer to the out-of-core components such as steam generator, secondary coolants. In this section, general heat transfer between the fuel and the coolant is first discussed. The temperature distribution, the maximum fuel temperature, and hydrodynamic considerations in single-phase coolant core and boiling core are then discussed.

### 22.7.2 Axial Temperature Distribution in Single Phase

Here one-dimensional core analysis is considered. An axial variation in the neutron flux and hence the heat flux in the fuel element is considered.

The assumptions for the heat balance along the channel are (1) steady-state flow; (2) constant properties; (3) closed channel analysis, no cross-flow between channels; (4) channel flow area  $A_c$  is constant in the axial direction; (5) no boiling in the coolant, single-phase flow; (6) constant heat transfer coefficient between the coolant and the heated surface; (7) the channel is vertical channel; and (8) the volumetric heat generation rate is independent of radial position in the fuel pellet.

Figure 22.26 shows the axial heat source distribution in the reactor core where a chopped cosine distribution for heat source is assumed. The extrapolated height of the core is  $L_e$  and the core center place is at  $z = 0$ .

If the volumetric heat thermal source strength at  $z = 0$  is  $q_0'''$ , then volumetric heat thermal source strength  $q'''$  at any point  $z$  is given as

$$q'' = q_0'' \cos \frac{\pi z}{L_e} \quad (22.205)$$

From Figure 22.27, the sensible heat gained by the coolant stream from the fuel element over a differential length of  $dz$  given through the energy balance is given as,

$$mc_p dT_m = q'' A_c dz \quad (22.206)$$

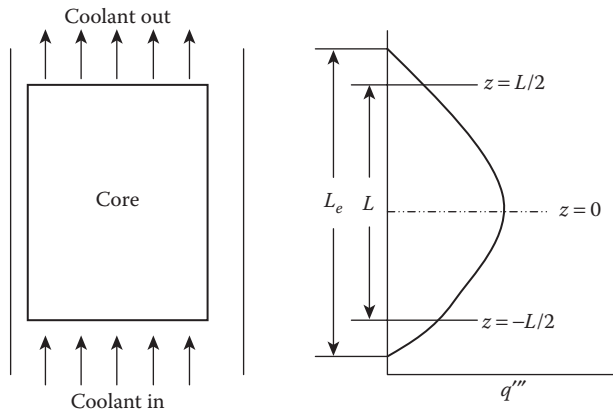
where:

$m$  is the mass flow rate of coolant fluid per fuel element

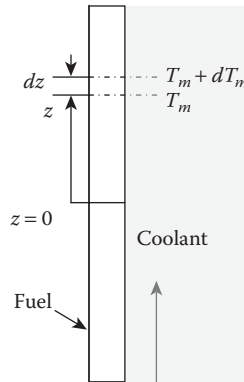
$c_p$  is the specific heat of coolant fluid

$dT_m$  is the coolant fluid temperature rise between





**FIGURE 22.26** Axial heat distribution in a reactor core.  $L$ , active core height;  $L_e$ , extrapolated height.



**FIGURE 22.27** Heat balance between fuel element and coolant.

By using the above two equations and integrating between  $z = 0$ – $z$ , the temperature of the coolant at any location  $z$  is given as

$$T_m(z) = T_{in} + \frac{q_0'' A_c L_e}{\pi m c_p} \left( \sin \frac{\pi z}{L_e} + \sin \frac{\pi L}{2L_e} \right), \tag{22.207}$$

where  $T_{in}$  is the inlet coolant temperature. The coolant outlet temperature  $T_{out}$  is obtained by substituting  $z = L/2$  in the above equation.

$$T_m(z) = T_{in} + \frac{2q_0'' A_c L_e}{\pi m c_p} \sin \frac{\pi L}{2L_e}, \tag{22.208}$$

If  $L_e \approx L$  then the total temperature rise in coolant temperature for this specific channel is given as

$$T_m(z) - T_{in} = + \frac{2q_0'' A_c L_e}{\pi m c_p}, \tag{22.209}$$

As the coolant temperature varies along the channel, the outside cladding temperature will also vary. A balance for the heat transfer from the fuel rod to the coolant gives

$$q'''(z)A_c dz = 2\pi R_{co} dz h [T_{co}(z) - T_m(z)], \quad (22.210)$$

where:

$R_{co}$  is the clad outer radius

$h$  is the convective heat transfer coefficient between coolant and clad surface

$T_{co}$  is the cladding temperature

Using expression for  $q'''$  and for  $T_m$ , the cladding temperature is given as

$$T_{co}(z) = T_{in} + q_0''' A_c \left[ \frac{L_e}{\pi m c_p} \left( \sin \frac{\pi z}{L_e} + \sin \frac{\pi L}{2L_e} \right) + \frac{1}{2\pi R_{co} h} \cos \frac{\pi z}{L_e} \right]. \quad (22.211)$$

The location of maximum clad surface temperature,  $z_c$ , can be determined by the condition

$$\frac{dT_{co}(z)}{dz} = 0, \quad (22.212)$$

and is given as,

$$z_c = \frac{L_e}{\pi} \tan^{-1} \frac{2R_{co} L_e h}{m c_p}. \quad (22.213)$$

Inside cladding temperature can be derived from an energy balance as

$$q'''(z)A_c dz = \frac{2\pi dz k_c [T_{co}(z) - T_{ci}(z)]}{\ln(R_{co}/R_{ci})}, \quad (22.214)$$

where:

$k_c$  is the cladding thermal conductivity

Subscript ci refers to clad inside surface

Similar energy balance equation can be written for heat transfer between fuel element center and the cladding using the gap conductance model and the fuel radial conductivity. Using these expressions, the fuel central line temperature is obtained as

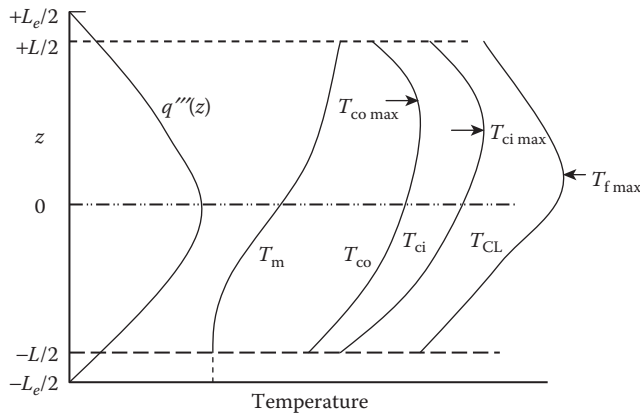
$$T_{CL}(z) = T_{in} + q_0''' A_c \left[ \frac{L_e}{\pi m c_p} \left( \sin \frac{\pi z}{L_e} + \sin \frac{\pi L}{2L_e} \right) + \left\{ \frac{1}{2\pi R_{co} h} + \frac{1}{2\pi k_c} \ln \left( \frac{R_{co}}{R_{ci}} \right) + \frac{1}{2\pi R_g h_g} + \frac{1}{2\pi k_f} \right\} \cos \frac{\pi z}{L_e} \right] \quad (22.215)$$

where:

$k_f$  is the thermal conductivity of fuel

$h_g$  is the gap conductance

The position of maximum fuel temperature is obtained by differentiating the above equation to find the maximum and is given as



**FIGURE 22.28**  
Axial variation of coolant, cladding surface, and fuel element center temperatures.

$$z_f = \frac{L_e}{\pi} \tan^{-1} \left[ L_e \left\{ \frac{1}{2\pi R_{co} h} + \frac{1}{2\pi k_c} \ln \left( \frac{R_{co}}{R_{ci}} \right) + \frac{1}{2\pi R_g h_g} + \frac{1}{2\pi k_f} \right\}^{-1} \right]. \quad (22.216)$$

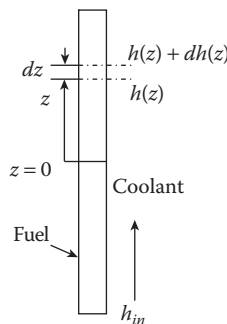
In Figure 22.28, the axial distribution of coolant, cladding surface, and fuel element temperatures are illustrated. Maximum temperatures for coolant, clad, and the fuel occur above the mid plane. The effect of the coolant is to shift upward the radial temperatures along the rod.

### 22.7.3 Two-Phase Heat Transfer

If the coolant undergoes a phase change (boiling), a different approach must be used since the design limit is clad heat flux, not clad temperature. The CHF is used as the limit to prevent film boiling and fuel rod failure. Figure 22.29 shows the heat balance.

Since a phase change can occur, the coolant enthalpy is used instead of the temperature. The coolant energy balance becomes

$$m dh(z) = q'' A_c dz. \quad (22.217)$$



**FIGURE 22.29**  
Enthalpy balance for coolant heat transfer with phase change.

Using the sine law power distribution the axial enthalpy can be written as

$$h(z) - h_{\text{in}} = \frac{q''' A}{m c} \int_{-L_e/2}^z \cos\left(\frac{\pi z}{L_e}\right) dz. \quad (22.218)$$

By integrating this equation the axial enthalpy is

$$h(z) = h_{\text{in}} + \frac{q_0''' A_c L_e}{\pi m} \left( \sin \frac{\pi z}{L_e} + \sin \frac{\pi L}{2L_e} \right). \quad (22.219)$$

Note here that the heat generation rate per channel,  $\dot{q} = q_0''' A_c L_e / \pi$ . The net heat energy transfer per unit mass from inlet to outlet of the channel is given as  $h_{\text{out}} - h_{\text{in}} = \dot{q} / m$ .

For liquid entering in subcooled condition, the liquid is heated along the channel and will reach saturation condition. When the cladding temperature increases above the coolant saturation temperature and has sufficient superheat, boiling occurs. Boiling is a constant temperature process; once boiling begins the cladding temperature remains nearly constant. The enthalpy at distance  $Z_b$  where boiling begins is the saturation liquid enthalpy, that is,  $h(Z_b) = h_f$ . Thus, we have

$$h_f = h_{\text{in}} + \frac{\dot{q}}{2m} \left( \sin \frac{\pi Z_b}{L_e} + \sin \frac{\pi L}{2L_e} \right) = h_{\text{in}} + \frac{(h_{\text{out}} - h_{\text{in}})}{2} \left( \sin \frac{\pi Z_b}{L_e} + \sin \frac{\pi L}{2L_e} \right). \quad (22.220)$$

This can be solved for  $Z_b$  as

$$Z_b = \frac{L}{\pi} \sin^{-1} \left[ 2 \left( \frac{h_f - h_{\text{in}}}{h_{\text{out}} - h_{\text{in}}} \right) - \sin \frac{\pi L}{2L_e} \right]. \quad (22.221)$$

In the two-phase section, the fluid enthalpy is related to the two-phase mixture quality  $x$  as

$$h(z) = h_f + x_e(z) h_{\text{fg}}. \quad (22.222)$$

The local mixture quality can be calculated along the channel as

$$x_e(z) = \frac{h(z) - h_f}{h_{\text{fg}}} = \frac{\dot{q}}{2m h_{\text{fg}}} \left( \sin \frac{\pi z}{L_e} + \sin \frac{\pi L}{2L_e} \right) + \frac{h_{\text{in}} - h_f}{h_{\text{fg}}} \quad (22.223)$$

The quality can be negative (subcooled flow) or positive, when vapor exists and the two phases are saturated. The inlet quality is given as

$$x_{e\text{in}} = \frac{h_{\text{in}} - h_f}{h_{\text{fg}}}. \quad (22.224)$$

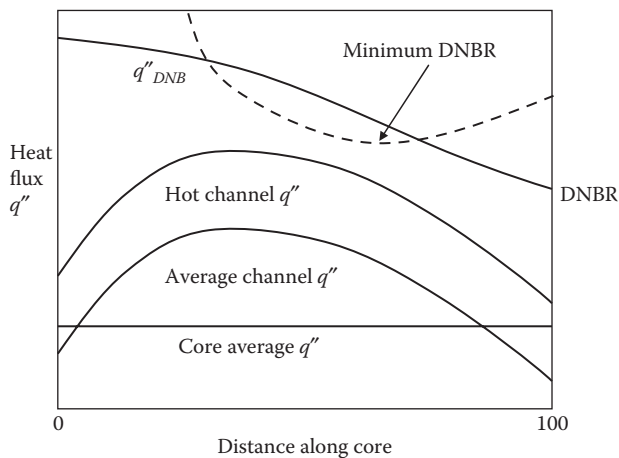
Thus, the quality in the channel is given as

$$x_e = \frac{\dot{q}}{2m h_{\text{fg}}} \left( \sin \frac{\pi z}{L_e} + \sin \frac{\pi L}{2L_e} \right) + x_{e\text{in}}. \quad (22.225)$$

### 22.7.4 Hot Channel and Burnout

The reactor core has several thousand parallel coolant channels, and heat removal from each channel may be slightly different from the other channel. This variation can occur due to design, flow dynamics, and neutron dynamics. The deviations in channel temperature are characterized by defining the *hot channel* as the flow channel which has the maximum enthalpy rise and *hot spot* as the location in the core where the heat flux is maximum. The uncertainties may produce underprediction or an overprediction of temperatures and design and operation constraints within fuel elements. The underprediction of fuel temperature may have serious consequences. In order to protect against underpredicting fuel element design limits, hot channel and hot spot factors are defined. These factors are obtained by estimating for each physical variation, a factor representing maximum change in temperature rise from nominal value. These various factors are combined to find multipliers that when applied to the nominal temperature give more realistic maximum temperature in the core.

The most important phenomenon in the LWR is the burnout which limits the reactor power level. The burnout occurs due to DNB. As an example in Figure 22.30 the heat fluxes along the core length for core average and the average channel for hot channel and for the DNB are shown. The total heat generated in an average channel is the total reactor heat divided by number of channels in the core. The core average heat flux is the average of the channel average heat flux. The hottest channel line represents the maximum temperature in the core. This represents the maximum heat flux permissible in the core. The calculated DNB heat flux curve is the locus of computed values of the critical heat flux. The ratio between the DNB heat flux predicted by application correlation and the local reactor hot channel heat flux is called the DNB ratio DNBR. The minimum DNBR occurs downstream from the position of maximum heat flux because of the effect of increasing fluid enthalpy on decreasing  $q''_{DNB}$ . The design basis for the water cooled reactors is that the minimum DNBR is 1.3. This is the extra factor for safety associated with CHF during the design overpower condition and anticipated transients.



**FIGURE 22.30**

Axial distribution of heat flux for core, channel, and the DNB, minimum DNBR.

### 22.7.5 Thermal-Hydraulics Codes

The complexity of the heat transfer processes in nuclear reactor core requires computer codes to handle both local and system-wide behavior under normal, transient, and accident conditions. The code models are assessed with experimental data to ensure that they were working properly. Some of the largest and most widely used codes in United States are as follows:

- RELAP5 (Reactor Excursion and Leak Analysis Program) is an LWR transient analysis code developed for use in rulemaking, licensing audit calculations, evaluation of operator guidelines, and as a basis for a nuclear plant analyzer. It is a generic PWR code capable of modeling large- and small-break LOCAs, operational transients, and transients in which the entire balance-of-plant must be modeled. This code was developed at Idaho National Engineering Laboratories.
- TRAC (Transient Reactor Analysis Code)-PWR is an advanced best-estimate systems code designed primarily for analysis of large-break LOCAs (LB-LOCAs) in PWRs, although its versatility allows for the analysis of a wide range of scenarios. This code was developed at Los Alamos National Laboratory under sponsorship of the US Nuclear Regulatory Commission (USNRC).
- TRAC-BWR is an advanced best-estimate systems code designed primarily for analysis of LB-LOCAs in BWRs, although its versatility allows for the analysis of a wide range of scenarios. This code was developed at Idaho National Engineering Laboratories under sponsorship of the USNRC.
- TRACE (TRAC/RELAP Advanced Computational Engine) is recent thermal-hydraulics code designed to consolidate and extend the capabilities of safety codes—TRAC-P, TRAC-B, and RELAP. It is intended for analysis of large- and small-break LOCAs and system transients in both PWRs and BWRs. The code has the capability to model thermal-hydraulic phenomena in both 1D and 3D spaces.
- RAMONA was developed at Brookhaven National Laboratory for analyzing BWR system transients. Until recently, RAMONA was the only best-estimate BWR system transient code capable of predicting 3D power in the core coupled with the fuel and cladding temperature and vessel thermal-hydraulic phenomena.
- RETRAN (REactor TRansient ANalysis) is a best-estimate transient thermal-hydraulic analysis computer program (sponsored by EPRI) designed to provide analysis capabilities for BWR and PWR transients, small-break LOCAs, balance-of-plant modeling, and anticipated transients without scram (ATWS).
- SNAP (Symbolic Nuclear Analysis Package) is a graphical user interface with pre-processor and post-processor capabilities that assists code users in the development of TRACE and RELAP5 input decks and in running the codes.
- CONTAIN (Containment transient analysis tool for PWRs or BWRs) has the capability to model thermal-hydraulics phenomena for existing containment designs.

Severe accident codes are used to model the progression of accidents in LWR NPPs.

- MELCOR (Integral Severe Accident Analysis Code) Fast-Running, parametric models.
- SCDAP/RELAP5 (Integral Severe Accident Analysis Code) uses detailed mechanistic models.

- CONTAIN (Integral Containment Analysis Code) uses detailed mechanistic models. The MELCOR code has similar containment capabilities (but less detailed in some areas) and should generally be used instead of CONTAIN.
- IFCI (Integral Fuel-Coolant Interactions Code).
- VICTORIA (Radionuclide Transport and Decommissioning Codes) is suit of radionuclide transport and decommissioning codes providing dose analyses in support of license termination and decommissioning.

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## 22.8 Safety Analysis

### 22.8.1 Risk from NPP

The safety of NPP is of paramount concern both for public and regulators. The key reasons for these concerns are that the nuclear reactor core is potential source of large amount of radioactive material, and in case of accident the radioactive material can leak into atmosphere and contaminate the environment with large public consequences. In Table 22.16 the major radioactivity contributing fission product and higher actinides are listed for typical LWRs spent fuel after 150 days cooling period. Thus reactor core has large source of radioactive material; nuclear safety has three main objectives: (1) to ensure that NPP operate normally as designed and without operating staff and the environment being exposed to radiation from the NPP radioactive materials, (2) to prevent accidents or events from happening, and (3) to minimize and limit the consequences of any incidents that might occur.

So, safety study or analysis is very important in the design and operation of NPP. The purpose of the safety study is to assess the risks to the public from potential accidents in NPP. The safety analysis involves study of accidents or events that might lead to release of a significant amount of radioactive material. The risk from any event is expressed as

$$\begin{aligned} \text{Risk}(\text{consequence}/\text{time}) &= \text{Frequency}(\text{event}/\text{time}) \\ &\times \text{Magnitude}(\text{consequence}/\text{event}). \end{aligned} \quad (22.226)$$

To estimate risk from NPP accident or event, probabilities and consequences from each event have to be evaluated. In Figure 22.31 shows the generic flowchart highlighting main components in risk assessment from an NPP. The risks from NPP are estimated by methodology called *probabilistic risk analysis* (PRA) (Lee and McCormick, 2011) which will be introduced later.

There are two basic types of safety analysis: (1) deterministic safety analysis and (2) probabilistic safety analysis (PSA). The deterministic safety analysis approach involves evaluations of physical phenomena occurring at NPPs to demonstrate that safety requirements, such as the requirement for ensuring the integrity of barriers against the release of radioactive material and various other acceptance criteria, are met for all initiating events over a broad range of operational states, including different levels of availability of the safety systems. Deterministic safety analysis for an NPP predicts the response to postulated initiating events (PIEs). A specific set of rules and acceptance criteria is applied and they focus on neutronic, thermo-hydraulic, radiological, thermomechanical, and structural aspects that are often analyzed with different computational codes such as one listed in Section 22.7. These analyses provide the radioactive source terms for risk estimation.

TABLE 22.16

Major Radioactivity Contributors in Spent Light Water Reactor Fuel after 150 Days Cooling

Nuclide	Half-Life Years	Main Decay Mode	Activity	
			Ci/1000 kg U	Bq/1000 kg U
<i>Fission Product</i>				
Strontium-89	0.14	$\beta$	$9.6 \times 10^4$	$3.6 \times 10^{15}$
Strontium-90	29	$\beta$	$7.7 \times 10^4$	$2.8 \times 10^{15}$
Zirconium-95	0.18	$\beta, \gamma$	$2.8 \times 10^5$	$1.0 \times 10^{16}$
Niobium-95	0.095	$\beta, \gamma$	$5.2 \times 10^5$	$1.9 \times 10^{16}$
Ruthenium-106	1.0	$\beta$	$4.1 \times 10^5$	$1.5 \times 10^{16}$
Cesium-134	2.05	$\beta, \gamma$	$2.1 \times 10^5$	$7.7 \times 10^{15}$
Cesium-137	30	$\beta, \gamma$	$1.1 \times 10^5$	$4.1 \times 10^{15}$
Cerium-144	0.78	$\beta, \gamma$	$7.7 \times 10^5$	$2.8 \times 10^{16}$
Promethium-147	2.6	$\beta$	$9.9 \times 10^4$	$3.7 \times 10^{15}$
Technetium-99	211,000	$\beta$	14	$5.9 \times 10^{11}$
Iodine-129	$15.7 \times 10^6$	$\beta$	0.033	$1.3 \times 10^9$
Iodine-131	0.022	$\beta$	0.19	$7.9 \times 10^9$
Yttrium-91	0.160	$\beta$	$1.5 \times 10^5$	$6.2 \times 10^{15}$
<i>Heavy Isotopes</i>				
Plutonium-238	88	$\alpha$	$2.8 \times 10^2$	$1.0 \times 10^{13}$
Plutonium-239	24,400	$\alpha$	$3.3 \times 10^2$	$1.2 \times 10^{13}$
Plutonium-240	6,540	$\alpha$	$4.8 \times 10^2$	$1.8 \times 10^{13}$
Plutonium-241	14	$\beta$	$1.1 \times 10^5$	$4.1 \times 10^{15}$
Plutonium-242	387,000	$\alpha$	1.36	$5.0 \times 10^{10}$
Americium-241	433	$\alpha, \gamma$	$2.0 \times 10^2$	$7.4 \times 10^{12}$
Americium-243	7,370	$\alpha, \gamma$	17.4	$6.4 \times 10^{11}$
Curium-242	0.45	$\alpha, sf$	$1.5 \times 10^4$	$5.5 \times 10^{14}$
Curium-244	18	$\alpha, sf$	$2.5 \times 10^3$	$9.3 \times 10^{13}$

Source: Glasstone, S. and Sesonske, A., *Nuclear Reactor Engineering: Reactor Systems Engineering*, 4th edn., Vol. 2, Springer Science+Business Media, Dordrecht, the Netherlands, 1994, p. 642. With permission.  
 sf, spontaneous fission accompanied by neutron emission.

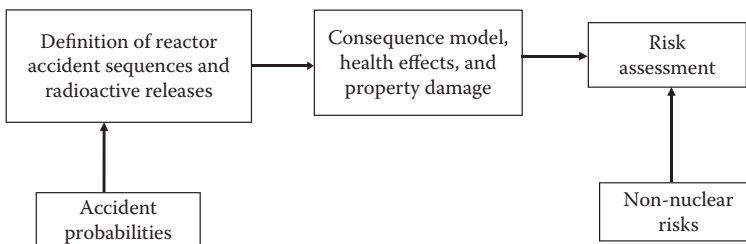


FIGURE 22.31  
 Reactor risk assessment study flowchart.



As designed, the radioactive material in the reactor is safely contained in the core in fuel assemblies during normal operation. The key aspect of maintaining this integrity of the core is having proper cooling of the fuel assembly. Bulk of the activity (98%) is within the fuel as long as the fuel is adequately cooled. Thus, major objectives of nuclear reactor operation and safety are to provide adequate heat removal and control of energy release to prevent overheating and in most severe case, melting. During normal operation, the core is cooled with adequate flow of water through forced convection cooling or flow boiling process. However, if the reactor is shut down still decay heat is generated, and there is large amount of stored energy in the core, coolant, and structural support material which adds substantial heat energy. There are also chemical reactions that can happen during an accident that add energy to the reactor vessel or containment. Some of these reactions include the exothermic reaction of cladding material zirconium alloy and structural steel reaction with steam if they exceed ignition temperature for oxidation. If there is failure to remove this heat, redistribution of this energy and resulting temperature profile in fuel, clad, and coolant may lead to damage of the fuel resulting in radioactive release from core. Thus, safety analysis involves detailed study of the reactor heat removal using various methods, reactor safety systems, consequences of failure of the cooling of the core and risk from any event or transient in a reactor.

The safety matters in nuclear facilities are covered by governmental rules and guides and by a large body of industrial codes and standards. The USNRC has the statutory responsibility of reactor safety regulation. The USNRC rules have the force of law, and compliance with them is required. Regulatory guides are published by the USNRC staff to implement the rules and provide detailed guidance. The guides identify design approaches and solutions to safety issues that are acceptable to the staff, but also allow other approaches that give an equivalent level of safety. In addition to USNRC rules and guides, nuclear facility design, construction, and operation are governed by codes and standards published by the American National Standards Institute, the American Society of Mechanical Engineers (ASME), the Institute of Electrical and Electronics Engineers, and a number of other professional engineering societies and standards organizations.

### **22.8.2 Reactor Safety Basics**

The basic safety philosophy of nuclear plant design and operation is that the design should be conservative, there should be ample safety margins in operation, and redundant and diverse components and systems should be provided to cover the unknown uncertainties. A concept called defense-in-depth is used in developing reactor safety.

#### **22.8.2.1 Defense-in-Depth**

According to the USNRC (1999), the definition of the defense-in-depth is

An element of the USNRC's safety philosophy that employs successive compensatory measures to prevent accidents or lessen the effects of damage if a malfunction or accident occurs at a nuclear facility. The USNRC's safety philosophy ensures that the public is adequately protected and that emergency plans surrounding a nuclear facility are well conceived and will work. Moreover, the philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a nuclear facility.

The characteristics of defense-in-depth involve five levels of safety-oriented effort (IAEA, 1996). The first is based on the objective that nuclear facilities should function with a high

degree of reliability. This objective involves designing the plant to conservative standards and engineering practices so there will be a large tolerance for system transients, operator errors, and off-normal operation. Plant systems are designed to allow monitoring of components for signs of wear or incipient failure.

The second level is based on the fact that failures and malfunctions must be anticipated during the service life of the plant. Measures to forestall or cope with such events are provided. All important safety systems are required to be redundant to assure that no single failure will cause the loss of a needed safety function. The third level involves additional margins in the plant design to protect the public against highly unlikely accidents. To establish these additional margins, major failures of plant components are postulated and the accident sequences that would follow these events are analyzed. Safety systems and other plant design features are then provided to mitigate the consequences of these major failures. In addition to plant component failures, environmental phenomena such as tornadoes, floods, and earthquakes are hypothesized and plant safety elements are designed to withstand these events. In most cases, a limited number of these postulated major failures or natural events turn out to be limiting from the standpoint of the plant safety design; these are called the design basis accidents (DBAs) and natural events.

Should the third level fail, the fourth level limits accident progression through accident management, so as to prevent or mitigate severe accident conditions with external releases of radioactive materials. The last objective (fifth level of protection) is the mitigation of the radiological consequences of significant external releases through the off-site emergency response (Kadambi, 2006).

Thus, defense-in-depth concept of reactor safety involves three basic principles: prevention, protection, and mitigation (Sehgal, 2006). The prevention principle comprises robust design of the reactor and training of personnel for maintenance and operation of the reactor. The reactor plant is designed to have inherent reactor stability such as to have negative temperature coefficient of reactivity for the core. This is typically accomplished with a core design having negative fuel Doppler coefficient, adequate expansion of coolant-moderator, and initial loading of burnable poison. The components design and materials are chosen with highest quality assurance program. Use of nuclear specialized-grade codes and standards provide adequate confidence in component structure of system. Systems are designed with redundancy with similar parallel systems to reduce probabilities of failure and with diverse systems to avoid common mode failure. The nuclear personnel are training and qualified and are required to have a license for operation and maintenance of any nuclear facility.

### **22.8.2.2 Multilayer Barriers and Protection**

As part of defense-in-depth strategy multilayer protections are provided to an NPP to prevent escape of the radioactive material from the core. As shown in Table 22.17 these protection layers are designed to eliminate or minimize the radioactive material contamination from fuel pellet to the outside environment.

### **22.8.3 Reactor Safety System and Functions**

In addition to the barriers and protection layers, the reactor is also designed with safety system. The three primary objectives of nuclear safety systems are (1) shut down the reactor, (2) maintain it in a shutdown condition, and (3) prevent the release of radioactive material during events and accidents. These objectives are accomplished using a variety of equipment, which are part of different systems, of which each performs specific functions. The safety systems are provided for each NPP and are explained below.

**TABLE 22.17**

Multilayer Protection from Nuclear Power Plant Radioactive Material

Barrier or Protection Layer	Purpose
1. Ceramic fuel pellets	Only a fraction of the gaseous and volatile fission products is released from the pellets.
2. Metal cladding	The cladding tubes contain the fission products released from the pellets. During the life of the fuel, less than 0.5% of the tubes may develop pinhole-sized leaks through which some fission products escape.
3. Reactor pressure vessel and primary piping	Thick steel vessels and pipes contain the reactor cooling water. A portion of the circulating water is continuously passed through filters to keep the radioactivity low.
4. Containment	The nuclear primary system is enclosed in a containment building strong enough to withstand the major pipe break in the reactor primary coolant system.
5. Exclusion area	A designated area around each plant separates the plant from the public. Restricted entrance.
6. Low population zone/evacuation plan	Residents in the low population zone are protected by emergency evacuation plans.
7. Population center distance	Plants are located at a distance from population centers.

Source: Haskin, F.E. et al., Perspectives on reactor safety, NUREG/CR-6042, SAND93-0971, Revision 2, March U.S. Nuclear Regulatory Commission Report, 2002. With permission.

### 22.8.3.1 Reactor Protection System

The reactors can handle over power rating from 112% to 120% of full design power, and if power is more than this over power capability, they are set to automatically shut down or trip or scram. The reactors have control rods with neutron absorbers such as boron and gadolinium that stop nuclear chain reaction in the core and shut down the reactor instantaneously. The trip is a manually triggered or automatically triggered rapid insertion of all control rods into the reactor. In case of PWR, the control rods which are held by electromagnetic clutches drop down shutting down the reactor. In case of BWR shutdown, control rods are pushed up by hydrostatic pressure and the recirculation pumps are stopped to eliminate re-criticality of the core. A nuclear reaction can also be stopped by injecting liquids containing chemicals that absorb neutrons directly into the core. In BWRs this usually consists of a solution containing boric acid, which can be injected to displace the water in the core. In PWR, boron solution is used in addition to control rods to control the reaction, and so the concentration is simply increased to slow or stop the reaction.

The reactor is shut down with various trip signals that are indicative of the abnormal transient status of reactor. To avoid false trip two or more independent detectors for trip signal are used. Typical trip signals used in PWR are

1. *High power*: high nuclear power and high differential temperature between the inlet and outlet of the reactor vessel
2. *High startup rate*: (active below 10%–4% power) at low power levels
3. High pressurizer pressure
4. Low reactor coolant flow
5. *Thermal margin/low pressure* (reactor power versus RCS pressure)
6. High containment pressure

7. Low steam generator level
8. Low steam generator pressure
9. Loss of load (main turbine trip)

Actuation of any two channels would result in an automatic SCRAM or reactor shutdown. The system also allows manual actuation by the operator.

In case of BWR the following trip signals can shut down the reactor:

1. Turbine stop-valve or turbine control-valve closure due to pressure transient or generator load rejection
2. Loss of offsite power
3. Neutron monitor trips—during startup-uprate, variable coolant flow to match power
4. Low reactor water level either due to loss of coolant or loss of flow
5. High drywell (primary containment) pressure
6. Main steam isolation valve (MSIV) closure either due to presence of radiation in main steam line or break of MSL
7. High reactor vessel pressure which can happen due to MSIV closure, void collapse, or failure to open pressure relief valve

### **22.8.3.2 Essential Service Water System**

The reactors operate based on Rankine cycle and typically the thermal efficiency of this power cycle is 33%. So remaining heat produced in the reactor core is discharged to the environment. The essential service water system circulates the water that cools the NPP's various heat exchangers and other components before dissipating the heat into the environment. During a shutdown status, this service water system removes decay heat from both the primary system and the spent fuel rod cooling ponds, and it is safety-critical system. The water is frequently drawn from an adjacent river, the sea, or other large body of water and in locations without a large body of water in which to dissipate the heat, water is recirculated via a cooling tower.

### **22.8.3.3 Emergency Core Cooling System**

While the reactor protection system is designed to prevent accidents from happening, the emergency core cooling system (ECCS) is designed to respond to those accidents if they do happen. The ECCS is a set of interrelated safety systems that are designed to protect the fuel within the reactor pressure vessel, which is referred to as the *reactor core*, from overheating. These systems accomplish this by maintaining reactor pressure vessel (RPV) cooling water level, or if that is impossible, by directly flooding the core with coolant. Under normal conditions, heat is removed from a nuclear reactor by condensing steam after it passes through the turbine. In a BWR, condensed steam (water) is fed back into the reactor. In a PWR, it is fed back through the heat exchanger. In both cases, this keeps the reactor core at a constant temperature. During an accident, the condenser is not used, so alternate methods of cooling are required to prevent damage to the nuclear fuel. These systems allow the plant to respond to a variety of accident conditions, and

additionally introduce redundancy so that the plant can be shut down even with one or more subsystem failures.

#### 22.8.3.3.1 BWR ECCS

The BWR ECCSs consist of two high-pressure and two low-pressure systems. The high-pressure systems are the high-pressure coolant injection (HPCI) system and the automatic depressurization system (ADS). The low-pressure systems are the low-PCI (LPCI) mode of the residual heat removal system and the core spray (CS) system. The operation of the high-pressure and low-pressure cooling systems depends on the rate at which reactor coolant inventory is lost from the primary coolant line break and rate of decreases of reactor vessel pressure. The HPCI system is designed to operate at high pressure, while the CS system and LPCI mode of the residual heat removal system are designed for operation at low pressures. If the break in the nuclear system process barrier is of such a size that the loss of coolant exceeds the capability of the HPCI, reactor pressure decreases at a rate fast enough for the low-pressure ECCSs to start injection of coolant into the reactor vessel in time to cool the core. Automatic depressurization is provided to automatically reduce reactor pressure if a break has occurred and the HPCI system is inoperable.

HPCI system is an independent ECCS requiring no auxiliary AC power, plant air systems, or external cooling water systems to perform its purpose of providing make up water to the reactor vessel for core cooling under small- and intermediate-size LOCAs. This system consists of a pump or pumps that have sufficient pressure to inject coolant into the reactor vessel while it is pressurized. It is designed to monitor the level of coolant in the reactor vessel and automatically inject coolant when the level drops below certain setpoints. The HPCI is powered by steam from the reactor, and takes approximately 10 s to spin up from an initiating signal, and can deliver large volume of water (19,000 L/min or 5,000 US gal/min) to the core at any core pressure above 6.8 atm (690 kPa, 100 psi). This is usually enough to keep water levels sufficient to avoid automatic depressurization except in a major accident, such as a large break in the makeup water line. The water is pulled from suppression pool (wet well) and in some cases from condensate storage tank.

The ADS consists of redundant logics capable of opening selected safety relief valves, when required, to provide reactor depressurization for events involving small- or intermediate-size LOCAs if the HPCI system is not available or cannot recover reactor vessel water level.

The low-pressure ECCSs consist of two separate and independent systems, the CS system and the LPCI mode of the residual heat removal system. The CS system consists of two separate and independent pumping loops, each capable of pumping water from the suppression pool into the reactor vessel. Core cooling is accomplished by spraying water on top of the fuel assemblies. The LPCI mode of the residual heat removal system provides makeup water to the reactor vessel for core cooling under LOCA conditions.

The residual heat removal system is a multipurpose system with several operational modes, each utilizing the same major pieces of equipment. The LPCI mode is the dominant mode and normal valve lineup configuration of the residual heat removal system. The LPCI system operates automatically when reactor vessel pressures are below 465 psi. The LPCI consists of four pumps driven by diesel engines and is capable of injecting a large amount of water (150,000 L/min or 40,000 US gal/min) into the core to restore and, if necessary, maintain the reactor vessel coolant inventory to preclude fuel cladding temperatures in excess of 1204°C. During LPCI operation, the residual heat removal pumps take water from the suppression pool and discharge to the reactor vessel.

Containment spray system is part of BWR ECCS. This system consists of a series of pumps and spargers which spray coolant into the primary containment structure. It is designed to condense the steam into liquid water within the primary containment structure to prevent overpressure, which could lead to involuntary depressurization. If the reactor building is isolated from the control and turbine buildings, a standby cooling system called isolation cooling system, often driven by a steam turbine, is used to provide enough water to safely cool the reactor. As it does not require large amounts of electricity to run, and runs off the plant batteries, rather than the diesel generators, it is a defensive system against a condition known as station blackout.

#### 22.8.3.3.2 PWR ECCS

The PWR ECCS consists of four separate systems: the high-pressure injection (or charging) system, the intermediate pressure injection system, the cold leg accumulators, and the low-pressure injection system (residual heat removal) to cope with any pipe break ranging from a large break to a small one. Two sets of each system inject large amounts of cool, borated water into the reactor coolant system, each of which is capable of providing sufficient flow to minimize fuel damage following a LOCA. Since these systems must be able to operate when the normal supply of power is lost to the plant, they are powered from the plant emergency (diesel generators) power system.

The high-pressure injection system uses the pumps in the chemical and volume control system. Upon receipt of an emergency actuation signal, the system will automatically realign to take water from the refueling water storage tank and pump it into the reactor coolant system. The high pressure injection system is designed to provide water to the core during emergencies in which reactor coolant system pressure remains relatively high (such as small break in the reactor coolant system, steam break accidents, and leaks of reactor coolant through a steam generator tube to the secondary side). The intermediate pressure injection system is also designed for emergencies in which the primary pressure stays relatively high, such as small- to intermediate-size primary breaks. Upon an emergency start signal, the pumps will take water from the refueling water storage tank and pump it into the reactor coolant system.

The cold leg accumulators do not require electrical power to operate. These tanks contain large amounts of borated water with a pressurized nitrogen gas bubble in the top. If the pressure of the primary system drops below low enough, the nitrogen will force the borated water out of the tank and into the reactor coolant system. These tanks are designed to provide water to the reactor coolant system during emergencies in which the pressure of the primary drops very rapidly, such as large primary breaks.

The low-pressure injection system (residual heat removal) is designed to inject water from the refueling water storage tank into the reactor coolant system during large breaks, which would cause a very low reactor coolant system pressure. In addition, the residual heat removal system has a feature that allows it to take water from the containment sump, pump it through the residual heat removal system heat exchanger for cooling, and then send the cooled water back to the reactor for core cooling. This is the method of cooling that will be used when the refueling water storage tank goes empty after a large primary system break. This is called the long-term core cooling or recirculation mode.

#### 22.8.3.4 Emergency Electrical Systems

During normal operating conditions, NPPs receive power from off-site. However, during an accident a plant may lose access to this power supply and hence emergency electrical

systems are of reactor safety systems. These electrical systems usually consist of diesel generators and batteries. Diesel generators power the site during emergency situations. They usually are sized such that a single one can provide all the required power for a facility to shut down during an emergency situation which allows facilities to have multiple generators for redundancy. Additionally, systems which are not required to shutdown the reactor have separate electrical sources (often their own generators) so that they do not affect shut down capability.

Loss of electrical power can occur suddenly, and it can damage or undermine equipment. To prevent damage, motor-generators can be tied to flywheels which can provide uninterrupted electrical power to equipment for a brief period of time in the order of 30 s. Often they are used to provide electrical power until the plant electrical supply can be switched to the batteries and/or diesel generators. Batteries often form the onsite redundant backup electrical system and are also capable of providing sufficient electrical power to shut down a plant. The DC power generated by batteries can be converted to AC power to run AC devices such as motors using an electrical inverter.

### **22.8.3.5 Containment Systems**

The containment systems provide successive barrier to the limit or avoid escape of radioactive material from the reactor to the environment. After the fuel pellet, cladding, and the primary boundary, the containment structure is the last physical barrier between reactor and the environment. If every possible measure standing between safe operation and core damage fails, the containment can be sealed indefinitely, and it will prevent any substantial release of radiation to the environment from occurring in nearly any circumstance. The containment designed such that if all primary coolant flashes and pressurizes the containment, it should be less than the containment design pressure. Typical containment design pressure is 350 kPa (50 psig).

#### *22.8.3.5.1 BWR Containment*

During the evolution of the BWRs, three major types of containments were built: Mark I (page 3–16), Mark II, and the Mark III (page 3–18). Unlike the Mark III, that consists of a primary containment and a drywell, the Mark I and Mark II designs consist of a drywell and a wetwell (suppression pool). All three containment designs use the principle of pressure suppression for LOCAs. The primary containment is designed to condense steam and to contain fission products released from a LOCA so that offsite radiation doses specified are not exceeded and to provide a heat sink and water source for certain safety related equipment.

The Mark I containment design consists of the following components: (1) the drywell, made of steel dome of 1 in. thick that surrounds the reactor vessel and recirculation loops; (2) a suppression chamber, which stores a large body of water (suppression pool); (3) an interconnecting vent network between the drywell and the suppression chamber; and (4) the secondary containment, which surrounds the primary containment (drywell and suppression pool) and houses the spent fuel pool and ECCSs.

The Mark II primary containment consists of a steel dome head and either a post-tensioned concrete wall or reinforced concrete wall standing on a base mat of reinforced concrete. The inner surface of the containment is lined with a steel plate that acts as a leak-tight membrane. The drywell, in the form of a frustum of a cone or a truncated cone, is located directly above the suppression pool. The suppression chamber is cylindrical and separated from the drywell by a reinforced concrete slab. The drywell is topped by an elliptical steel dome called a drywell head. The drywell inerted atmosphere is vented into

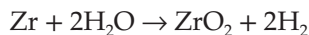
the suppression chamber through series of downcomer pipes penetrating and supported by the drywell floor.

The Mark III primary containment consists of drywell which is a cylindrical, reinforced concrete structure with a removable head and suppression pool. The drywell and wetwell are connected via the weir wall and the horizontal vents. The suppression pool contains a large volume of water for rapidly condensing steam directed to it. A leak-tight, cylindrical, steel containment vessel surrounds the drywell and the suppression pool to prevent gaseous and particulate fission products from escaping to the environment. The containment system has water spray system at top of the drywell that activates when the containment pressure increases from a set point.

#### 22.8.3.5.2 PWR Containment

PWR containment is relatively large compared to BWR containment as it houses reactor primary system and the steam generators. The PWR containment is large cylindrical concrete structure with steel liner that covers the inside surface of the containment building. This liner acts as a vapor-proof membrane to prevent any gas from escaping through any cracks that may develop in the concrete. There are also two systems designed with the purpose of reducing containment temperature and pressure after an accident in the containment building. The fan cooler system circulates the air through heat exchangers to accomplish the cooling. The second system is the containment spray system. Upon the occurrence of either a secondary break or primary break inside the containment building, the containment atmosphere would become filled with steam. To reduce the pressure and temperature of the building, the containment spray system is automatically started. The containment spray pump will take a suction from the refueling water storage tank and pump the water into spray rings located in the upper part of the containment. The water droplets, being cooler than the steam, will remove heat from the steam, which will cause the steam to condense. This will cause a reduction in the pressure of the building and will also reduce the temperature of the containment atmosphere (similar to the operation of the pressurizer). Like the residual heat removal system, the containment spray system has the capability to take water from the containment sump if the refueling water storage tank goes empty.

As mentioned before during an accident when the nuclear fuel overheats, zirconium in zircaloy cladding used in fuel rods oxidizes in reaction with steam



When mixed with air, hydrogen is flammable, and hydrogen detonation or deflagration may damage the reactor containment. In reactor designs with small containment volumes, such as in Mark I or II containments, the preferred method for managing hydrogen is pre-inerting with inert gas—generally nitrogen—to reduce the oxygen concentration in air below that needed for hydrogen combustion, and the use of thermal recombiners. Pre-inerting is considered impractical with larger containment volumes where thermal recombiners and deliberate ignition are used.

## 22.8.4 Reactor Accidents and Transients

### 22.8.4.1 Safety Analysis Report

When a new reactor is designed and the vendor plans to market the reactor, the reactor design must be licensed by the regulators. A company planning to construct and operate a nuclear facility must obtain construction permit and operation license from the regulators.



For this a design and safety analysis for the proposed facility is required. For new reactors the USNRC reviews applications submitted by prospective licensees, and when appropriate issues standard design certifications, early site permits, limited work authorizations, construction permits, operating licenses, and or combined licenses for construction and to operate. The standard design certifications need a detailed safety analysis report (SAR) and environmental report. SAR will contain details on safety requirements, the design basis, site and plant characteristics, operational limits and conditions, and safety analyses in such a way that the regulatory body will be able to evaluate independently the safety of the plant. The SAR also referred as design control document after design certification, provides information to support the USNRC's approval and certification of new reactor, under the provisions of Title 10, Part 52, of the *Code of Federal Regulations* (10 CFR Part 52), "Licenses, Certifications, and Approvals for Nuclear Power Plants."

The SAR is divided into two parts (called *tiers*): Tier 1 document is derived from the more-detailed Tier 2 document—high-level information on the plant design, characteristics and safety functions, and significant site parameters. It includes inspections, tests, analyses, and acceptance criteria to provide reasonable assurance that the as-built plant will operate in conformance with the combined license, the provisions of the Atomic Energy Act, and applicable USNRC regulations. Tier 2 document gives detailed information on the plant design analysis.

The SAR identifies events or transient, event sequences, and event combinations that can potentially challenge the safety or control functions of the NPP. These events or transients are further discussed in Sections 22.8.4.2 and 22.8.4.3.

#### **22.8.4.2 Transient and Accidents**

The plant conditions considered in the safety analyses for LWRs range from relatively mild events to DBAs and natural events that have potential for serious consequences. The transient or event in an NPP can result due to (1) component and system failures or malfunctions, (2) operator errors, and (3) common-cause internally and externally initiated events. Most of the transient leading away from normal operation are results of imbalance between core heat generation and heat removal from the core. Reactor transients generally are classified based on the results of probabilistic studies and engineering judgment, into the four classes of events as shown in Table 22.18.

The AOT may reasonably be expected to occur sometime in the service life of a plant. These events lead to no significant releases of radioactivity. The partial list of AOT include RPV pressure increase, RPV water (moderator) temperature decrease, control rod withdrawal, RPV coolant inventory decrease, reactor core coolant flow decrease, reactor core coolant flow increase, core coolant temperature increase, and excess of coolant inventory.

The DBA are postulated in order to establish performance requirements for the safety systems of the plant. The DBA are due to (1) mechanical failure of a single component leading to the release of radioactive materials from one or more barriers; (2) arbitrary rupture of any single pipe up to and including complete severance of the largest pipe in the reactor coolant pressure boundary (LB-LOCA). In addition, very severe postulated natural events form part of the design basis conditions. These include a tornado substantially more severe than any yet observed and flooding.

Beyond DBA (BDBA) and severe accidents can result from one of the initiating events as in case of DBA; however combination of failure of ECCS system and emergency systems can lead to these events. Such accidents include those initiated by events, such as RPV rupture or a seismic event more severe than the safe shutdown earthquakes, that

TABLE 22.18

Classes of Reactor Events or Transients and Their Occurrences

Occurrence (1/reactor year)	Characteristics	Terminology	Acceptance Criteria
10 <sup>-2</sup> –1 (Expected in the life of the plant)	Expected Anticipated operational occurrences	Anticipated transients, transients, frequent faults, incidents of moderate frequency, upset conditions, abnormal conditions	No additional fuel damage
10 <sup>-4</sup> –10 <sup>-2</sup> (Chance greater than 1% over the life of the plant)	Possible, DBAs	Infrequent incidents, infrequent faults, limiting faults, emergency conditions	No radiological impact at all or no radiological impact outside the exclusion area
10 <sup>-6</sup> –10 <sup>-4</sup> (Chance less than 1% over the life of the plant)	Unlikely, BDBAs	Faulted conditions	Radiological consequences outside exclusion area within limits
<10 <sup>-6</sup> (Very unlikely to occur)	Remote chance, severe accidents	Faulted conditions	Emergency response needed

Source: IAEA, Accident analysis for nuclear power plants, Safety Reports Series No. 23, International Atomic Energy Agency, Vienna, Austria, 2002. With permission.

are not analyzed in the SAR. Other accidents beyond the design basis include accidents involving multiple component failures or operator errors, that is, failures beyond those postulated under the single failure criteria. N include station blackout (loss of offsite and onsite AC power), LOCAs, and ATWS. The results of analysis of these events help to determine measures to prevent severe accidents and to mitigate the radiological consequences. Accident management and emergency response measures are necessary if all the barriers against radioactive releases are significantly degraded in a BDBA. For severe accidents, containment and/or confinement typically remains as the only barrier to limit accidental releases.

### 22.8.4.3 LOCA and ECCS

LOCA is an important DBA in the safety analysis of reactor designs (Sehgal, 2006). LOCA varies with each reactor design and phenomena depending on the size of break. In case of primary side piping break the LOCA is classified as follows:

- *LB-LOCA*: break in 6–36 in. equivalent diameter pipe
- *Medium-break LOCA*: break in 2–6 in. equivalent diameter pipe
- *Small-break LOCA*: break in ½–2 in. equivalent diameter pipe

Reactor vessel rupture, steam generator rupture, reactor coolant system interface rupture can lead to LOCA.

The postulated LOCA begins with the assumption of a break in a primary coolant system pipe. The coolant discharges from the break and flashes to steam in the containment

**TABLE 22.19**

Core Heat Up and Its Consequences Phenomena

Temperature (°C)	Phenomena
350	Normal clad temperature during operation
800–1450	Clad swelling- Kr, I, Xe release, reaction between SS and Zr, channel blockage
1450–1500	Steam Zr reaction, Zr embrittlement, H <sub>2</sub> release
1550–1600	Steam Zr autocatalytic reaction, H <sub>2</sub> release
1900	Zr melt, significant fission product release-UO <sub>2</sub>
2700	Both ZrO <sub>2</sub> and UO <sub>2</sub> melt

building surrounding the reactor. At the end of the blowdown, which would take only 20 or 30 s for the largest pipe breaks, the reactor system is filled mostly with low-pressure saturated steam. The loss of water from the reactor core stops the nuclear chain reaction, and insertion of the control rods and neutron-absorbing solutions keep the reactor shut-down after the core is reflooded. The ECCS supplies water to the core after the blowdown to maintain acceptable temperature conditions in the fuel rods. The ECCS must continue to function in a recirculation mode, with heat rejection to sinks outside the containment, for a long period after the accident. In case of failure to cool the core, core will heat up with increase in temperature leading to core component damage as listed in Table 22.19.

The typical ECCS for PWRs is composed of three independent subsystems, each with equipment and flow path redundancy. This means that the ECCS will function in a fully effective way even with a major component failure in any subsystem. In fact, a number of component failures can occur in the various ECCS subsystems and not result in overall failure of the cooling function, provided at least one of the redundant trains in each subsystem remains operative. The first subsystem is an accumulator injection system that operates by forcing borated water stored under gas pressure in the accumulator tanks into the primary system. The second subsystem is a high-volume, low-pressure pumping system that comes into operation following blowdown and injection of water from the accumulator tanks. The third system is a high-pressure pumping system provided to supply coolant if a small break occurs and the primary system pressure does not decrease enough for the accumulator system or low-pressure pumping systems to operate effectively.

The typical ECCS for BWRs is also composed of three subsystems with equipment and flow path redundancy. The first subsystem is a high-pressure spray system that sprays water on the reactor core. For small pipe breaks, the high-pressure spray can maintain the water level in the reactor vessel so that the core remains covered. For larger breaks, this system cools the core by spraying water on the fuel rods. The second subsystem is a low-pressure spray system that delivers a large-volume water spray to the top of the core. The third subsystem is a low-pressure core injection system that provides a large flow of water to the reactor vessel to reflood and cover the core.

#### 22.8.4.4 Safety Analysis Methods

Performance of safety analysis consists of identification and selection of PIEs, evaluation of event sequences, evaluation of event consequences, and comparison against acceptance criteria. The analysis should identify the DBAs and the Beyond DBAs (BDBAs) for the purpose of establishing acceptance criteria.

PIEs are unintended events that directly or indirectly challenge the basic safety functions. Such events necessitate protective actions to prevent or mitigate undesired consequences. The results of the safety analysis are used for the determination of the operating limits and conditions, development of the periodic testing and inspection programs, development of the operating procedures and emergency planning. The well-accepted approach for safety analysis for reactors is to consider credible PIEs and then apply deterministic methods with sufficient conservatism for the safety demonstration and estimate the maximum possible releases to the environment. The results are compared against acceptance criteria established by the regulatory body.

For reactor safety analysis DBA analyses such as LOCA is primarily used for LWRs. USNRC has established minimum requirements for the principal design criteria of LWRs as given in Appendix A to USNRC publication 10 CFR 50 (USNRC 2012, 2014a). General Design Criterion No. 35 establishes that ECCSs are required for cooling reactor cores in the event of a rupture of pipes of the reactor coolant system or an inadvertent opening of a relief valve or safety valve of this system. In the event that the ECCS should be used, there are a list of requirements that the ECCS is responsible to uphold to ensure the stability of the core and safety of the reactor. Below is a list of these criteria as given in 10 CFR 50.46 (USNRC 2014b):

1. Peak fuel cladding temperature shall not exceed 1204°C (2200°F).
2. The local cladding oxidation will not exceed 0.17 times the thickness of the cladding prior to oxidation.
3. Hydrogen generation from the cladding shall not exceed 0.01 times the hydrogen that would be generated if all the metal in the cladding were to react and produce hydrogen.
4. The calculated changes in core geometry must allow the core to remain willing to cooling.
5. The ECCS must be capable of providing long term cooling such that core temperature remains at an acceptable low value as decay heat is generated.

Additionally, this US law adopted a *prescriptive regulatory approach*, requiring that the evaluation models to be used in licensing accident analysis must follow the conservative requirements established on Appendix K to 10 CFR 50 (USNRC 2014c). These conservative assumptions are stored heat 102% of peak power, low thermal conductivity of fuel pellet, conservative heat transfer coefficients, decay heat 20% larger than ANS Standard decay heat and 1200°C at the hottest channel.

In the deterministic methods (1) conservative or bounding data are used, (2) most critical reactor configuration (e.g., beginning of operation, etc.) is considered, (3) uncertainties associated with the parameter that affects the variable of interest are considered, and (4) actions by operators are considered not to be taken for a prescribed period of time. The probabilistic methods such as PSA may be used to complement deterministic methods. The probabilistic methods are used to (1) evaluate which accident sequence is of a greater likelihood, (2) identify hidden weakness in the design, and (3) quantify the value of possible improvements.

Safety analysis is used for design, licensing to construct and operate a reactor, and for supporting emergency planning (IAEA, 2004). The safety analysis is used to support design of a new reactor or modification to an existing one and to confirm that the design meets the relevant safety requirements. Typically, the safety analysis is carried in parallel

with the design process. The scope and level of details increase as design progresses. During design stage iterative process between the design and safety performance is used, and it accounts for all the challenges that the reactor may expect during its lifetime. In the design stage, safety analysis results are used in setting characteristics such as equipment sizing, operational limits, and dose to public and operator. The final analysis should reflect the as-built facility. In licensing process, the safety analysis is used to demonstrate that facility design features and the selected operational limits ensure that no credible accident should lead to unacceptable radiological consequences and to demonstrate compliance with regulatory requirements. Safety analysis for supporting emergency planning describes the reactor behavior in BDBAs. The results of BDBAs, defining the source term and radiological releases, could also be used for emergency planning.

The basic steps in the safety analysis are (1) identification and selection of PIEs, (2) evaluation of event sequences, (3) evaluation of event consequences, (4) comparison against acceptance criteria (IAEA, 2002). Identification and selection of PIEs consist of methods such as use of fault tree analysis. Through fault tree analysis failure modes and events that lead to these failures are identified. The operating experience from the facility or from other similar facilities and various databases are used. During identification and selection of PIEs, they are grouped in categories depending on the nature and components involved. For each category, the event that would be limiting is determined for further analysis. Selections of the limiting events are based on more detailed calculations, qualitative comparisons with other events, or engineering judgment. These various PIEs are listed under DBA in above section. Evaluation of event sequences in every PIE is performed by determination of the contributors (occurrences which lead to this event), sequence of event and reactor system operation, significant occurrences on timescale, indication of correct/incorrect functioning of system, instrumentation, reactor protection, evaluation of dependent failures, and required operator actions.

In safety analysis, the systems response are applied uniformly in the analysis. They constitute accident sequence and determine the sequences that are outside the design basis and thus excluded from further analysis. Evaluation of event sequences are done by calculations using qualified computational methods and tools such as best-estimate codes.

Best-estimate thermal hydraulics and neutronic codes are used to perform the analysis for the LOCA and other DBAs. It is required to identify and assess the associated uncertainties in the analytical models and input data in such a way that the uncertainties in the calculations can be quantified when the results are compared with the acceptance criteria, giving adequate assurance that the acceptance criteria are met. Mature best-estimate codes are widely available around the world, an extensive database exists for nearly all power reactor designs, and best-estimate plant calculations are well documented. The analysis of accidents with best-estimate codes using combinations of best-estimate and conservative inputs is particularly appropriate since this approach provides some estimates of the uncertainties in the overall plant behavior. These estimates can then be compared with the uncertainty estimates developed through relevant activities in different countries in code validation, as well as studies on representation and uncertainties in plant data, to help establish confidence in the predicted behavior of the plant. This approach is dependent on the continued emphasis on activities in development and validation of best-estimate codes to ensure that such codes can be used with a high degree of confidence.

The process in the computational works involves identification of the input parameters and initial conditions for the analysis. Results of the analysis provide values of the key

parameters as a function of time during the course of the transient. From these results damage state and event consequence are determined. The radiological consequences of the events that are bounding (or limiting) for each category are then estimated. The results of the safety analysis are compared against pre-established acceptance criteria. Acceptance criteria may be basic or specific. Examples of basic acceptance criteria are the maximum allowed dose to the public or prevention of fuel cladding failure. Specific acceptance criteria specified by the regulatory body are used to include additional margins beyond the basic criteria, such as maximum cladding temperature lower than certain value (e.g., 1204°C), maximum heat flux not exceeding the CHF, or onset of significant void.

### 22.8.5 Probabilistic Safety Assessment

In a nuclear facility, as in any industrial plant, risk assessment distinguishes between the potential hazards that might be encountered in the absence of any protective measures, and the residual risks that will still remain despite the measures taken. The problem lies in assessing the latter, since there is no way of ensuring that they have been completely eliminated. The concept of event probability and its associated consequences was rapidly incorporated into safety analysis procedures, by taking account of the fact that the probability of an accident must be inversely proportional to the severity of the potential consequences for the public and the environment.

The deterministic approach to the design of nuclear reactors was rapidly supplemented by the development of probabilistic studies, referred to as PSAs and also as PRA. The first study of this kind carried out in the United States was published in 1975 (Rasmussen report—USNRC 1975) and provided the first assessment of the potential risk of core damage for two power reactors. The accident in 1979 at the Three Mile Island plant generated renewed interest in this type of study. One of the recommendations made after the accident was that probabilistic analysis techniques should be used to supplement conventional safety assessment procedures for NPPs, and that probabilistic objectives should be developed in order to facilitate the determination of acceptable safety levels for nuclear facilities.

PSAs can be used to calculate the probability of damage to the core as a result of sequences of accidents identified by the study. With the development of this type of analyses, PSAs can now also be used to assess the size of radioactive releases from the reactor building in the event of an accident, as well as the impact of such releases on the public and the environment. For NPPs it is usual to perform PSAs for three different ranges (IAEA, 1995; Kröger and Chang Sang-Lung, 2006):

- Level I: Risk assessment based on event scenarios that can lead to core damage.
- Level II: Risk assessment based on event scenarios that can lead to core damage and consequently to a loss of containment.
- Level III: Risk assessment based on event scenarios that can lead to core damage, a loss of containment, and consequently to an impact on the surrounding public and the environment.

A typical PSA involves acquiring an in-depth understanding of the facility and collecting a large volume of related information, identifying initiating events and states of plant damage, modeling the main systems within the plant using event and fault trees, assessment of the relationships between events and human actions, and development of a database on

the reliability of a specific plant's systems and components. First part of a PSA generally comprises an initial section known as a probabilistic assessment of initiating events, which is aimed at identifying and estimating the frequencies of initiating events that might lead to severe core damage, or even meltdown, as a result of either a safety system failure or human error. The second part of the analysis assesses the reliability of systems designed to meet safety requirements. This assessment consists in the identification, for each system and function reviewed, of failures that might result in the loss of the system's function. The probability of each type of failure occurring is then calculated and the failures can be ranked by decreasing order of probability. Potential weaknesses in the facility may thus be revealed. This part of the assessment is particularly important because its results will largely depend on the reliability of the data used in calculations. Reliability values must be based on data which are representative of plant operating experience and thus on the incidents and events observed in the systems concerned.

The third part of the analysis is aimed at identifying and assessing sequences of events that might lead to a severe accident, that is, damage to the core resulting in core melt. For this, analysts generally use the event-tree method which consists in identifying accident sequences from individual initiating events and then postulating the failure of the safety systems triggered by the event in question. The safety system failures postulated are those identified and calculated in the previous stage of the assessment. This underlines the importance of collecting reliable data, as noted above.

In Figure 22.32, a simplified PRA logic is shown. It is by rule a *cause–consequence logic* composed of fault trees and event trees. The trees are intermeshed to a global model. Normally, about 30 event trees and 100 fault trees are combined. The event trees describe a success/failure logic, the fault trees the undesired TOP events. In Figure 22.32, the summing up procedures *via* binning of similar outcomes of event trees and so on is not visualized. The entire combination of event and fault trees represents a probabilistic model. This means all the causes (basic events), the logic gates (AND, OR, *etc.*) in the fault trees, the initiating events, the gates in the event trees (success or failure, respectively yes or no), the core damage, the containment failure, and all the various estimated consequences are characterized by quantified probabilities (ASME, 2002).

Just as the deterministic approach has its limitations, so too does probabilistic assessment. This is due to the fact that the results of a PSA invariably contain uncertainties

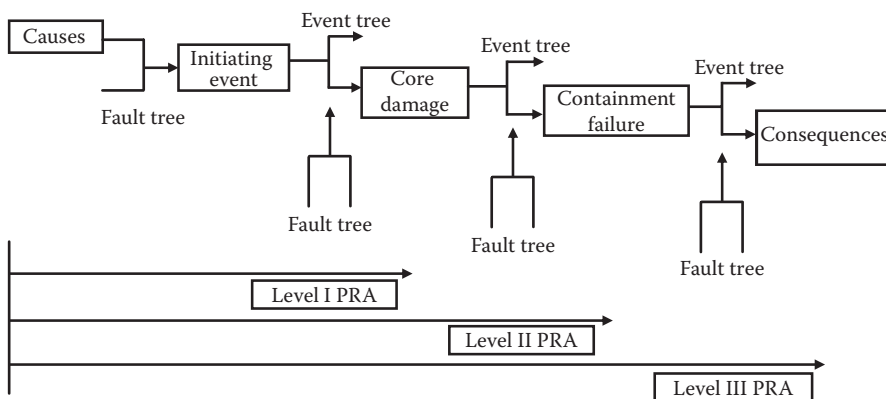


FIGURE 22.32

Simplified PRA logic without binning procedures shown.

arising from three main sources: (1) uncertainties due to a lack of comprehensive data regarding the area under consideration. It is impossible to demonstrate the exhaustiveness of a PSA, even when the scope of the analysis has been extended to as large a number of situations as possible—notably in terms of various reactor operating states and potential initiating events; (2) uncertainties regarding data. Such uncertainties concern the reliability data for plant components, the frequency of initiating events, common-mode failures, and failures resulting from human actions. The main uncertainties are those relating to the frequency of rare initiating events (e.g., the combination of a steam piping break and a steam-generator tube break), as well as data relating to human factors; and (3) uncertainties associated with modeling assumptions that cannot easily be quantified, such as the resistance of certain components under accident conditions, poorly understood physical phenomena, and human actions.

In view of these uncertainties, the assumptions on which PSAs are based are designed to ensure sufficient safety margins. It is worth noting that the uncertainties are not intrinsic to PSAs, but may generally be attributed to lack of detailed knowledge. Indeed, one of the benefits of conducting PSAs is that they can identify areas about which we need to learn more.

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# 23

## *Thermodynamics and Power Cycles*

Peter D. Friedman

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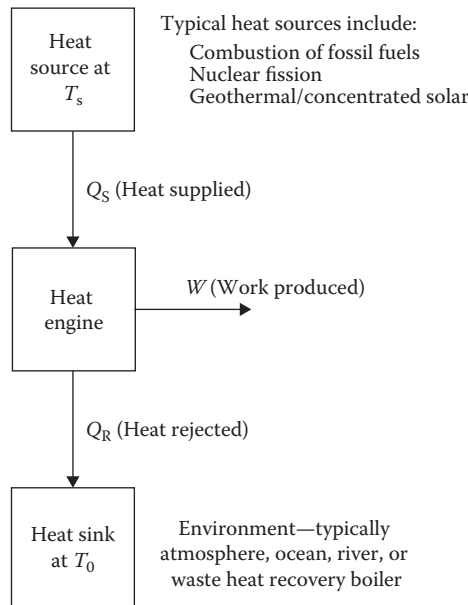
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### 23.1 Introduction

This chapter summarizes fundamental principles of thermodynamics with a focus on applications related to analysis of nuclear power plants. For brevity, relatively few references are given in introductory material, but most of the introductory material may be found in a good thermodynamics textbook, such as Moran and Shapiro (2008). References on second law analysis tend to be advanced thermodynamic textbooks, such as Wark (1995) or more specialized references such as Moran (1982).

Thermodynamics is the study of energy transformations and the relationships among properties of substances. The *first law of thermodynamics* states that energy cannot be created or destroyed. In other words, energy is conserved. In this chapter, we are concerned with the application of thermodynamics to the analysis of *heat engines* (Figure 23.1) and their components. Heat engines are thermodynamic systems that receive heat from a heat



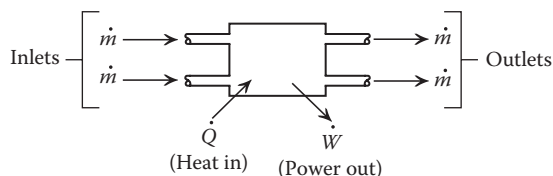
**FIGURE 23.1**  
Thermodynamic heat engine.

source and produce work. According to the *second law of thermodynamics*, heat engines must also reject a portion of the heat received to a lower temperature heat sink. The heat source for a heat engine can result from burning fossil fuels, process waste heat, geothermal sources, solar energy, or nuclear fission. The heat sink can be an environmental component, such as the atmosphere or a body of water, or alternatively, the heat can be used for other applications such as industrial process heat or as a heat source for heat engine operating at lower temperatures.

Heat engines operate in a *thermodynamic cycle*, which is simply a series of processes that repeat over time and use a working fluid, which is most commonly either a gas or a vapor. All heat engine cycles consist of a pressure increase process, a heat addition process where heat is added from the heat source, an expansion process where work is extracted, and a waste heat rejection process. In an attempt to improve the efficiency (the amount of work produced for a given heat input), the cycle can become greatly more complicated and include a number of intermediate processes. However, a heat engine must always include the basic elements of a heat supply and heat rejection.

In addition to the first law (conservation of energy), thermodynamic analysis applies conservation of mass and the second law of thermodynamics, which states that processes occur in a direction that the total quantity of entropy increases (entropy can be created, but not destroyed). Combining the first and second laws yields *exergy*, which is defined as the maximum quantity of useful work that can be produced as a system is reduced to the dead state (Wark, 1995). Unlike energy, exergy is not conserved; the quantity of exergy destroyed is proportional to the entropy generated. Exergy analysis allows the engineer to better analyze the sources of inefficiency in heat engines, which are often not evident using energy relationships.

Thermodynamic analysis applies energy, mass, entropy, and exergy balance equations to *thermodynamic systems*, which are simply the matter analyzed within a defined boundary. A closed system (or control mass) is a fixed collection of matter, while an open system (or control volume) allows a fluid to cross the system boundary at various inlets or outlets (see Figure 23.2). A steam turbine, for example, is an open thermodynamic system since the working fluid crosses the system boundary at the inlet and outlet. This chapter develops the thermodynamic balance equations and provides the background material required to use them in analysis of nuclear power plants. In developing generalized balance equations the sign convention shown in Figure 23.2 is used, in which work out is positive and heat in is negative. However, a more physically based sign convention is also used where appropriate, such as defining positive pump work as work supplied to a pump. The chosen sign convention is not universal; balance equations are frequently written based on work input as positive (Eastop and McConkey, 1993).



**FIGURE 23.2**  
Open thermodynamic system.

## 23.2 Fundamental Concepts

### 23.2.1 Units and Conversion Factors

Thermodynamics relationships are valid regardless of the unit system selected. Although formulas used by practicing engineers frequently include dimensional constants that account for unit conversions, prudent engineering practice dictates that units be carefully tracked and equations checked for dimensional consistency. Failure to account for units has resulted in a number of high-profile engineering failures, including the loss of a Mars Orbiter by NASA (Mystery, 1999). In the English Engineering (EE) unit system, Newton's second law ( $\vec{F} = m\vec{a}$ ) is often written with a proportionality constant ( $\vec{F} = m\vec{a}/g_c$ ). Regardless of whether  $g_c$  is specifically written, it can be included as necessary in any equation (including calculations with SI units) when required for dimensional consistency, where its value is

$$g_c = 32.2 \frac{\text{lb}_m \cdot \text{ft}}{\text{lb}_f \cdot \text{s}^2} = \frac{\text{kg} \cdot \text{m}}{\text{N} \cdot \text{s}^2} \quad (23.1)$$

### 23.2.2 Pressure and Temperature

Pressure gages measure *gage pressure*, which is the difference in pressure between system pressure and the surrounding atmosphere. In analyzing thermodynamic properties, the absolute pressure must be determined by adding the local atmospheric pressure.

$$P_{\text{abs}} = P_{\text{gage}} + P_{\text{atm}} \quad (23.2)$$

System pressures that are below atmospheric pressure are referred to as vacuum,

$$P_{\text{abs}} = P_{\text{atm}} - P_{\text{vac}} \quad (23.3)$$

For precise calculations, the atmospheric pressure should be measured using a barometer. If a high degree of accuracy is not required, standard atmospheric pressure for the local elevation above sea level may be used. At sea level, standard atmospheric pressure is

$$P_{\text{Std}}^{\text{Atm}} = \begin{cases} 1 \text{ atm} \\ 14.696 \text{ Psi} \\ 29.92 \text{ in Hg} \\ 1.01325 \text{ Bar} = 101.325 \text{ kPa} \end{cases} \quad (23.4)$$

Temperature scales are *relative* ( $^{\circ}\text{C}$ ,  $^{\circ}\text{F}$ ) or *absolute* ( $\text{K}$ ,  $^{\circ}\text{R}$ ). In any calculation involving temperature changes over time or temperature differences between locations, a relative temperature scale may be used. Calculations involving a single temperature (such as the ideal gas law) or ratio of temperatures require an absolute temperature scale. Absolute temperatures are determined from relative scales with the following relationships:

$$\text{Kelvin: } T(\text{K}) = T(^{\circ}\text{C}) + 273.15 \quad (23.5)$$

$$\text{Rankine: } T(^{\circ}\text{R}) = T(^{\circ}\text{F}) + 459.67 \quad (23.6)$$



### 23.2.3 Heat and Work

Work is a mechanical energy transfer that equals the product of the applied force and the displacement in the direction of the force,

$$W = \int \mathbf{F} \cdot d\mathbf{x} \quad (23.7)$$

Electrical energy transfer in a conductor is a work term since the charge and voltage are equivalent to a force and displacement. Electrical resistance, where electrical work is transformed into heat, is analogous to mechanical friction.

The definition of work in Equation 23.7 may be generalized by defining work as *any energy transfer that can completely result in the raising of an external weight*. Fundamentally, work is any energy transfer that has zero associated entropy (later work will be shown to be exergy, which is distinguished from heat transfer). Excluding irreversibilities, work and other forms of mechanical energy are 100% convertible. In contrast, in accordance with the second law of thermodynamics (discussed later) heat energy can only be partially converted into work, with the balance rejected at a lower temperature. By convention, the term *shaft work* includes all forms of work excluding *boundary work* and *fluid flow work*. Physicists generally account for elevation changes as gravitational work terms in energy balances, while in engineering it is common practice to include a potential energy term. The energy balances in this chapter include potential energy terms, so gravitational work should not be included in the work terms. The sign convention in equations representing the first law is to define work done by the system (work removed from the system) as positive work. For a work-consuming device (pump or compressor), an implied sign convention is that work into the system is positive.

Specific work is the work per unit mass contained in a closed system or the work per unit mass that passes through an open system.

$$w = \frac{W}{m} \quad (23.8)$$

Boundary work is the work done by a system boundary as it expands. Conversely, work done to compress a system is negative boundary work.

$$W = \int P dV \quad \text{or} \quad w = \int P dv \quad (23.9)$$

Boundary work is zero for rigid systems and in *steady-state* applications (conditions at a given location do not change with respect to time). Because a portion of the boundary work must overcome atmospheric pressure, the useful boundary work excludes the work done on the atmosphere.

$$W_{\text{Useful}} = \int (P - P_{\text{atm}}) dV \quad (23.10)$$

The boundary work associated with a fluid crossing a system boundary is referred to as the *flow work*. Applying Equation 23.9, the flow work can be shown to be,

$$w_{\text{flow}} = Pv \quad (23.11)$$

As a fluid enters a system, it does work on the system that equals the magnitude of the flow work and when a fluid exits the system, it removes the flow work from the system. Because flow work must be considered in the energy balances of open systems, the property *enthalpy* is defined as the sum of the internal energy and the flow work,

$$h = u + pv \quad (23.12)$$

*Power* is the rate at which work is performed or time derivative of work. It is related to specific work and the mass flow rate through an open system.

$$\dot{W} = \frac{\delta W}{dt} = \dot{m}w \quad (23.13)$$

where the  $\delta$  is used to represent the differential of a *path-dependent* quantity.

Heat is the transfer of thermal energy due to a temperature difference. By convention heat added to the system is positive. Analogous to power and work in Equations 23.8 and 23.13, net specific heat transfer and heat rate are defined as

$$\dot{Q} = \frac{\delta Q}{dt} = \dot{m}q \quad (23.14)$$

and

$$q = \frac{Q}{m} \quad (23.15)$$

### 23.2.4 Relativity and Fission Energy

According to the theory of relativity,  $E = mc^2$ , nuclear fission decreases mass and converts it to energy. However, because the mass reduction is insignificant for engineering purposes (one atomic mass unit for 931.4 MeV), energy and mass conservation equations are normally decoupled in thermodynamics. Instead, mass is considered to be conserved and fission energy is treated as an energy source term.

The source of the fission energy is an increase in binding energy per nucleon of fission products in comparison to the original fuel. The difference in binding energy is released in kinetic energy of the products as well as gamma particles, neutrons, and neutrinos (Shultis and Faw, 2008). On average, one fission releases about 200 MeV. The majority of this energy is released immediately in the form of kinetic energy to the fission products, which is converted to heat in the fuel. Additional energy is thermalized in thermal shields and the pressure vessel as particles are captured, but about 15% of the fission energy completely escapes the reactor vessel and is unavailable for use in power generation.

Some of the heat generated results from the decay of fission products in the core. In an operating reactor, the decay heat and the promptly released heat are indistinguishable; however, the decay heat continues after the reactor is shut down, so it is important to have decay heat removal capacity. The decay heat results from the decay of a variety of fission products with widely varying half-lives. In a reactor that has been operating for a long time, at its peak, decay heat is about 7% of the power level immediately prior to shut down and reduces to about 1% within an hour and 0.5% in 1 day (Ragheb, 2011). Because of its high enthalpy of vaporization, water, which can be boiled, provides the highest density to

store decay heat removal capacity. The amount of water required to remove the decay heat of a reactor can be calculated from the decay heat curve:

$$m = \frac{\int \dot{Q}_{\text{DH}}(t) dt}{h_{\text{fg}}} \quad (23.16)$$

The decay heat curve,  $\dot{Q}_{\text{DH}}(t)$ , is proportional to the reactor size and is also dependent on the power history,  $h_{\text{fg}}$ , which is the enthalpy of vaporization of water. It is further discussed below.

### 23.2.5 Thermodynamic Properties

Thermodynamic *properties* are characteristics of a system (e.g., pressure, temperature, density, specific volume, enthalpy, entropy, etc.). Because properties depend only on the state of a system, they are said to be *path independent* (unlike heat and work). *Extensive properties* are mass dependent (e.g., total system energy and system mass), whereas *intensive properties* are independent of mass (e.g., temperature and pressure). *Specific properties* are intensive properties that represent extensive properties divided by the system mass, for example, specific enthalpy is enthalpy per unit mass,  $h = H/m$ . In order to apply thermodynamic balance equations, it is necessary to develop thermodynamic property relationships. Properties of certain idealized substances (incompressible liquids and ideal gases with constant specific heats) can be calculated with simple equations of state; however, in general, properties require the use of tabulated data or computer solutions of generalized equations of state.

#### 23.2.5.1 The State Postulate

According to the state postulate, if two independent properties are known for a simple compressible system (SCS), all other properties may be determined. Briefly, an SCS is a system in which the only work interaction is boundary work; magnetic, electrostatic, surface free energy, gravitational work, for example, are excluded. SCS examples include liquids and ideal gas mixtures. A vapor is generally treated as an SCS as long as surface tension energy may be neglected. The importance of the state postulate is that from any two independent properties in an SCS, it is possible to determine all other properties. For example, in the case of an ideal gas given a change in pressure and temperature, the change in volume, internal energy, and entropy may be determined. In cases where there are additional work interactions, one additional property is required to establish the state for each work interaction. For example, if gravitational work is relevant, then elevation would suffice. Total energy would depend on system internal energy and the potential energy. In the case of wet vapors with significant surface free energy, the size of the droplets would be required to define the state. In the analysis of thermodynamic cycles, the surface free energy is not normally significant.

#### 23.2.5.2 Saturation Conditions

In cases where two phases (e.g., liquid and vapor) exist together, the system is to be at saturation conditions. In thermodynamic cyclic devices, we are most often concerned with liquid–vapor mixtures (frequently referred to as *wet vapors*). Under saturation conditions,

pressure and temperature are not independent properties. For example, a mixture of water and steam at one atmosphere will have a temperature of 100°C. At higher pressure, the temperature must increase to remain at saturation conditions. Properties for liquids and vapors at saturation conditions are found in saturation tables and subscripts  $f$  and  $g$ , respectively.

Properties of wet vapors depend on the *quality* ( $X$ ), which is defined as the mass fraction of the wet mixture that is in the vapor form, based on the relationship,

$$y = y_f + Xy_{fg} \quad (23.17)$$

where:

$y$  is any property ( $s$ ,  $u$ ,  $v$ , or  $h$ )

The subscript  $fg$  is the difference between vapor and liquid properties,  $y_{fg} = y_g - y_f$

The enthalpy of vaporization ( $h_{fg}$ ), for example, is the energy that must be added at constant pressure to vaporize a liquid.

### 23.2.5.3 Gases and Superheated Vapors

Gases and superheated vapors exist at temperatures above the saturation or boiling temperature for a given pressure. Pressure and temperature are independent properties and knowledge of both, according to the state postulate, fully defines the state and all other properties.

### 23.2.5.4 Ideal Gases

Ideal gases follow the ideal gas equation of state (ideal gas law),

$$Pv = RT \quad \text{or} \quad PV = mRT \quad (23.18)$$

where  $R$  is the gas constant for the specified gas (the universal gas constant divided by the molecular weight),

$$R = \frac{\bar{R}}{MW} \quad (23.19)$$

The most common ideal gas in thermodynamic cycles is air (a mixture of mostly nitrogen and oxygen along with about 1% other components). Air has a molecular weight, based on its constituent components of 29 g/mole. There is also an increasing interest in other gases for use in high-temperature gas-cooled nuclear reactors employing a closed Brayton cycle.

According to *Joules law*, internal energy and enthalpy for ideal gases are functions of temperature only,  $u = u(T)$ , and  $h = h(T)$ . For this reason, ideal gas properties are tabulated as functions of temperature only. Entropy for an ideal gas is a function of both pressure and temperature, so property tables include  $s^o(T)$ , which is the entropy at the given temperature and one atmosphere of pressure. To determine entropy at other pressures, it is necessary to apply a correction to the tabulated values.

$$s(T, P) = s^o(T) - R \ln \left( \frac{P}{P_o} \right) \quad (23.20)$$

where  $P_0$  is one atmosphere. Thus, entropy changes are calculated,

$$s_2 - s_1 = s_2^o(T) - s_1^o(T) - R \ln \left( \frac{P_2}{P_1} \right) \quad (23.21)$$

Isentropic processes can be analyzed by determining the reference entropy. Because of the frequency of its use for air, isentropic processes can also be analyzed using the relationships,

$$\left. \frac{P_2}{P_1} \right|_{s=\text{const}} = \frac{P_{R2}}{P_{R1}} \quad (23.22)$$

$$\left. \frac{v_2}{v_1} \right|_{s=\text{const}} = \frac{v_{R2}}{v_{R1}} \quad (23.23)$$

where  $P_R$  and  $v_R$  are the *reduced pressure* and the *reduced volume* and are tabulated as functions of temperature.

Ideal gases may be analyzed assuming constant specific heats, although the results are less accurate than using tabulated data and the error increases with the change in temperature. Assuming constant specific heats, the following simplified relations may be used

$$u_2 - u_1 = C_v (T_2 - T_1) \quad (23.24)$$

$$h_2 - h_1 = C_p (T_2 - T_1) \quad (23.25)$$

$$s_2 - s_1 = C_v \ln \left( \frac{T_2}{T_1} \right) + R \ln \left( \frac{v_2}{v_1} \right) = C_p \ln \left( \frac{T_2}{T_1} \right) - R \ln \left( \frac{P_2}{P_1} \right) \quad (23.26)$$

Assuming constant specific heats, integrating Equation 23.26 generates the following relationships for isentropic processes,

$$\left. \frac{P_2}{P_1} \right|_{s=\text{const}} = \left( \frac{v_1}{v_2} \right)^k \quad (23.27)$$

$$\left. \frac{T_2}{T_1} \right|_{s=\text{const}} = \left( \frac{v_1}{v_2} \right)^{k-1} \quad (23.28)$$

$$\left. \frac{T_2}{T_1} \right|_{s=\text{const}} = \left( \frac{P_2}{P_1} \right)^{\frac{k-1}{k}} \quad (23.29)$$

where  $k = C_p/C_v$ . For air at standard conditions,  $k = 1.4$ .

### 23.2.5.5 Subcooled Liquids

*Subcooled* or *compressed liquids* exist at temperatures below the saturation or boiling temperature for a given pressure. In general, subcooled liquids exhibit little change in properties resulting from pressure changes and properties are driven primarily by temperature,

although subcooled liquids show increasing compressibility with increasing temperature and pressure. At elevated temperatures associated with light water reactors, compressibility becomes an important consideration. Properties of subcooled liquids are tabulated as  $f(P, T)$ , in tables or computerized programs.

### 23.2.5.6 Incompressible Liquids

Incompressible liquids are subcooled liquids, where it is assumed that increasing pressure has no effect on system volume. As a result, increasing pressure does no work on a system and adds no internal energy. Most properties are thus equal to the value of a liquid at saturation conditions at the same temperature.

$$u = u_{f@T} \qquad s = s_{f@T} \qquad v = v_{f@T} \qquad (23.30)$$

For enthalpy, a slight pressure correction improves the accuracy. This amounts to adding an ideal pump work term to account for the enthalpy added due to increased pressure:

$$h_{\text{scl}} \approx h_{f@T} + \underbrace{v_{f@T}(P - P_{\text{sat}@T})}_{\substack{\text{Ideal pump work} \\ \text{(very small correction)}}} \qquad (23.31)$$

A nonideal pump will increase enthalpy due to both the pressure increase and viscous heating of the fluid.

$$\Delta h = c\Delta T + v_{f@T}(P - P_{\text{sat}@T}) \qquad (23.32)$$

The change in entropy for incompressible liquids can be approximated in cases where the variation in specific heat is small,

$$\Delta s = c \ln\left(\frac{T_2}{T_1}\right) \qquad (23.33)$$

Within normal operating conditions in most engineering systems, subcooled liquids may be treated as incompressible for typical accuracy required with engineering calculations. However, the errors in assuming that liquids are incompressible increase with both increasing temperature and increasing pressure. In nuclear systems, where pressures and temperatures are both high, significant errors may result from using incompressible liquid assumptions. Fortunately, computerized property tables accurately account for compressibility and are convenient to use.

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## 23.3 Power Cycles

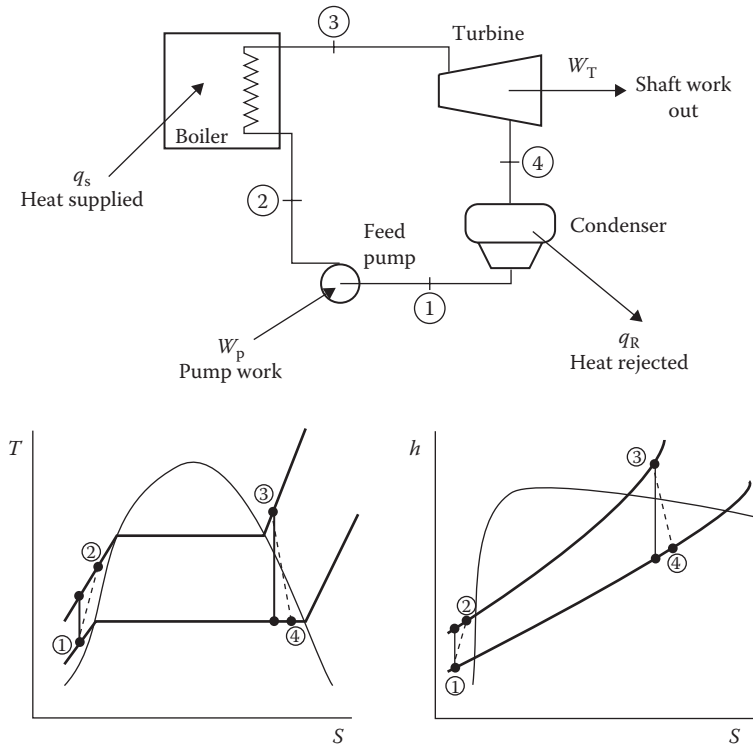
The Brayton and Rankine power cycles, which are most commonly used in power generation, are identical except that the Rankine cycle employs a vapor with a phase change and the Brayton cycle operates on a single phase gas. In both cases, there is a pressure increase process (using a pump or compressor), a heat addition process, expansion in a turbine, and a heat removal process. Nuclear power generation currently uses the Rankine cycle almost

exclusively; however, there is great interest in closed Brayton cycles for use in Generation IV reactors, both in direct cycles using high-temperature gas-cooled reactors and using indirect cycles, as would be seen in molten salt and metal-cooled reactors.

### 23.3.1 Rankine Cycle

Rankine cycles employ a vapor as the working fluid, which is typically water, although other vapors have been used where appropriate based on the temperature of the heat source. For example, geothermal power plants operating at lower temperatures frequently use refrigerant R-134a. Mercury vapor was used in several experimental plants because of its higher boiling point and it proved to be thermodynamically appealing, although its toxicity has prevented widespread commercialization.

Figure 23.3 shows the plant components and the thermodynamic cycle on  $T$ - $s$  and  $h$ - $s$  diagrams. Liquid water at point 1 is pumped into a boiler at point 2. The ideal pumping process is isentropic (constant entropy); however, real pumps generate entropy in the pumping process. Heat is supplied in a boiler from points 2 to 3 where the liquid is converted to a vapor, before expansion in a turbine from 3 to 4. Ideal expansion would be isentropic, but entropy is generated in real turbines. Finally, heat is rejected in a condenser in which the working fluid is converted back into a liquid. The outlet vapor at point 3 is superheated where practical because it is advantageous to eliminate moisture from the turbine and higher temperatures result in higher efficiencies; however, most light water reactor plants generate saturated steam at the steam generator exit. Work is supplied to a pump, which

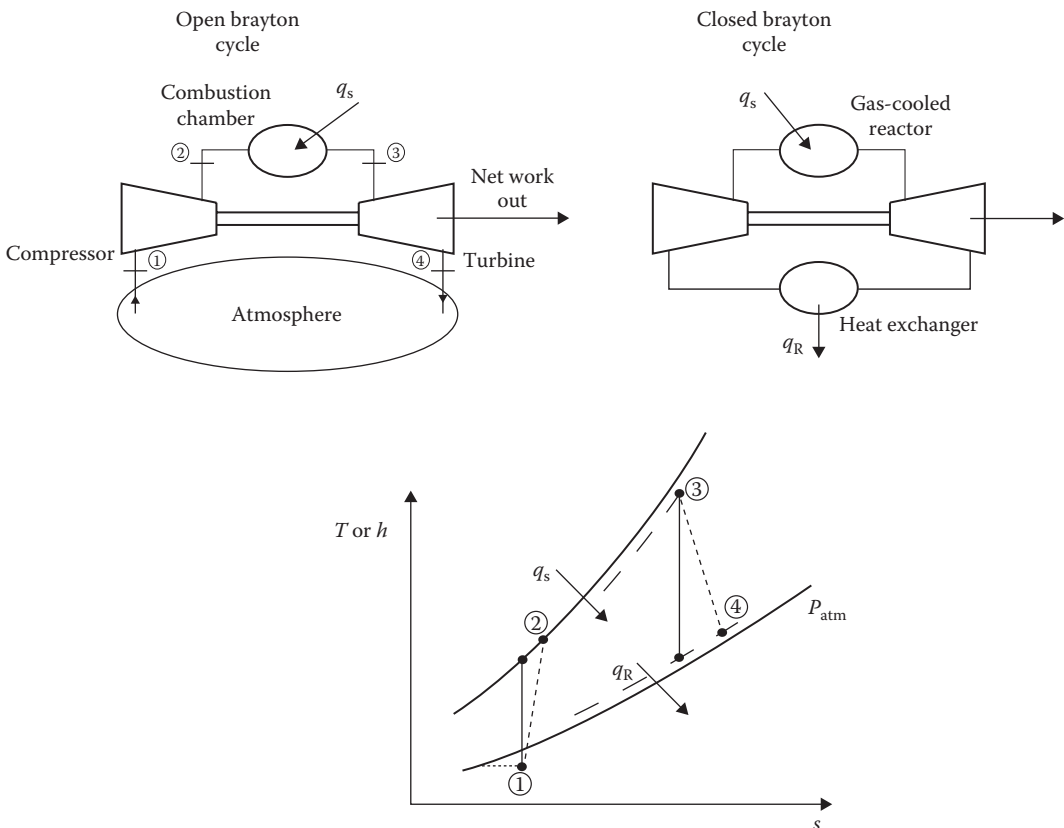


**FIGURE 23.3**  
Rankine heat engine cycle.

raises the working fluid pressure and work is extracted in a turbine. Point 1 requires some subcooling to prevent cavitation in the pump, but because excessive subcooling requires additional heat to be supplied in the boiler, it is desirable to limit the amount of subcooling. A major advantage of Rankine cycle plants is that the required pump work is very small in comparison to the turbine work, since the fluid volume is greatly decreased by condensing it. The heat source can result from fossil fuels, nuclear energy, or renewable sources, such as geothermal or concentrated solar energy. In boiling water reactors, the steam is directly heated in a nuclear reactor, while in a pressurized water reactor water under high pressure circulates through the reactor and transfers heat to the secondary coolant in a steam generator (see Section 23.3.3).

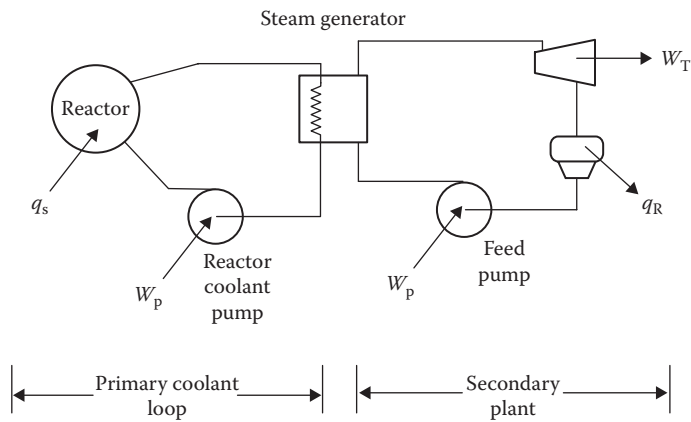
### 23.3.2 Brayton Cycle

The processes in a Brayton cycle (see Figure 23.4) are similar to a Rankine cycle, with the exception that in a Brayton cycle, the working fluid is a gas (most frequently air). However, because the pressure increase process occurs on a gas working fluid, the magnitude of the work supplied to a compressor (referred to as *back work*) is large in comparison to the pump process in a Rankine cycle. Gas turbine engines operate on an open Brayton cycle, drawing in atmospheric gas and heating it through combustion before expanding it



**FIGURE 23.4**  
Brayton heat engine cycle.




**FIGURE 23.5**

Indirect heat engine cycle. Heat is transferred to a secondary coolant, which serves as the thermodynamic working fluid.

through a turbine and discharging it to the atmosphere. The increase in the temperature of the exhausted gas can be treated thermodynamically as a heat rejection process. Closed Brayton cycles recirculate the gas to a heat exchanger to reject heat to the environment or heat recovery steam generator (HRSG). Helium is a popular coolant in high-temperature gas-cooled reactors and there is a great deal of research focused on using supercritical carbon dioxide in closed Brayton cycles for a wide variety of applications. The accompanying process diagram represents the cycle on either a  $T$ - $s$  diagram or an  $h$ - $s$  diagram, which look identical since enthalpy is a function of temperature only in an ideal gas.

### 23.3.3 Indirect Cycles

In an indirect cycle, heat is transferred from a primary coolant to a secondary system where the thermal to mechanical conversion takes place, such as the pressurized water reactor (PWR) plant shown in Figure 23.5. Indirect cycles have the advantage of separating the working fluid from activation in the reactor. However, they add complexity to the design, and heat transfer in the steam generator destroys exergy. Currently, about two-thirds of the nuclear power generated in the United States takes place in PWR plants, while the balance takes place in boiling water reactor (BWR) direct cycle plants.

The critical temperature of water limits its temperature in PWRs, which in turn limits efficiency. To overcome this limitation, other primary coolants such as liquid metals, which are capable of operating at higher temperatures, have been used.

## 23.4 Thermodynamic Conservation and Balance Equations

Neglecting relativistic effects, the total quantities of mass and energy are constant, or conserved, although it is possible to convert energy from one form to another. Therefore, it is possible to write mass and energy conservation relationships. Entropy, on the other hand, is generated in real processes. Thus, an entropy balance equation must include an entropy

source term, which is always positive. Exergy, which represents the useful potential to perform work, decreases as entropy is generated. An exergy balance equation includes an exergy destruction or *irreversibility* term, which is related to entropy generation. In this section, balance equations for mass, energy, entropy, and exergy are developed.

### 23.4.1 Mass Balance Equation (Conservation of Mass)

Mass conservation can be written in a general form for an open system (see Figure 23.2) as,

$$\left[ \begin{array}{l} \text{The rate of mass} \\ \text{flow into a system} \\ \text{from all inlets} \end{array} \right] - \left[ \begin{array}{l} \text{The rate of mass} \\ \text{flow out of a system} \\ \text{through all outlets} \end{array} \right] = \left[ \begin{array}{l} \text{The time rate of} \\ \text{change of mass stored} \\ \text{within the system} \end{array} \right] \quad (23.34)$$

Equation 23.34 may be written symbolically as,

$$\sum_{\text{inlets}} \dot{m} - \sum_{\text{outlets}} \dot{m} = \frac{dm}{dt} \quad (23.35)$$

where:

$m$  is the mass of a system

$\dot{m}$  is a mass flow rate

The mass flow rate is related to density ( $\rho$ ), fluid velocity ( $V$ ), flow area ( $A$ ), and volumetric flow rate ( $\dot{V}$ ).

$$\dot{m} = \rho VA = \rho \dot{V} \quad (23.36)$$

The velocity used here is an average velocity across the cross section,

$$V = \frac{\int \rho \mathbf{V} \cdot \hat{\mathbf{n}} dA}{\int \rho dA} \quad (23.37)$$

In cases of single-phase flow, the density can generally be assumed constant across the cross section, although it can vary greatly in cases of two-phase flow.

### 23.4.2 Energy Balance Equation (First Law of Thermodynamics)

The first law of thermodynamics, which predates the theory of relativity, states that the total quantity of energy is conserved. It is convenient to treat energy released from nuclear reactions as a volumetric heat generation source term and include it with heat transferred into the system. For an open system (see Figure 23.2) undergoing a transient, the first law may be written

$$\left[ \begin{array}{l} \text{The net rate} \\ \text{of heat} \\ \text{transfer in} \end{array} \right] - \left[ \begin{array}{l} \text{The power} \\ \text{produced by} \\ \text{the system} \end{array} \right] + \left[ \begin{array}{l} \text{The net rate that energy} \\ \text{is brought into the system} \\ \text{by mass flow} \end{array} \right] = \left[ \begin{array}{l} \text{The net rate} \\ \text{change of energy} \\ \text{in the system} \end{array} \right] \quad (23.38)$$

The power produced by the system includes shaft power and the rate of boundary work ( $P dV/dt$ ). Typically, shaft power includes all nonboundary work including electrical power. The rate of boundary work is zero for either steady-state conditions or for any system with a rigid boundary such as a pump or a turbine. Flow streams that cross the system boundary carry their associated energy ( $e = u + pe + ke$ ). Streams entering the system perform flow work on the system at a rate of  $\dot{m}pv$  and the system performs flow work on exiting streams. Substituting the definition of enthalpy ( $h = u + pv$ ) results in a general expression for the first law of thermodynamics:

$$\dot{Q} - \dot{W}_{\text{shaft}} - P \frac{dV}{dt} + \sum_{\text{Inlets}} \dot{m}(h + pe + ke) - \sum_{\text{Outlets}} \dot{m}(h + pe + ke) = \frac{d}{dt} m(u + pe + ke) \quad (23.39)$$

For a closed system undergoing a process over an interval of time, Equation 23.39 may be written as,

$$q - w = \Delta e = \Delta u + \Delta pe + \Delta ke \quad (23.40)$$

where the  $\Delta$  represents the change in a property over time. Frequently potential and kinetic energy are not significant, so Equation 23.40 can be further simplified as

$$q - w = \Delta u \quad (23.41)$$

In steady flow systems, boundary work and the energy storage term are eliminated, so that Equation 23.39 simplifies to,

$$\dot{Q} - \dot{W} = \sum_{\text{out}} \dot{m}(h + pe + ke) - \sum_{\text{in}} \dot{m}(h + pe + ke) \quad (23.42)$$

For a single inlet and outlet, the mass flow rate may be divided out of Equation 23.42 using Equations 23.13 and 23.14, producing the steady flow energy equation for single inlet and single outlet systems:

$$q - w = \Delta h + \Delta pe + \Delta ke \quad (23.43)$$

Here the  $\Delta$  represents the difference between exit and entrance properties. The  $ke$  term is significant in analyzing the internals of a turbine, but at the component level in power plant analysis,  $pe$  and  $ke$  are not normally significant resulting in a simplified expression,

$$q - w = \Delta h \quad (23.44)$$

### 23.4.3 Entropy Balance Equation (Second Law of Thermodynamics)

The second law of thermodynamics dictates that certain processes are irreversible. For example, heat travels in a direction of decreasing temperature. There are two commonly cited equivalent qualitative statements to the second law. The *Kelvin–Planck statement* states that it is impossible to construct any cyclic device that receives heat from a single thermal reservoir and converts it entirely into work. According to the *Clausius statement*, it is impossible to construct a device, which operates in a cycle that produces no other effect on the environment other than the transfer of heat from a low temperature reservoir to a higher temperature reservoir. Both of the Kelvin–Planck and Clausius statements of the second

law can be proven from the increase in entropy principle, which states that the *total entropy change for any process is always greater than or equal to zero.*

$$\Delta S|_{\text{Total}} \geq 0 \tag{23.45}$$

The implication of the word *total* is that it is possible for entropy of a system to decrease when heat is transferred out; however, the increase in entropy outside the system will be greater than the entropy reduction within the system. Physically, entropy is a measure of the amount of energy that is not available to produce work. Work and mechanical energy are fully useable and therefore have no associated entropy and any process that generates entropy is irreversible.

The formal definition of entropy is the integral of the heat added divided by the absolute temperature along any internally reversible path between two states:

$$\Delta S = \int \frac{\delta Q}{T} \Big|_{\text{Int Rev}} \tag{23.46}$$

In the case where a system is isothermal, the entropy associated with heat transfer to or from a heat source or heat sink at a temperature  $T$  is,

$$\Delta S = \frac{Q}{T (\text{°R or K})} \tag{23.47}$$

Fortunately, it is not normally necessary to select an internally reversible path and carry out this integration of Equation 23.46, because entropy is a property. Based on the state postulate, entropy can be determined from changes in other properties.

The entropy balance equation may be written as,

$$\left[ \begin{array}{l} \text{The net rate} \\ \text{that entropy is} \\ \text{transferred in with} \\ \text{heat transfer} \end{array} \right] + \left[ \begin{array}{l} \text{The} \\ \text{rate of} \\ \text{entropy} \\ \text{generation} \end{array} \right] + \left[ \begin{array}{l} \text{The net rate that} \\ \text{entropy is brought} \\ \text{into the system} \\ \text{by mass flows} \end{array} \right] = \left[ \begin{array}{l} \text{The net rate} \\ \text{change of} \\ \text{entropy} \\ \text{in the system} \end{array} \right] \tag{23.48}$$

Symbolically,

$$\sum_j \frac{\dot{Q}_j}{T_j} + \dot{S}_{\text{Gen}} + \sum_{\text{Inlets}} \dot{m}s - \sum_{\text{Outlets}} \dot{m}s = \frac{dS}{dt} \tag{23.49}$$

The entropy transferred in is dependent on the temperature at the system boundary ( $T_j$ ). If heat is transferred from a higher temperature source, moving the system boundary to the edge of the heat source would raise  $T_j$  and reduce entropy transferred in; however, the difference would increase the  $\dot{S}_{\text{Gen}}$  within the system. In the case of a single flow system at steady state, Equation 23.49 simplifies to

$$s_{\text{Gen}} = s_2 - s_1 - \frac{q}{T} \tag{23.50}$$

### 23.4.4 Exergy Balance Equation

Although first law analysis is relatively simple to master and easy to conceptualize, it has the disadvantage of considering only the magnitude of the energy transfer and assuming that all forms of energy have the same usefulness. In reality, large quantities of low temperature heat have very little potential to perform useful work. Many processes, which are very *efficient* from a first law standpoint, such as using electrical energy to heat low temperature water, are actually very wasteful. Second law analysis, which integrates first law analysis with entropy considerations, considers the potential of energy to do useful work and clearly identifies the source of irreversibility (lost available work). In that manner, it is possible to appropriately focus attention on components that have the greatest impact on performance.

*Exergy* (also called availability and available energy) is defined as the useful work that could be produced by reducing a system to the dead state (at equilibrium with the environment). The standard dead state is 537°R (298.15K) and 14.696 psi (1.01325 bars), with zero velocity and zero potential energy. The notation for exergy used in this chapter is  $\Phi$  and  $\phi$  for exergy and specific exergy and  $\psi$  for flow exergy. Nonflow exergy is a property that depends on the system properties and the dead state and can be calculated from the relationship (Moran and Shapiro, 2008),

$$\phi = (e - u_o) + P_o(v - v_o) - T_o(s - s_o) \quad (23.51)$$

where the  $o$  subscript represents the properties of the dead state. For a closed system, any deviation from the dead state, results in a potential to do work, so  $\phi$  is always positive. Similarly, *flow exergy* is calculated (Moran and Shapiro, 2008),

$$\psi = (h - h_o) - T_o(s - s_o) + ke + pe \quad (23.52)$$

In each case, the kinetic and potential energy of the dead state are zero. Unlike  $\phi$ ,  $\psi$  can be negative, indicating that work would be required to return a fluid to the dead state.

The principle of exergy can be illustrated by considering a source of heat ( $Q$ ) at a temperature  $T$ . The exergy of the heat ( $\Phi_H$ ) is the useful work that could be produced in a reversible heat engine if the heat source were used in an engine that rejected heat to the environment at  $T_o$ . Thus, the exergy is the magnitude of the heat supplied multiplied by the Carnot efficiency.

$$\Phi_H = W_{\max} = Q \left( 1 - \frac{T_o}{T} \right) \quad (23.53)$$

When heat is transferred by conduction to a lower temperature (say from  $T_1$  to  $T_2$ ), its exergy is reduced, though the total quantity of energy is conserved:

$$\Phi_{\text{Lost}} = Q \left( 1 - \frac{T_o}{T_1} \right) - Q \left( 1 - \frac{T_o}{T_2} \right) \quad (23.54)$$

Combining with the entropy balance, Equation 23.50, and defining destroyed exergy as *irreversibility* gives,

$$I = T_o S_{\text{Gen}} \quad \text{or} \quad i = T_o s_{\text{Gen}} \quad (23.55)$$

A more generalized derivation proves that the relation between irreversibility and entropy generation, Equation 23.55, is always valid (Moran, 1982). On a rate basis,

$$\dot{I} = T_0 \dot{S}_{Gen} \tag{23.56}$$

The exergy balance equation is very similar to the first law of thermodynamics with the exception that useful energy is considered instead of total energy. An important difference is that exergy is not a conserved quantity; it is destroyed as entropy is generated.

$$\begin{aligned} & \left[ \begin{array}{l} \text{The net rate} \\ \text{that heat exergy} \\ \text{is transferred} \\ \text{into the system} \end{array} \right] - \left[ \begin{array}{l} \text{The useful} \\ \text{power} \\ \text{produced by} \\ \text{the system} \end{array} \right] + \left[ \begin{array}{l} \text{The net rate that} \\ \text{exergy is brought} \\ \text{into the system} \\ \text{by mass flows} \end{array} \right] \\ & - \left[ \begin{array}{l} \text{The net rate that} \\ \text{exergy is} \\ \text{destroyed by} \\ \text{irreversibility} \end{array} \right] = \left[ \begin{array}{l} \text{The net rate} \\ \text{change of} \\ \text{exergy} \\ \text{in the system} \end{array} \right] \end{aligned} \tag{23.57}$$

Symbolically,

$$\dot{\Phi}_H - \dot{W}_{Useful} + \sum_{Inlets} \dot{m}\psi - \sum_{Outlets} \dot{m}\psi - \dot{I} = \frac{d\Phi}{dt} \tag{23.58}$$

For a closed system for a given time interval, the exergy balance simplifies to

$$\Phi_{Heat} - W_{Useful} - I = \Phi_2 - \Phi_1 \tag{23.59}$$

which is analogous to the closed system energy balance,  $Q - W = E_2 - E_1$ , excluding the irreversibility term. Similarly, the single inlet and single outlet steady-state exergy balance can be written on a per unit mass flow basis,

$$\phi_{Heat} - w - i = \psi_2 - \psi_1 \tag{23.60}$$

### 23.5 Thermodynamic Analysis

Thermodynamic analysis of power plants seeks to characterize efficiency and identify sources of losses. First law analysis assesses performance based on energy balance equations, while second law analysis uses exergy balances and looks for locations of exergy destruction. In this section, analysis methods are developed to apply thermodynamic balance equations to analyze heat engines and power plant components. Results are summarized in Appendix B of this chapter and detailed examples are provided in Section 23.6.

First law efficiencies, which measure energy utilization, are ratios of energy output to the energy supplied.

$$\eta_I = \frac{\text{Useful energy out}}{\text{Energy supplied}} = 1 - \frac{\text{Energy lost}}{\text{Energy supplied}} \quad (23.61)$$

Although simple to employ and easily understood by a wide audience, first law analysis frequently fails to identify areas of waste and opportunities for improvement. For example, a residential electric space heater has a first law efficiency of 100% since the electrical input is converted entirely into heat in the room. However, a typical heat pump has a coefficient of performance of 4.5, which means it supplies over four times more heat to the occupied space with the same electrical energy input as the comparable resistance heater. Second law analysis overcomes this shortfall and allows the engineer to focus on areas where the greatest impact can be achieved.

There are two widely employed second law analysis methods: *exergy analysis* and *entropy generation minimization*. Exergy analysis uses exergy balances to track where exergy is lost or destroyed. Because entropy generation is proportional to exergy destruction, in entropy generation minimization, locations of entropy generation are identified. Second law efficiencies are the fraction of exergy that survives a process (Moran, 1982):

$$\eta_{II} = \frac{\text{Exergy out}}{\text{Exergy supplied}} = 1 - \frac{\text{Irreversibility}}{\text{Exergy supplied}} \quad (23.62)$$

The value of exergy analysis is evident by comparing the results of Rankine and Brayton cycle power plants in Section 23.6. Example 2 (Section 23.6.2) shows the results of first and second law analysis of a Rankine steam plant. As the example shows, a large fraction of the input heat is rejected in the condenser and the first law boiler efficiency is very high (100% in the case of a nuclear steam generator). By itself, this result would allow the engineer to look for ways to harness the rejected heat. However, second law analysis shows that the heat rejected in the condenser has very little exergy, and although energy is conserved in the steam generator, a tremendous amount of exergy is destroyed as heat is transferred to a lower temperature. Armed with this information, the engineer can look to identify ways to reduce the boiler exergy destruction. Most commonly, feedwater heaters are used to reduce the heat transfer temperature difference. Example 3 (Section 23.6.3) demonstrates that a single feedwater heater improves energy utilization by over 10%.

On the other hand, second law analysis of a Brayton cycle (Example 5; Section 23.6.5) reveals a high exergy content of the turbine exhaust. For this reason, great improvements in gas turbine plant efficiency can be achieved by recovering exergy in the turbine outlet with either recuperation (Example 7; Section 23.6.7) or through the use of a heat recovery boiler used to supply a secondary steam cycle in a cogeneration plant (Example 6; Section 23.6.6).

### 23.5.1 Overall Power Plant Analysis

The thermal efficiency measures the efficiency of a heat engine in converting a heat input into shaft work,

$$\eta_{th} = \frac{\dot{W}_{NET}}{\dot{Q}_S} = 1 - \frac{\dot{Q}_R}{\dot{Q}_S} = \frac{w_{NET}}{q_S} = 1 - \frac{q_R}{q_S} \quad (23.63)$$

Typically, the thermal efficiency accounts for only the thermal to mechanical conversion process within the engine. Mechanical efficiency accounts for frictional losses within a heat engine. In an internal combustion engine, the power generated within the engine is known as the indicated horsepower (IHP) and the output power is the brake horsepower (BHP). The two are related by the mechanical efficiency (Heywood, 1988),

$$BHP = \eta_{\text{Mech}} IHP \quad (23.64)$$

Where energy conversions occur sequentially, overall efficiency is the product of the various process efficiencies in the production of the output product. For example, in a conventionally fueled steam plant, where chemical energy is converted to heat in a boiler, heat is converted to work in a heat engine, and shaft work is converted to electrical work in a generator, the overall efficiency would be

$$\eta_{\text{OA}} = \frac{\dot{W}_{\text{Electrical}}}{\dot{Q}_{\text{Fuel}}} = \eta_{\text{Boiler}} \cdot \eta_{\text{th}} \cdot \eta_{\text{Generator}} \quad (23.65)$$

Heat rate (a dimensional reciprocal of efficiency with units of Btu/kWh) is frequently used in the US power industry:

$$HR = \frac{Q_{\text{Supplied}} (\text{Btu})}{W_{\text{Elec}} (\text{kW} \cdot \text{hr})} = \frac{3412.14}{\eta} \quad (23.66)$$

Based on the increase in entropy principle, it can be proven that the maximum theoretical thermal efficiency for any heat engine is equal to the Carnot efficiency,

$$\eta_{\text{CARNOT}} = 1 - \frac{T_L}{T_H} \quad (23.67)$$

To account for this variation in the *quality* of the source heat, the second law efficiency is frequently defined as the fraction of the exergy input converted into work,

$$\eta_{II} = \frac{\dot{W}_{\text{NET}}}{\dot{\Phi}_{\text{HEAT}}} = \frac{\eta_{\text{TH}}}{\eta_{\text{CARNOT}}} \quad (23.68)$$

### 23.5.2 Reactors, Combustion Chambers, and Boilers

Heat engines use a thermal energy source that is transferred to a working fluid in a combustion chamber, boiler, or reactor. Energy and exergy diagrams for this process are shown in Figure 23.6. The rejected heat ( $\dot{Q}_R$ ) represents stack losses in a fossil fuel boiler.

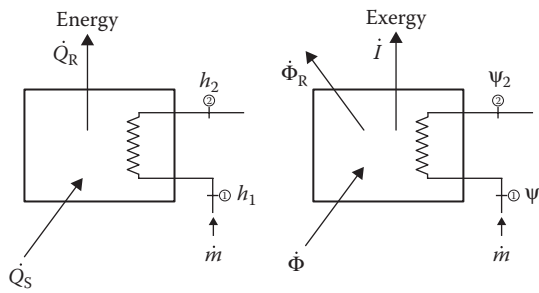
Normalizing by  $\dot{m}$  to develop relationships on a per unit mass basis (e.g.,  $\dot{\Phi} = \dot{m}\phi$ ), the resultant energy and exergy balance equations are

$$q_S - q_R = h_2 - h_1 \quad (23.69)$$

$$\phi_H - \phi_R - i = \psi_2 - \psi_1 \quad (23.70)$$

The irreversibility ( $i$ ) represents exergy destruction within the component, whereas the rejected exergy ( $\phi_H$ ) leaves the component. The distinction between these two terms is that  $\phi_H$  could potentially be recovered externally, whereas  $i$  is destroyed internally. In cases





**FIGURE 23.6** Energy and exergy process diagrams for a reactor, combustion chamber, or boiler.

where the stack losses are not recovered for useful purposes (typical of most applications),  $\phi_H$  and  $i$  can be lumped together and the first and second law efficiencies are

$$\eta_I = \frac{h_2 - h_1}{q_s} \tag{23.71}$$

$$\eta_{II} = \frac{\Psi_2 - \Psi_1}{\phi_H} \tag{23.72}$$

In cases where the heat is generated by combustion, first law efficiencies can also be defined in terms of the heating value of the fuel to account for unburnt fuel. Since, in comparison to the heat produced, relatively little heat is lost to ambient from nuclear reactors, first law reactor efficiencies are close to 100%. However, lowering temperatures associated with the internal heat transfer destroys exergy and reduces second law efficiencies. The exergy destroyed is,

$$i = \phi_H - (\psi_2 - \psi_1) \tag{23.73}$$

The exergy supplied from the heat source ( $\phi_H$ ) is dependent on assumptions and judgment. The combustion process is a highly irreversible process, which destroys much of the chemical exergy ( $\phi_{Chm}$ ) in the fuel. Conventionally fueled thermal power plants convert chemical energy to heat, which is in turn converted to work. The chemical exergy contained in a fuel is generally close to its heating value (Wark, 1995); however, practical fuel cells capable of harnessing  $\phi_{Chm}$  of most fuels have not been developed. For this reason, chemical energy is typically combusted to produce heat, which is then converted into work in a conventional heat engine, so it may be appropriate to consider the source exergy to be heat at an appropriate temperature.

$$\phi_H = q \left( 1 - \frac{T_o}{T_H} \right) \tag{23.74}$$

Thus, the chemical conversion process destroys exergy

$$i = \phi_{Chm} - \phi_H \approx \frac{T_o}{T_H} \tag{23.75}$$

Material considerations limit boiler and combustion chamber temperatures resulting in further exergy destruction. Similarly, nuclear exergy cannot be directly harnessed; instead, fission is used to generate thermal energy that is converted in a heat engine. In the process, some of the fission energy escapes from the nuclear reactor and the temperature of the fission reaction is limited by material considerations in the reactor. In analyzing a nuclear reactor, the input exergy can be considered, which is the fission energy that remains in the core at a temperature consistent with the material limitations.

Irreversibility could also be calculated from Equation 23.55. Assuming no stack losses, the entropy generated must account for the entropy transferred out of the heat source,

$$s_{Gen} = s_2 - s_1 - \frac{q}{T_H} \tag{23.76}$$

### 23.5.3 Turbines

For a turbine operating at steady state (Figure 23.7), potential and kinetic energy in the flow streams may normally be neglected so energy balance of Equation 23.44 is

$$w = h_1 - h_2 \tag{23.77}$$

Isentropic efficiencies (first law efficiencies) compare the actual turbine to an isentropic (ideal) turbine operating between the same inlet and outlet pressure,

$$\eta_T = \frac{\text{Actual work}}{\text{Ideal work}} = \frac{w_R}{w_s} = \frac{h_1 - h_2}{h_1 - h_{2s}} \tag{23.78}$$

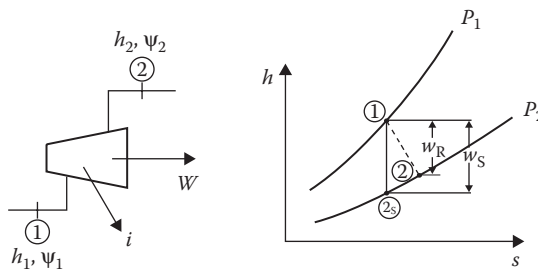
where point 2s is the point that would be reached in an isentropic process ending at the same final pressure.

The exergy balance of Equation 23.60

$$w = \psi_1 - \psi_2 - i \tag{23.79}$$

Irreversibility could also be calculated from Equation 23.55 and noting that  $s_{Gen} = s_2 - s_1$ . Turbine second law efficiencies are,

$$\eta_{II} = \frac{h_1 - h_2}{\psi_1 - \psi_2} = 1 - \frac{T_0 (s_2 - s_1)}{\psi_1 - \psi_2} \tag{23.80}$$



**FIGURE 23.7**  
Typical turbine process and  $h$ - $s$  diagram.

For a turbine,  $\eta_{II}$  is generally higher than  $\eta_I$  because some of the unextracted work remains available as exergy in the exit stream, which can be recovered in later processes.

### 23.5.4 Pumps and Compressors

Because pumps and compressors are work-consuming devices, the sign convention is normally reversed from the generalized balance equations in Section 23.4, such that positive work is defined as into the system (see Figure 23.8). For a pump or compressor operating at steady state where potential and kinetic energy of the inlet and exit streams is neglected, the energy balance of Equation 23.44 simplifies to

$$w = h_2 - h_1 \tag{23.81}$$

Entropy and exergy balances from Equations 23.50 and 23.60 yield

$$s_{Gen} = s_2 - s_1 \tag{23.82}$$

and

$$w = \psi_2 - \psi_1 + i \tag{23.83}$$

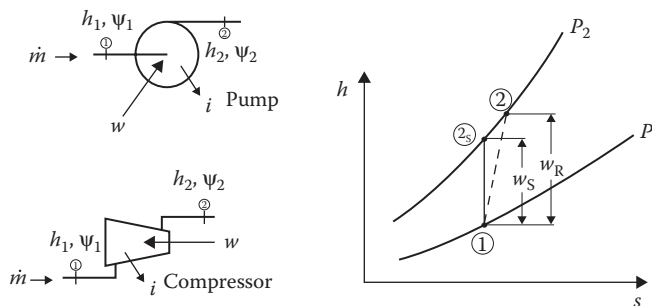
Similar to turbines, first and second law efficiencies are,

$$\eta = \frac{\text{Ideal work}}{\text{Actual work}} = \frac{w_s}{w_R} = \frac{h_{2s} - h_1}{h_2 - h_1} \tag{23.84}$$

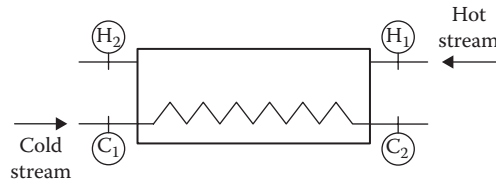
and

$$\eta_{II} = \frac{\psi_2 - \psi_1}{h_2 - h_1} = 1 - \frac{i}{h_2 - h_1} \tag{23.85}$$

As in the case of a turbine,  $\eta_{II}$  is generally higher than  $\eta_I$  because the additional work supplied raises the exergy of the exit stream. It is possible to recover this exergy in later processes. For example in a Brayton cycle plant, efficiency less than unity results in a higher compressor exit temperature and enthalpy, which reduces the heat requirements of the combustion chamber.



**FIGURE 23.8** Typical pump and compressor process and  $h$ - $s$  diagrams.



**FIGURE 23.9**  
Closed (noncontact) heat exchanger.

### 23.5.5 Closed (Noncontact) Heat Exchangers

Noncontact heat exchangers, which transfer heat between fluids through a conductive barrier (Figure 23.9), are used in a wide variety of applications, including nuclear steam generators, condensers, and closed feedwater heaters.

Assuming no heat loss to the environment and insignificant potential and kinetic energy, the energy balance from Equation 23.42 becomes

$$\dot{m}_H (h_{H1} - h_{H2}) = \dot{m}_C (h_{C2} - h_{C1}) \quad (23.86)$$

where the subscripts  $H$  and  $C$  represent the hot and cold fluids and 1 and 2 are inlet and exit points of the respective streams. Similarly, the exergy balance from Equation 23.58 results in

$$\dot{m}_H (\psi_{H1} - \psi_{H2}) = \dot{m}_C (\psi_{C2} - \psi_{C1}) + \dot{I} \quad (23.87)$$

where

$$\dot{I} = T_o \dot{S}_{\text{Gen}} = T_o [\dot{m}_C (s_{C2} - s_{C1}) + \dot{m}_H (s_{H2} - s_{H1})] \quad (23.88)$$

By definition, since energy is conserved, the first law efficiency is unity ( $\eta_I = 1$ ). The second law, whose efficiency (sometimes referred to as effectiveness) accounts for the exergy destroyed, can be calculated using one of two alternate definitions (Li and Priddy, 1985)

$$\eta_{II} = \frac{\dot{m}_C (\psi_{C2} - \psi_{C1})}{\dot{m}_H (\psi_{H1} - \psi_{H2})} \quad (23.89)$$

or

$$\eta_{II} = \frac{\dot{m}_C \psi_{C2} + \dot{m}_H \psi_{H2}}{\dot{m}_C \psi_{C1} + \dot{m}_H \psi_{H1}} \quad (23.90)$$

Combining heat transfer considerations with exergy analysis reveals a number of important engineering tradeoffs in heat exchanger design. Heat transfer across the finite temperature difference generates entropy and consequently destroys exergy. Consider a heat transfer at a rate of  $\dot{Q}$  from  $T_1$  to  $T_2$ . The rate of exergy destruction is then,

$$\dot{I} = T_o \dot{S}_{\text{GEN}} = T_o \dot{Q} (1/T_2 - 1/T_1) \quad (23.91)$$

Thus by reducing the temperature difference, the rate of exergy destruction is also reduced. The temperature difference is related to heat transfer rate by the relationship,

$$\dot{Q} = \hat{U}A\Delta T \quad (23.92)$$

where  $\hat{U}A$  is the product of the overall heat transfer coefficient and the heat transfer area.

If the temperature of either fluid in a heat exchanger varies,  $\Delta T$  is the product of the log mean temperature difference and geometry-dependent configuration factor ( $\Delta T = F\theta_{LM}$ ). The configuration factor is unity for parallel and counterflow heat exchangers or if the temperature of either fluid is constant. In the case of steam generators, condensers, boilers, and feed heaters, where most of the heat transfer results in the phase change of one of the fluids,  $F$  is close to unity. In cases where both temperatures vary, the configuration for a variety of heat exchanger geometries is available in the literature (Rohsenow et al., 1998). The log mean temperature difference is defined as (Incropera and DeWitt, 2002)

$$\theta_{LM} = \frac{\theta_1 - \theta_2}{\ln(\theta_1/\theta_2)} \quad (23.93)$$

where  $\theta$  is based on the temperatures at a given position in the heat exchanger. In the case of a counterflow heat exchanger  $\theta_1 = T_{H,in} - T_{C,out}$  and  $\theta_2 = T_{H,out} - T_{C,in}$ . For a parallel flow heat exchanger,  $\theta_1 = T_{H,in} - T_{C,in}$  and  $\theta_2 = T_{H,out} - T_{C,out}$ . For all geometries other than a parallel flow,  $\theta_{LM}$  is defined in terms of a counterflow heat exchanger temperature differences.

Solving the above relationships for heat exchangers where  $\hat{U}A$  is known and it is necessary to find  $\dot{Q}$  requires an iterative solution. However, the heat exchanger effectiveness can be determined from the *effectiveness-NTU* method (Rohsenow et al., 1998). The effectiveness is defined as,

$$\varepsilon = \frac{\dot{Q}}{[\min(\dot{m}C_p)](T_{H,in} - T_{C,in})} \quad (23.94)$$

where  $\min(\dot{m}C_p)$  represents the minimum value of the product of mass flow rate and specific heat between the two flow streams. In cases where there is a phase change  $\dot{m}C_p = \infty$  for the given flow stream. The number of transfer units is defined as,

$$NTU = \frac{\hat{U}A}{\min(\dot{m}C_p)} \quad (23.95)$$

If one fluid remains at constant temperature (i.e., in cases of a phase change),  $\varepsilon$  can be determined from,

$$\varepsilon = 1 - \exp(-NTU) \quad (23.96)$$

For other conditions, expressions for  $\varepsilon = f(NTU)$  are available in the literature.

Although the term  $\hat{U}A$  frequently appears in engineering literature, the internal and external heat transfer area can be different, especially when surfaces with fins are employed. A more general relationship notes that the product  $\hat{U}A$  results from a series of thermal resistances:

$$\hat{U}A = \frac{1}{R_{\text{Total}}}; \quad R_{\text{Total}} = \sum_j R_j = R_{\text{Cond1}} + R_{\text{Conv1}} + \dots \quad (23.97)$$

For a heat exchanger, there are convective resistances on the interior and exterior pipe surfaces and conduction resistances through the pipe wall. Additionally, it is customary to include resistances that result from the buildup of scaling products and corrosion (Incropera and DeWitt, 2002).

$$R_{\text{Total}} = \underbrace{\frac{1}{(\eta_o \hat{h}A)_C} + \frac{R''_{f,C}}{(\eta_o A)_C}}_{\substack{\text{Overall cold side film} \\ \text{resistance—including} \\ \text{fins in addition to} \\ \text{fouling and corrosion}}} + R_W + \underbrace{\frac{1}{(\eta_o \hat{h}A)_H} + \frac{R''_{f,H}}{(\eta_o A)_H}}_{\substack{\text{Overall hot side film} \\ \text{resistance—including} \\ \text{fins in addition to} \\ \text{fouling and corrosion}}} \quad (23.98)$$

$R''_f$  is the fouling and corrosion resistance  
 $\eta_o$  is the overall efficiency of the finned surface  
 $\hat{h}$  is the convective heat transfer film coefficient  
 $A$  is the heat transfer area of the appropriate side of the array  
 $R_W$  is the appropriate wall resistance

$R_W$  may be calculated from planar or cylindrical geometry relationships, depending on the tube thickness. For thin tubes the planar relationship works well,

$$R_W = \frac{L}{kA} \quad (23.99)$$

For thick tubes, accuracy is improved with the cylindrical relationship (Incropera and DeWitt, 2002)

$$R_W = \frac{\ln(D_o/D_i)}{2\pi kL} \quad (23.100)$$

The overall surface efficiency of the finned surface can be calculated based on the fin efficiency

$$\eta_o = 1 - \frac{A_f}{A}(1 - \eta_f) \quad (23.101)$$

where the fin efficiency is

$$\eta_f = \frac{q_f}{q_{f,\text{max}}} = \left[ \frac{\text{Fin heat transfer}}{\text{Fin heat transfer rate if fin had infinite conductivity}} \right] \quad (23.102)$$

$A_f/A$  is the fraction of the total area that is on the fins.

Fin efficiency, fouling resistance, and heat transfer coefficients are available in the heat transfer literature.

Tradeoffs occur in the selection of tube size, thickness, and material, as well as heat exchanger configuration. For example, larger tube diameters increase turbulence for a

given flow velocity, thus increasing the heat transfer coefficient, but larger diameters result in reduced heat transfer area for a given flow area. Additionally, larger diameter tubes must be thicker, increasing the conductive resistance. Material selection requires the need to balance conductivity, cost, strength, corrosion resistance, and nuclear properties such as nuclear cross section and potential for activation. Based on these considerations, the materials selected in power plants frequently have relatively poor heat transfer properties.

The convective resistance can be reduced by a variety of methods including enhancing turbulence level or increasing the heat transfer area with fins. In the case of gases, fins are especially important because of the low values of  $\hat{h}$ .

### 23.5.6 Open (Direct Contact) Heat Exchangers

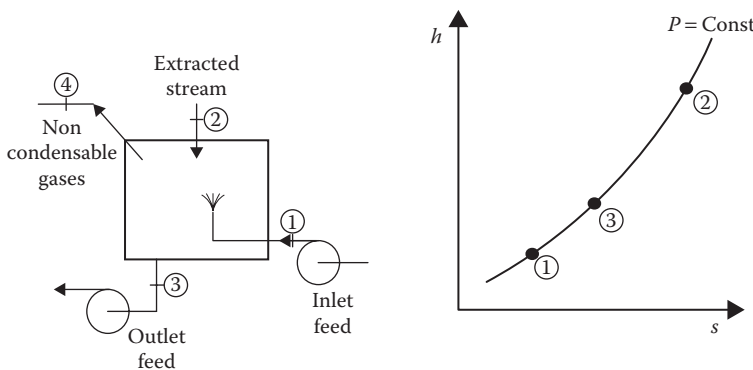
An open heat exchanger is a mixing chamber in which a hot fluid is mixed with a fluid at lower temperatures. As an example, open feed heaters (Figure 23.10) raise the temperature of boiler feed by mixing it with extracted steam. They have the added benefit of removing noncondensable gases from the feed stream, but require additional pumps to be installed in the system. Power plants typically have one open feedwater heater.

Because heat loss to the environment is very small in comparison to the heat transferred within an open heat exchanger, the first law efficiency is close to unity. However, mixing streams is an irreversible process that generates entropy and thus destroys exergy. Assuming no heat loss to the environment, using Equation 23.42 yields the energy balance of the open feed heater,

$$\dot{m}_1 h_1 + \dot{m}_2 h_2 = \dot{m}_3 h_3 + \dot{m}_4 h_4 \tag{23.103}$$

Typically,  $\dot{m}_4$  (the vent flow rate for noncondensable gases) is small, so that the energy and exergy contributions in the vent stream are small and may be neglected in the analysis. Excluding the vent stream under steady-state conditions, the exergy balance from Equation 23.58 yields,

$$\dot{m}_1 \psi_1 + \dot{m}_2 \psi_2 = \dot{m}_3 \psi_3 + \dot{I} \tag{23.104}$$



**FIGURE 23.10**  
Open (direct contact) feedwater heater.

Here,  $\dot{I} = T_o \dot{S}_{\text{Gen}}$  and represents the rate of exergy destruction. The rate of entropy generated ( $\dot{S}_{\text{Gen}}$ ) equals

$$\dot{S}_{\text{Gen}} = \dot{m}_3 s_3 - \dot{m}_1 s_1 - \dot{m}_2 s_2 \quad (23.105)$$

The greater the difference in temperature between the two inlet streams, the greater the destruction of exergy. The second law efficiency accounts for the exergy destroyed,

$$\eta_{II} = \frac{\dot{m}_3 \psi_3}{\dot{m}_1 \psi_1 + \dot{m}_2 \psi_2} \quad (23.106)$$

### 23.5.7 Ducts and Pipes

Exergy is lost in piping and duct work due to both head loss and heat transfer to the environment. The exergy loss due to heat transfer is simply the exergy of thermal energy at the temperature of the fluid. Head loss is a constant enthalpy process that converts flow work into internal energy. As a result, at elevated temperatures, some of the *lost* flow work is recovered as exergy in the form of useful thermal energy. On a per unit mass of flow basis, the combination of these two factors results in an exergy destruction in pipes and ducts,

$$i = q_{\text{LOST}} \left( 1 - \frac{T_o}{T} \right) + T_o (s_2 - s_1) \quad (23.107)$$

If heat loss can be neglected, the resulting irreversibility for ideal gases becomes

$$i = \psi_1 - \psi_2 = T_o R \ln \left( \frac{P_1}{P_2} \right) \quad (23.108)$$

Similarly for adiabatic pipes carrying liquids,

$$i = v(P_1 - P_2) \frac{T_o}{T} \quad (23.109)$$

To reduce this loss of exergy, systems should be designed to reduce head loss and heat loss to the environment. In the case of a closed reactor coolant loop, work supplied by the pump overcomes the flow resistance in the loop. The flow resistance converts the work supplied into thermal energy which has some exergy. As a result, the lost exergy from pump work in a closed loop system is difference between the work added and the exergy of heat at the same temperature,

$$i = w - w \left( 1 - \frac{T_o}{T} \right) = w \frac{T_o}{T} \quad (23.110)$$

The remaining energy supplied raises the exergy of the reactor coolant, which can be partially recovered.



## 23.6 Example Power Plant Applications

The following examples were selected to demonstrate the thermodynamic analysis principles presented in this chapter. For brevity, the later examples build on earlier examples and the level of detail in the calculations are reduced.

### 23.6.1 Example 1: PWR Power Plant Primary System

Consider the pressurized water reactor primary plant shown in Figure 23.11. The specifications in this example are roughly comparable in output to the Westinghouse AP1000, although for simplicity, the example uses a single loop with a single hot leg, and no attempt was made to exactly replicate AP1000 performance. Coolant exiting the steam generator is 2250 psia and 535°F and has a mass flow rate of  $110E6 \text{ lb}_m/\text{h}$ . The reactor coolant pump raises the pressure by 100 psi and has an isentropic efficiency of 85%. On the secondary side, the steam generator receives feed at 440°F and produces saturated steam at 830 psia. Assume a dead state of 75°F (535°R) and 14.7 psia.

Ideal pump work from Equation 23.31:

$$w_s = v\Delta P = \left( \frac{0.0209 \text{ ft}^3}{\text{lb}_m} \right) \left( \frac{100 \text{ lb}_f}{\text{in}^2} \right) \left( \frac{144 \text{ in}^2}{\text{ft}^2} \right) \left( \frac{\text{Btu}}{778 \text{ ft} \cdot \text{lb}_f} \right) = 0.385 \text{ Btu}/\text{lb}_m$$

Real pump work from Equation 23.84:

$$w_R = \frac{w_s}{\eta_I} = \frac{0.385 \text{ Btu}/\text{lb}_m}{0.85} = 0.451 \text{ Btu}/\text{lb}_m$$

Pump power from Equation 23.13:

$$\dot{W}_p = \dot{m}w_R = \left( \frac{110E6 \text{ lb}_b}{\text{h}} \right) \left( \frac{0.451 \text{ Btu}}{\text{lb}_m} \right) \left( \frac{\text{HP}}{2545 \text{ Btu}/\text{h}} \right) = 19.5E3 \text{ HP} = 14.5 \text{ MW}$$

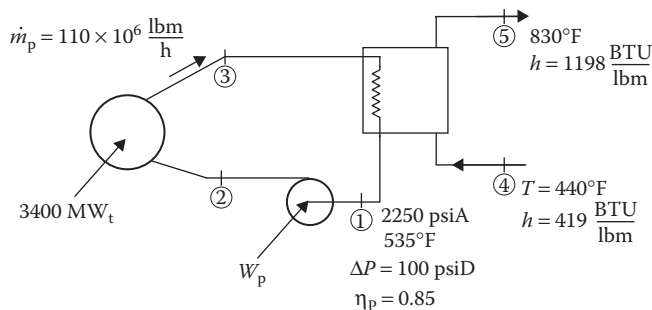


FIGURE 23.11

Example pressurized water reactor primary plant.

Reactor enthalpy change from Equations 23.14 and 23.69 where  $q_R$  is neglected:

$$\Delta h = \frac{\dot{Q}}{\dot{m}} = \left( \frac{3400 \text{ MW}}{110E6 \text{ lb}_m/\text{h}} \right) \left( \frac{3.413E6 \text{ Btu/h}}{\text{MW}} \right) = 105.5 \text{ Btu/lb}_m$$

This corresponds to an exit temperature of 622°F. The nuclear steam supply system (NSSS) thermal output is the heat transferred in the steam generator, which in this case equals the reactor thermal power and the energy added in the pump (roughly 3415 MW). The required steam flow rate on the secondary side is calculated by Equations 23.14 and 23.69 where  $q_R$  is neglected,

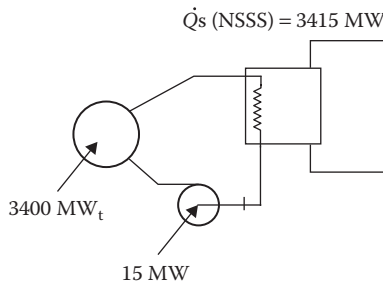
$$\dot{m} = \frac{\dot{Q}}{h_5 - h_4} = \left( \frac{3415 \text{ MW}}{(1198 - 419) \text{ Btu/lb}_m} \right) \left( \frac{3.413E6 \text{ Btu/h}}{\text{MW}} \right) = 14.96E6 \text{ lb}_m/\text{h}$$

The resultant energy balance is shown in Figure 23.12. Note that the steam generator has a first law efficiency of 100%.

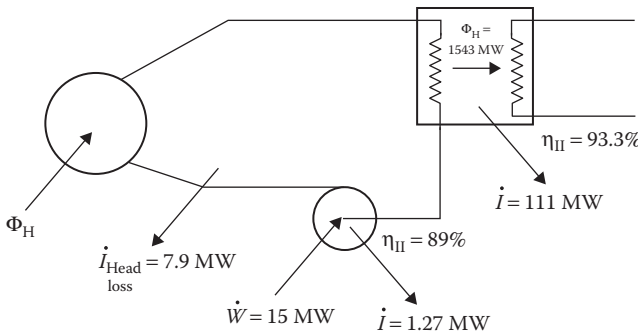
Second law analysis results are shown in Figure 23.13. The flow exergy ( $\psi$ ) of each point is calculated using Equation 23.52 where potential and kinetic energy are ignored.

The rate of exergy destruction in the steam generator from Equation 23.88 is,

$$\dot{I} = \dot{m}_{\text{Pri}}(\psi_3 - \psi_1) - \dot{m}_{\text{Sec}}(\psi_5 - \psi_4) = 111 \text{ MW}$$



**FIGURE 23.12**  
Energy balance for the PWR depicted in Figure 23.11.



**FIGURE 23.13**  
Exergy balance for the PWR depicted in Figure 23.11.

The resultant second law efficiency of the steam generator is diminished by the destruction of exergy resulting from heat transfer across a finite temperature difference from Equation 23.89.

$$\eta_{II} = \frac{\dot{m}_{\text{Sec}}(\psi_5 - \psi_4)}{\dot{m}_{\text{Pri}}(\psi_3 - \psi_1)} = 93.3\%$$

Most of the steam generator heat transfer occurs at saturation temperature for 830 psi steam, so reasonable heat transfer results can be achieved using a log mean temperature difference assuming that inlet and outlet temperatures are 523°F and a configuration factor of unity. As a result,  $\theta_{\text{LM}} = 38.4^\circ\text{F}$  and  $\dot{U}A = 3E8 \text{ Btu/h} \cdot ^\circ\text{F}$ .

Reactor coolant pump supplies 15 MW to overcome head loss in the primary loop. The resultant irreversibility from this head loss is calculated using Equation 23.110:

$$i = w \frac{T_o}{T} = 0.451 \text{ Btu/lb}_m \frac{535^\circ\text{R}}{995^\circ\text{R}} = 0.243 \text{ Btu/lb}_m$$

With the mass flow rate of 110 E6 lbm/h, the rate of exergy destruction resulting from pump work to overcome head loss in the primary plant is 7.8 MW. Note that this value includes the rate of exergy destruction in the pump itself (1.27 MW).

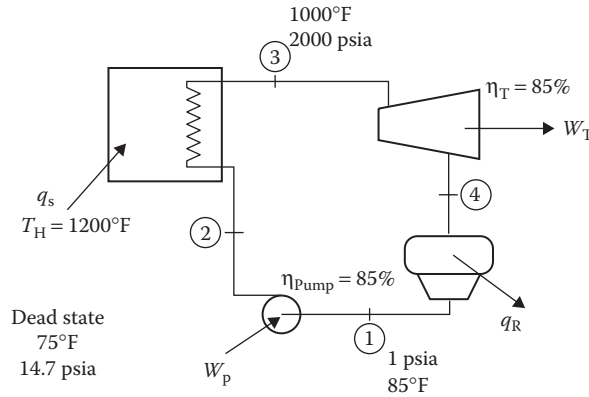
The first law efficiency of the reactor is 100%, if radiant energy leaving the vessel and heat loss are ignored. The second law efficiency depends on the assumption of the available temperature of the heat input. For example, assume that a reactor has a fission energy rate of  $\dot{Q}_F$ . The energy that remains in the core excludes energy carried off in neutrinos, which cannot be captured. It also excludes energy that leaves the core in the form of gamma and neutron energy; however, this quantity is dependent on the size and construction of the core, and larger cores will capture a greater fraction of this energy. The value of the then heat input to the core ( $\dot{Q} = \dot{Q}_F - \dot{Q}_{\text{Lost}}$ ) depends on the assumption of what can be reasonably captured. Furthermore, the exergy value of the heat depends on the temperature ( $T_H$ ) at which the heat is available and the dead state temperature ( $T_o$ ). Using Equation 23.74,

$$\Phi_H = \dot{Q} \left( 1 - \frac{T_o}{T_H} \right)$$

If the heat is considered available at the maximum peak centerline temperature allowed in a Westinghouse AP1000 (5160°R), then for a dead state of 535°R  $\Phi_H = 0.90 \dot{Q}$ . If on the other hand, the maximum temperature is considered to be the critical temperature of water (1166°R), then  $\Phi_H = 0.54 \dot{Q}$ .

### 23.6.2 Example 2: Rankine Cycle Power Plant

A simple Rankine steam plant is shown in Figure 23.14. Steam is supplied to a turbine at 1000°F and 2000 psia, where it is expanded in a turbine with an isentropic efficiency of 85% to a pressure of 1 psia. The exhausted steam is condensed and cooled to a subcooled liquid at 85°F. It is returned to the boiler with a pump that has an isentropic efficiency of 85%. The properties for each point in the cycle are calculated using computer generated steam tables and shown in Table 23.1. Points 2s and 4s represent conditions that would be achieved with an isentropic pump and turbine, respectively. In accordance with the state



**FIGURE 23.14**  
Example simple Rankine cycle.

**TABLE 23.1**  
Thermodynamic Properties for the Cycle Shown in Figure 23.14

Point	Temperature (°F)	Pressure (psia)	Quality	Specific Enthalpy (Btu/lbm)	Specific Entropy (Btu/lbm/R)	Exergy (Btu/lbm)	Phase
Dead state	<u>75</u>	<u>14.7</u>		43.13	0.084	0	Compressed liquid
1	<u>85</u>	<u>1</u>		53.07	0.1025	0.0425	Compressed liquid
2s	85.55	<u>2000</u>		59.01	<u>0.1025</u>		Compressed liquid
2	86.62	<u>2000</u>		<u>60.06</u>	0.1044	6.016	Compressed liquid
3	<u>1000</u>	<u>2000</u>		1474	1.56	641.21	Dense fluid ( $P > PC$ )
4s	101.7	<u>1</u>	0.7734	871	<u>1.56</u>		Liquid-vapor mixture
4	101.7	<u>1</u>	0.8613	<u>962</u>	1.722	42.54	Liquid-vapor mixture

postulate, two independent properties are required to determine the remaining properties at a point. The underlined cells indicate the two independent properties that were used to calculate the remaining cells at the state point. From the isentropic points, the actual enthalpies at points 2 and 4 were determined from the isentropic efficiencies of the devices.

Flow exergies of each real point are calculated using Equation 23.52.

In the boiler, no stack losses are present in the current example, so first law efficiency is 100% by definition. Using the data in Table 23.1 and Equations 23.69, 23.73, 23.74, and 23.80, the resultant calculations in the boiler are

$$q = h_3 - h_2 = 1474 - 60 = 1414 \text{ Btu/lb}_m$$

$$\phi_H = q \left( 1 - \frac{T_o}{T_H} \right) = 1414 \left( 1 - \frac{535^\circ\text{R}}{1660^\circ\text{R}} \right) = 958 \text{ Btu/lb}_m$$

$$i = \phi_H - (\psi_3 - \psi_2) = 958 - (641 - 6) = 323 \text{ Btu/lb}_m$$

$$\eta_{II} = \frac{\psi_3 - \psi_2}{\phi_H} = \frac{(641 - 6)}{958} = 66.3\%$$

The heat rejected in the condenser is transferred to the environment. Theoretically, it could be used as a process heat if there were a need for low temperature heat; however, any attempt to use this heat would likely result in a higher condenser pressure and lower plant efficiency. As a result, the irreversibility of the condenser is equal to the reduction of exergy in the condenser

$$i = \psi_4 - \psi_1 = 42.5 - 0.04 = 42.5 \text{ Btu/lb}_m$$

For the pump, work is supplied in the shaft, which raises the exergy and enthalpy of the feedwater. The energy and exergy calculations from Equations 23.81, 23.83, and 23.85 yield

$$w = h_2 - h_1; \quad i = w - (\psi_2 - \psi_1); \quad \eta_{II} = \frac{\psi_2 - \psi_1}{w} = 85.5\%$$

Similarly for the turbine using Equations 23.77, 23.79, and 23.80

$$w = h_3 - h_4; \quad i = (\psi_3 - \psi_4) - w; \quad \eta_{II} = \frac{w}{\psi_3 - \psi_4} = 85.5\%$$

Complete energy and exergy balances are shown in Figure 23.15.

The resultant plant thermal efficiency from Equation 23.63 is

$$\eta_{th} = \frac{w_{net}}{q_s} = \frac{w_{Turbine} - w_{Pump}}{q_s} = 35.7\%$$

Power and heat flow rate for each component and the plant are determined by multiplying the specific values by the mass flow rate. For example, a mass flow rate of 1 E6 lbm/h would produce a turbine power of

$$\dot{W} = \dot{m}w = (1E6 \text{ lbm/h})(512 \text{ Btu/lb}_m) = 512E6 \text{ Btu/lb}_m = 150 \text{ MW}$$

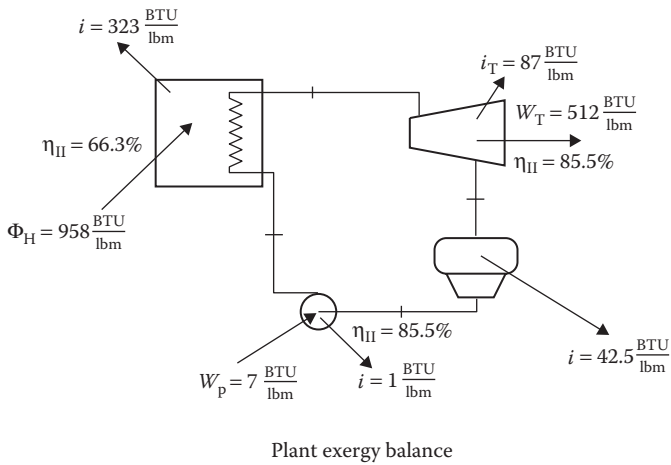
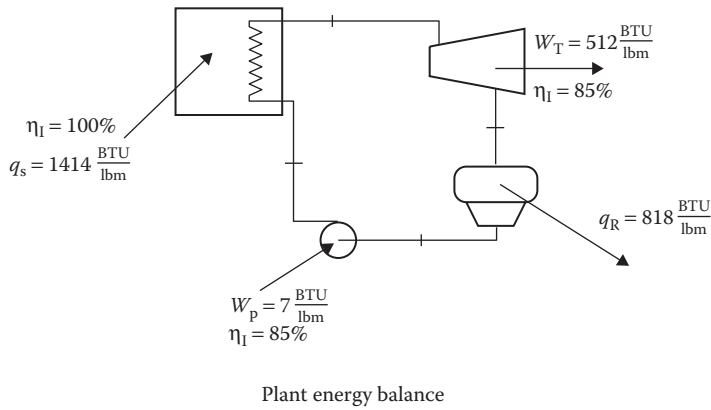
### 23.6.3 Example 3: Rankine Cycle with a Single Feedwater Heater

In this example, the effect of adding a single open feed heater (Figure 23.16) to the power plant in Example 2 (Section 23.6.2) is analyzed. The feed exiting the feed heater is assumed to be saturated liquid and all pump and turbine efficiencies are 85%.

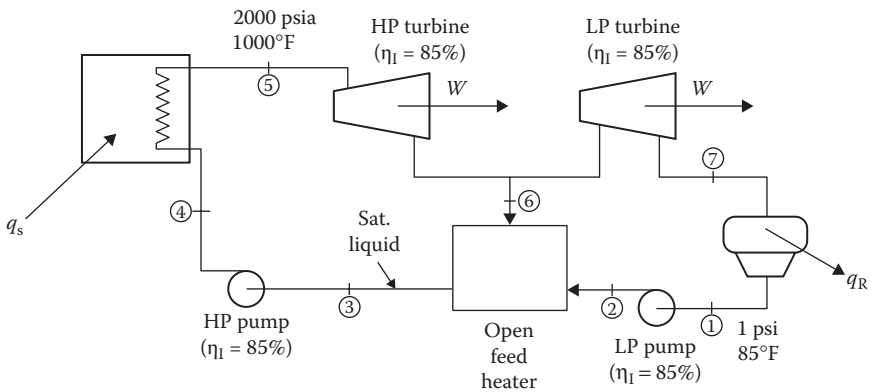
Determining the optimum extraction theoretically requires optimizing the efficiency of the plant as a function of extraction pressure; however, based on intuitive arguments, reasonable results can be obtained by selecting a saturation temperature that is the average of the condenser and boiler saturation temperatures (El Wakil, 1984).

$$P_{Ext} = P_{Sat} \left( \frac{T_{Sat \text{ boiler}} - T_{Sat \text{ condenser}}}{2} \right) \quad (23.111)$$

In this example, the boiler and condenser are at 2000 psia and 1 psia. These have saturation temperatures of 636°F and 102°F, respectively. The average of these saturation



**FIGURE 23.15**  
Energy and exergy balances for the cycle depicted in Figure 23.14.



**FIGURE 23.16**  
Rankine cycle with a single open feedwater heater.

**TABLE 23.2**

Thermodynamic Properties for the Cycle Shown in Figure 23.16

Point	Temperature (°F)	Pressure (psia)	Quality	Specific Enthalpy (Btu/lbm)	Specific Entropy (Btu/lbm/R)	Exergy (Btu/lbm)	Phase
Dead state	<u>75</u>	<u>14.7</u>		43.13	0.084	0	Compressed liquid
1	<u>85</u>	<u>1</u>		53.07	0.1025	0.0425	Compressed liquid
2s	85.05	<u>171</u>		53.58	<u>0.1025</u>		Compressed liquid
2	85.14	<u>171</u>		<u>53.67</u>	0.1027	0.5355	Compressed liquid
3	368.9	<u>171</u>	<u>0</u>	341.8	0.5276	61.344	Saturated liquid
4s	372.3	<u>2000</u>		348	<u>0.5276</u>		Compressed liquid
4	373.4	<u>2000</u>		<u>349.1</u>	0.5289	67.9485	Compressed liquid
5	<u>1000</u>	<u>2000</u>		1474	1.56	641.21	Dense fluid ( $P > PC$ )
6s	369.3	<u>171</u>		1197	<u>1.56</u>		Superheated vapor
6	439.9	<u>171</u>		<u>1239</u>	1.608	380.53	Superheated vapor
7s	101.7	<u>1</u>	0.7996	898.2	<u>1.608</u>		Liquid–vapor mixture
7	101.7	<u>1</u>	0.849	<u>949.3</u>	1.699	42.145	Liquid–vapor mixture

temperatures is 369°F, which has a saturation pressure of 171 psia. Properties are analyzed similar to the previous example and the results are summarized in Table 23.2. The underlined inputs are the independent properties used to determine other properties at the state point.

By convention when analyzing plants with steam extraction, energy balances are normalized by the flow exiting the steam generator. The fraction of steam extracted is defined as  $M = \dot{m}_6 / \dot{m}_5$ . Using the mass balance and energy balance from Equation 23.103

$$\dot{m}_2 + \dot{m}_6 = \dot{m}_3; \quad \dot{m}_2 h_2 + \dot{m}_6 h_6 = \dot{m}_3 h_3$$

yields an extraction fraction,

$$M = \frac{(h_3 - h_2)}{(h_6 - h_2)} = \frac{341.8 - 53.7}{1239 - 53.7} = 24.3\%$$

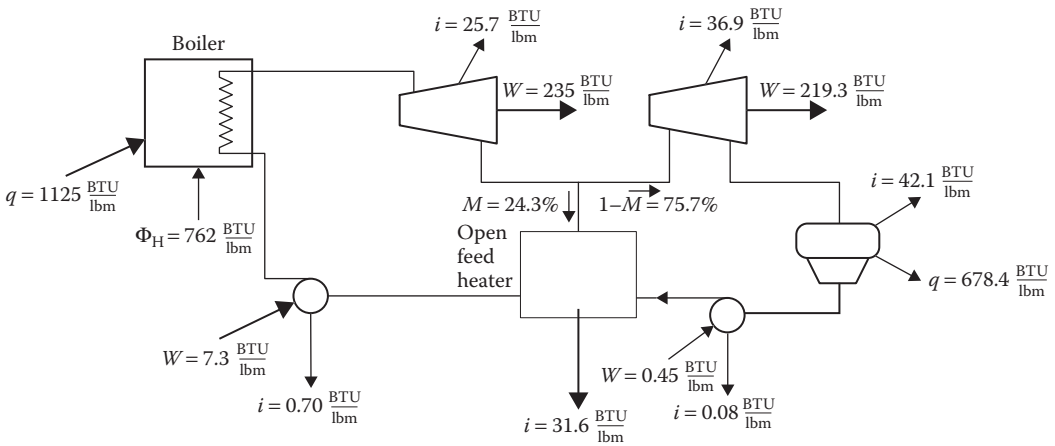
Heat and exergy balances are calculated in a similar manner to the previous example, except that all quantities are normalized on a per unit boiler flow basis. For example, the work and irreversibility of the low-pressure turbine are from Equations 23.77 and 23.79, modified for the flow rates are

$$w = (1 - M)(h_6 - h_7) = (1 - 0.243)(1239 - 949.3) = 219.3 \text{ Btu/lb}_m$$

$$i = (1 - M)[(\psi_6 - \psi_7) - w] = 38.9 \text{ Btu/lb}_m$$

From a first law basis, the open feed heater is 100% efficient. On a second law basis using Equation 23.104,

$$i = M\psi_6 + (1 - M)\psi_2 - \psi_3$$



**FIGURE 23.17**  
Energy and exergy balances for the cycle depicted in Figure 23.16.

The resultant energy and exergy balances for a plant with a single feed heater are shown in Figure 23.17.

In overall plant performance, the feed heater raised the thermal efficiency from 35.7% to 39.7% (roughly an 11% improvement in energy utilization).

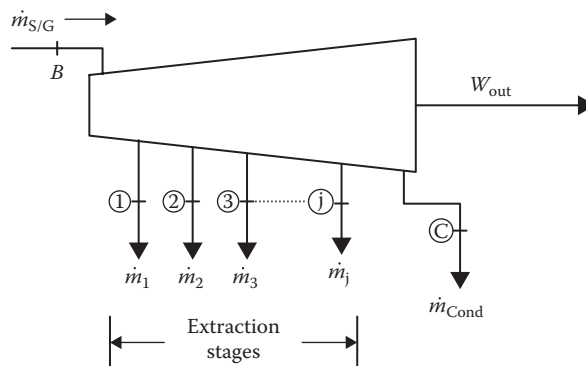
**23.6.4 Example 4: Rankine Cycle with Multiple Feed Heaters**

Generally, steam plants include multiple stages of extraction (typically around seven), which can be analyzed using the same principles. Consider the plant with  $j$  extraction points, each supplying one feed heater, shown in Figure 23.18.

The extraction fractions for the  $j$  extraction points are defined,

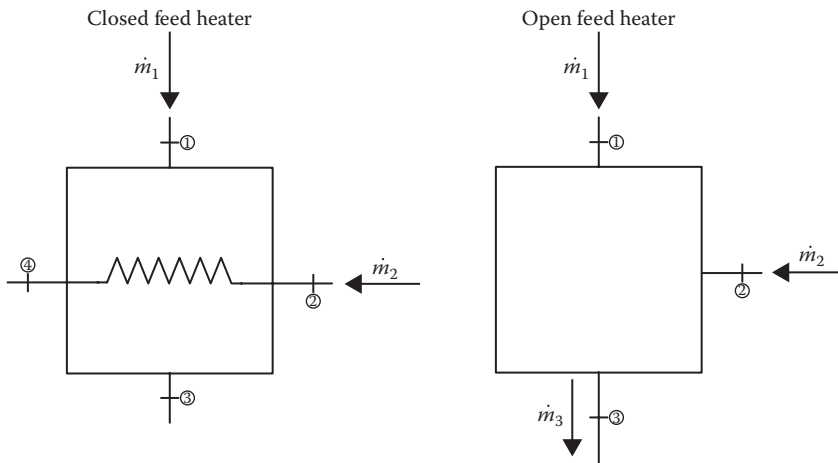
$$M_1 = \frac{\dot{m}_1}{\dot{m}_{S/G}}; \quad M_2 = \frac{\dot{m}_2}{\dot{m}_{S/G}} \quad \dots \quad M_j = \frac{\dot{m}_j}{\dot{m}_{S/G}}$$

Similar to Example 3 (Section 23.6.3), the values of  $M_i$  can be determined through mass and energy balances of the feedwater heaters.



**FIGURE 23.18**  
Steam turbine with multiple extraction points.





**FIGURE 23.19**  
Open and closed feedwater heaters.

For a closed feed heater (Figure 23.19), energy balances of Equation 23.86 yield

$$\frac{\dot{m}_1}{\dot{m}_2} = \frac{h_4 - h_2}{h_1 - h_3}$$

In the case of an open feed heater (Figure 23.19) from Equation 23.103 where the vent flow is neglected yields,

$$\frac{\dot{m}_1}{\dot{m}_3} = \frac{h_3 - h_2}{h_1 - h_2}$$

After each extraction stage, the mass flow rate of steam traveling through the turbine decreases, such that

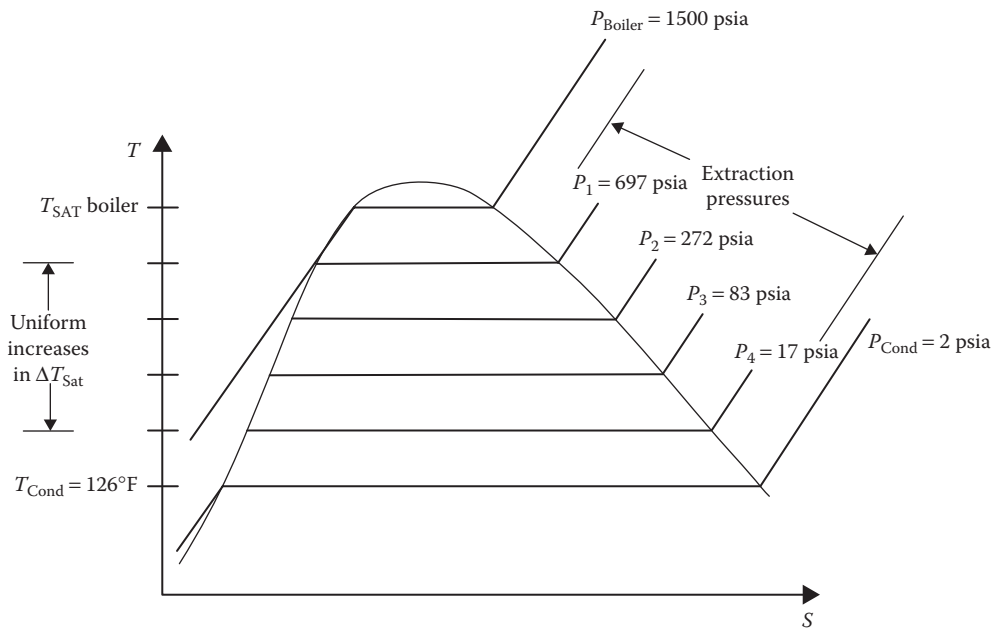
$$M_{i+1} = (1 - M_i) M_{i+1}^*$$

where  $M_{i+1}^*$  is defined as the fraction entering the  $i+1$  stage that is extracted. For the turbine in Figure 23.18, the net work on a boiler flow basis is,

$$w = (h_B - h_1) + (1 - M_1)(h_1 - h_2) + \dots + (1 - M_j)(h_j - h_C)$$

Optimizing the extraction pressure of multiple feed heaters, in principle, requires maximizing the plant efficiency while varying all of the extraction pressures. However, similar to the previous example, reasonable results can be obtained by intuitive arguments (El Wakil, 1984). Consider a plant with  $j$  stages of extraction. Based on intuitive arguments, feed temperature should be increased in equal increments from the condenser temperature until the boiling temperature at the boiler pressure  $T_{Boil}$ . The pressure  $i$ th extraction stage is calculated from

$$P_i = P_{Sat}(T_i)$$



**FIGURE 23.20**

Typical  $T$ - $s$  diagram for Rankine cycle with multiple steam extractions.

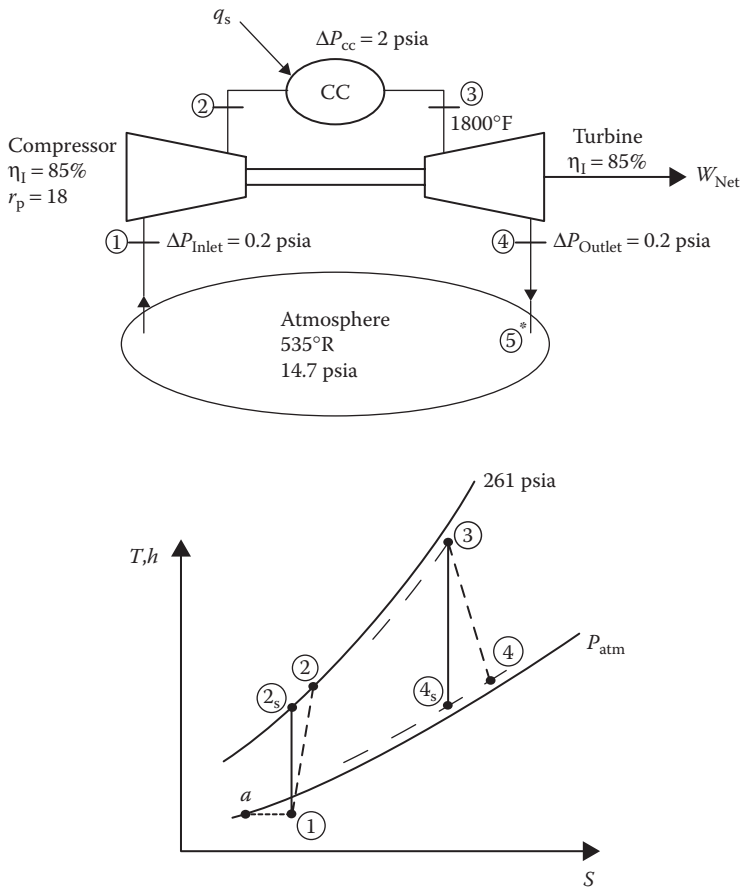
where temperatures are determined similar to Equation 23.111 in Example 3 (Section 23.6.3) using the following relationship:

$$T_i = T_{\text{Boil}} - i \left( \frac{T_{\text{Boil}} - T_{\text{Condenser}}}{j + 1} \right) \quad (23.112)$$

For super critical boilers,  $T_{\text{Boil}}$  is the critical temperature. As an example, consider a plant operating at a boiler pressure of 1500 psia and a condenser pressure of 2 psia with 4 extraction stages.  $T_{\text{Sat}}$  for the boiler and condenser are 597°F and 126°F, respectively. Saturation temperatures for extraction should then be, 502.8°F, 408.6°F, 314.4°F, and 220.2°F, which have saturation pressures of 697, 272, 83, and 17 psia. The  $T$ - $s$  diagram for this example is in Figure 23.20.

### 23.6.5 Example 5: Brayton Cycle Power Plants

Analysis procedures for Brayton cycles are similar to the previous examples of Rankine cycles with the exception that properties are determined from gas tables. The gas turbine in Figure 23.21 is analyzed using air *standard assumptions*, which consider the working fluid to be air and treat combustion as a heat addition process. A more accurate analysis could employ tables that account for combustion products, such as Keenan and Kay (1960). The heat to the combustion chamber is assumed to be available at 2500°F. In the example, air from the dead state (14.7 psia, 75°F) enters the intake structure. Pressure drops before the turbine inlet in the intake ducting to 14.5 psia. It is compressed in a compressor with a



**FIGURE 23.21**  
Example open Brayton cycle.

pressure ratio of 18:1 and an isentropic efficiency of 85%. In the combustion chamber, pressure drops 2 psia and the air is heated to 1800°F. It then passes through a turbine with an isentropic efficiency of 85%. The exhaust duct has a pressure drop of 0.2 psia.

Table 23.3 summarizes the conditions at each state point. (Points 2s and 4s represent points the conditions for isentropic processes; input properties in the table are underlined.) Point *a* is at atmospheric conditions (the dead state). Point 1, the compressor inlet, is reached after the atmospheric air undergoes a throttling process (constant enthalpy) and experiences a 0.2 psi pressure drop, and point 2s would be achieved in an isentropic compressor. The ratio of the reduced pressure equals the ratio of the actual pressures for an isentropic process. In point 2, the enthalpy is calculated from the compressor isentropic efficiency. In the combustion chamber, the pressure drops 2 psi and the temperature is increased to 1800°F. Point 4s and 4 are analyzed similarly to 2s and 2. Finally, the pressure drops to point 5 where the air enters the atmosphere. The accompanying *h-s* (or *T-s*) diagram shows the cycle and includes all of the pressure drops and turbine and compressor inefficiencies.

**TABLE 23.3**

Thermodynamic Properties for the Cycle Shown in Figure 23.21

Point	Temperature (°F)	Pressure (psia)	Specific Entropy (Mass) Btu/lbm/R	Specific Enthalpy (Mass) Btu/lbm	Reduced Pressure	Exergy (Btu/lbm)
a	75	14.7	1.637	127.9	1.077	0
1	75	14.5	1.638	127.9	1.077	-0.535
2s	741.7	261	1.638	291.8	19.39	
2	854.4	261	1.661	320.7	27.12	179.96
3	1800	259	1.808	577.4	230	358.015
4s	623.4	14.9	1.808	261.9	13.23	
4	809.8	14.9	1.849	309.2	23.82	67.88
5	809.8	14.7	1.85	309.2	23.82	67.345

Resultant exergy and energy balances are shown in Figure 23.22, where the irreversibilities are calculated using Equations 23.83, 23.73, 23.79, and 23.108, respectively,

$$i_{\Delta P \text{ Inlet}} = \psi_a - \psi_1$$

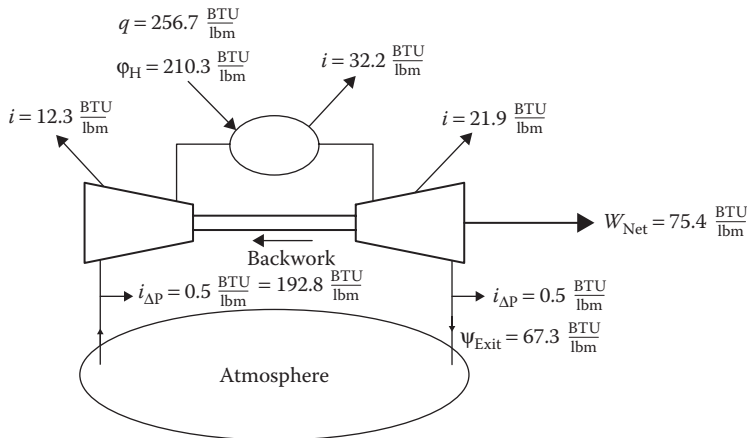
$$i_{\text{Comp}} = (h_2 - h_1) - (\psi_2 - \psi_1)$$

$$i_{\text{Comb Chamber}} = \phi_H - (\psi_3 - \psi_2) = (h_3 - h_2) \left( 1 - \frac{T_o}{T_H} \right) - (\psi_3 - \psi_2)$$

$$i_{\text{Turbine}} = (\psi_3 - \psi_4) - (h_3 - h_4)$$

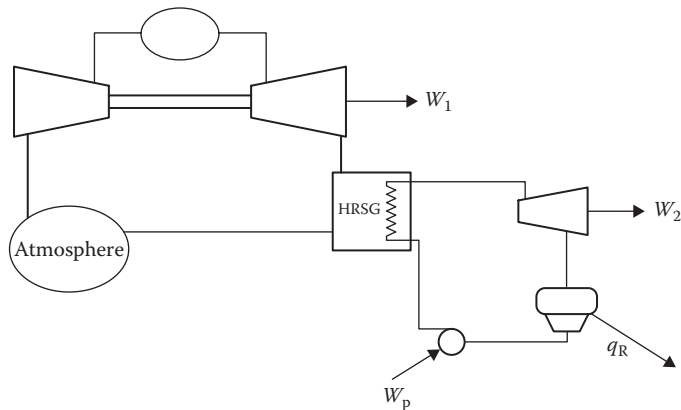
$$i_{\Delta P \text{ Outlet}} = \psi_4 - \psi_5$$

As Figure 23.23 indicates, the greatest loss of exergy in the gas turbine is in the high exergy content of the exhaust gas. Only marginal improvements can result from improved



**FIGURE 23.22**

Energy and exergy balances for the open Brayton cycle depicted in Figure 23.21.



**FIGURE 23.23**  
Combined cycle power plant.

turbine and compressor efficiencies. Consequently, the exergy analysis points to investigating methods to use the heat content of the exhaust gas.

### 23.6.6 Example 6: Combined Cycle Plant

The high temperature of the exhaust gases in Brayton cycle plants and the low cost of natural gas has led to popularity of combined cycle plants like the one shown in Figure 23.23. An HRSG extracts heat from the exhaust and uses it to generate steam, which is in turn used in a Rankine cycle *bottoming cycle*.

The principles presented here are easily adapted to a closed cycle high-temperature gas-cooled reactor cycle as shown in Figure 23.24. Individually, the bottoming and topping cycle are analyzed similarly to the Brayton and Rankine cycles in the previous examples. The HRSG energy and exergy balances are similarly analyzed to the steam generator in Example 1 (Section 23.6.1). Using

$$M = \frac{\dot{m}_{\text{Rankine}}}{\dot{m}_{\text{Brayton}}} = \frac{(h_4 - h_1)}{(h_7 - h_6)}$$

energy and exergy balances can all be normalized to a per reactor mass flow basis. For example, the net work and irreversibility in the HRSG can be calculated,

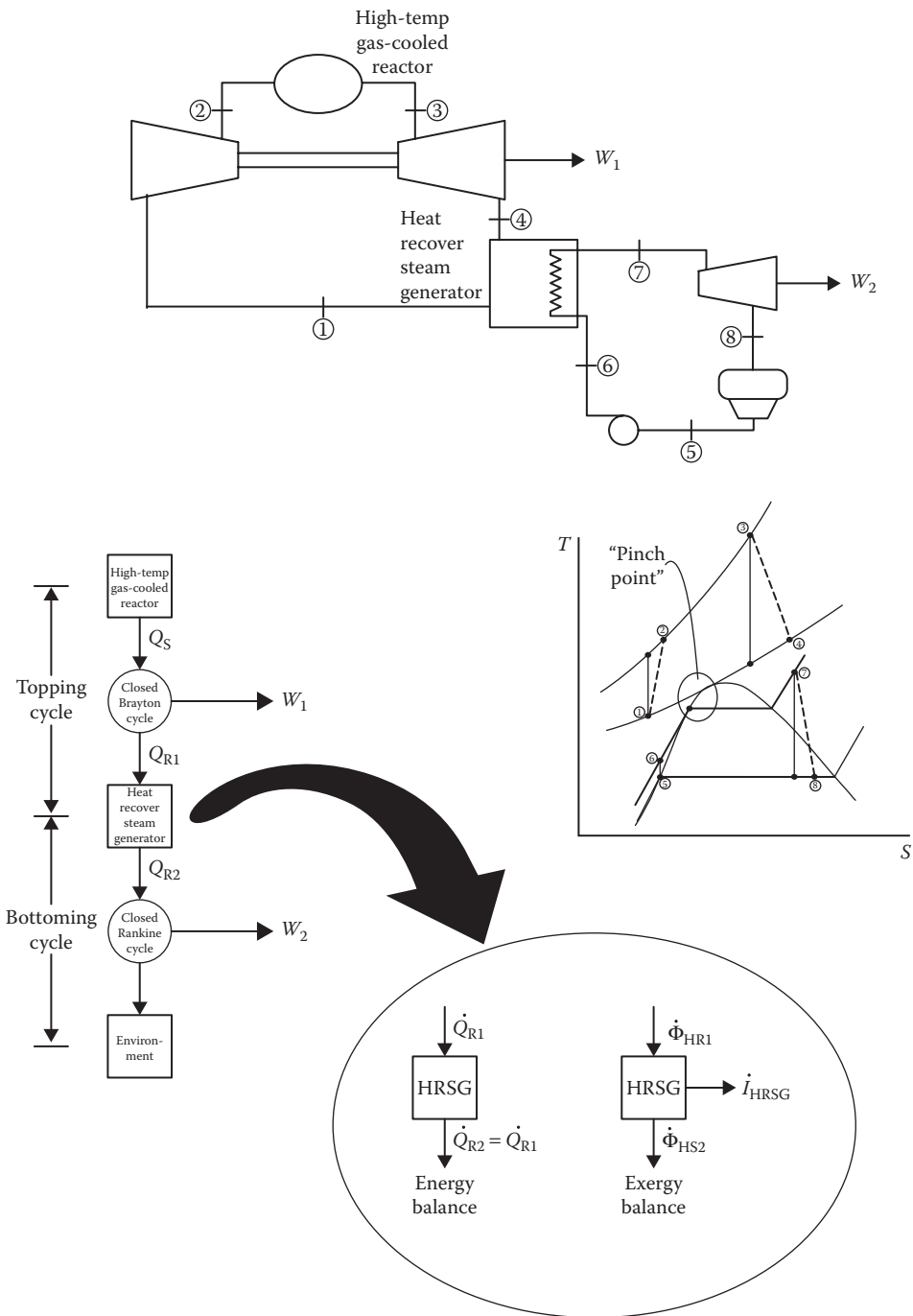
$$w_{\text{Net}} = (h_3 - h_4) - (h_2 - h_1) + M [(h_7 - h_8) - (h_6 - h_5)]$$

$$i_{\text{HRSG}} = (\psi_4 - \psi_1) - M (\psi_7 - \psi_6)$$

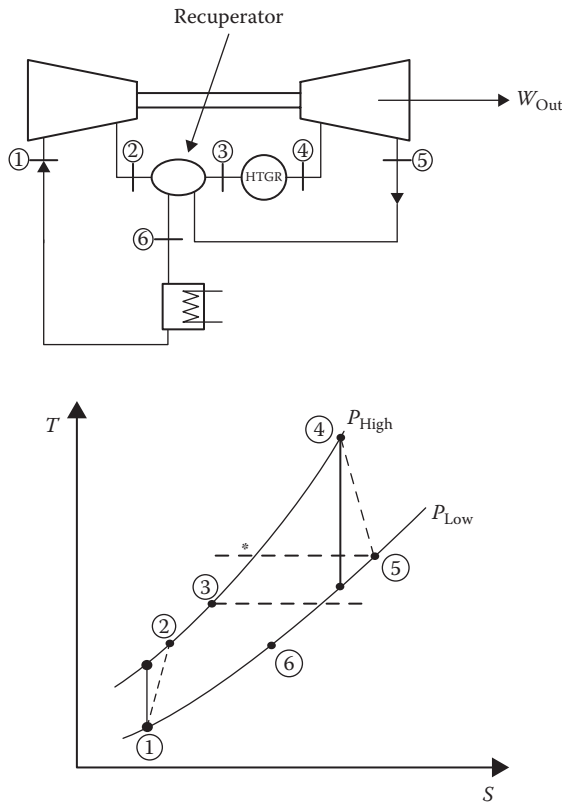
The *pinch point* shown in the  $T$ - $s$  diagram is a critical consideration for the design of HRSGs since the exhaust gas must remain hotter than the steam at all locations in the boiler.

### 23.6.7 Example 7: Recuperated High-Temperature Gas-Cooled Reactor

Gas turbine efficiency can be enhanced by the use of recuperator (see Figure 23.25), which pre-heats the gas prior to the heat addition process.



**FIGURE 23.24**  
High-temperature gas-cooled reactor operating on a combined cycle.



**FIGURE 23.25**  
Recuperated Brayton cycle.

The heat that must be added in the reactor is reduced by the heat extracted from point 5 to 6. Because the flow rate through both streams is identical,

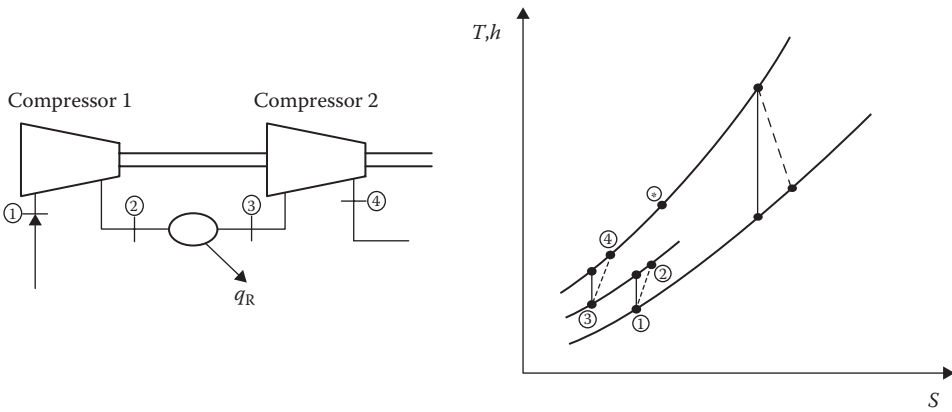
$$h_5 - h_6 = h_3 - h_2 \quad \text{and} \quad i = (\psi_5 - \psi_6) - (\psi_3 - \psi_2)$$

The recuperator effectiveness ( $\epsilon$ ) is the fraction of heat transferred that would be transferred in a counterflow heat exchanger of infinite heat transfer area. In this case

$$\epsilon = \frac{h_3 - h_2}{h_5 - h_2} = \frac{h_5 - h_6}{h_5 - h_2}$$

The \* in the process diagram represents the maximum temperature that would be achieved for  $\epsilon = 1$ . Depending on the outlet temperature at point 6, it would be possible to recover additional exergy in an HRSG run with a bottoming cycle.

The performance of a recuperated cycle can be further enhanced by using intercoolers in a compressor as shown in Figure 23.26. The \* indicates the state that would be achieved without intercooling. Since specific volume of the gas decreases as temperature decreases, the total work required is reduced as the result of an intercooler, and since the heat rejected is returned in the recuperator, the efficiency is further enhanced. However, intercooling by itself has little effect on plant efficiency since it results in greater heat requirements from the heat source.



**FIGURE 23.26**  
Intercooled cycle compression.

## Appendix A—Nomenclature

$A$	heat transfer area
$c$	specific heat for a solid or incompressible fluid
$c_v, c_p$	specific heat at constant volume and specific heat at constant pressure, respectively
$d$	differential of a property (path-independent quantity)
$e$	energy per unit mass
$h_c$	enthalpy per unit mass
$h$	convective heat transfer film coefficient
$i, I$	irreversibility per unit mass and total irreversibility, respectively
$\dot{I}$	rate of irreversibility (rate of exergy destruction)
$k$	ratio of specific heats ( $c_p/c_v$ )
$ke$	kinetic energy per unit mass
$F$	heat exchanger configuration factor
$HR$	heat rate (dimensional reciprocal or efficiency in Btu/kWh)
$NTU$	number of transfer units in a heat exchanger
$m, \dot{m}$	system mass and mass flow rate, respectively
$MW$	molecular weight
$P$	pressure (subscripts: $P_{Gage}$ , $P_{Abs}$ , $P_{Vac}$ )
$P_o$	atmospheric pressure
$pe$	potential energy per unit mass
$q, Q, \dot{Q}$	heat transfer per unit mass, total heat transfer, and rate of heat transfer, respectively
$R$	gas constant for a specific gas; thermal resistance to heat transfer
$\bar{R}$	universal gas constant
$s, \dot{S}_{Gen}$	entropy per unit mass and rate of entropy generation, respectively
$T$	temperature
$T_o$	environmental temperature (dead state temperature)
$u$	internal energy per unit mass



$\hat{U}$	overall heat transfer coefficient
$\hat{U}A$	receptacle of $R$ frequently used, although internal and external areas differ
$V$	velocity
$v, \mathcal{V}, \dot{\mathcal{V}}$	volume per unit mass, volume, and volumetric flow rate, respectively
$w, W, \dot{W}$	work per unit mass, work, and power, respectively
$\delta$	differential of a property (path independent quantity)
$\varepsilon$	heat exchanger effectiveness
$\phi, \Phi, \dot{\Phi}$	exergy per unit mass, exergy, and exergy transfer rate, respectively
$\phi_{\text{Chm}}$	chemical exergy of a combustible fuel
$\eta, \eta_I, \eta_{II}$	efficiency, first law efficiency, and second law efficiency, respectively
$\eta_{\text{Carnot}}$	Carnot efficiency (maximum theoretical efficiency in any heat engine)
$\rho$	density
$\psi$	flow exergy per unit mass
$\theta$	temperature difference at a given location in a heat exchanger
$\theta_{\text{LM}}$	log mean temperature difference

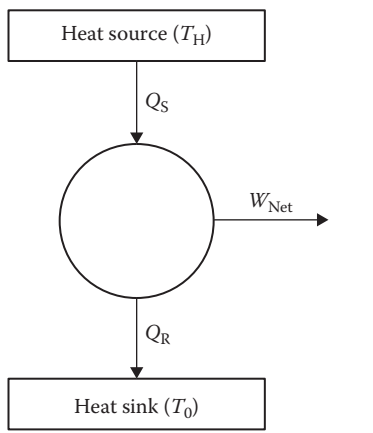
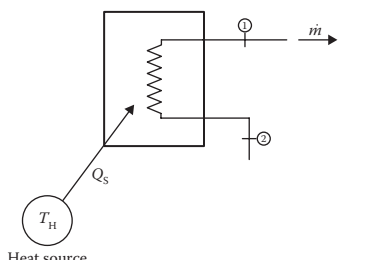
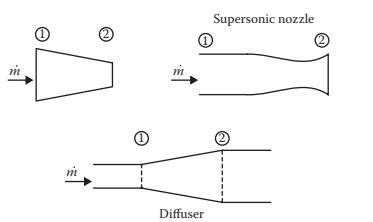
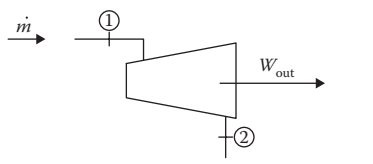
**Subscripts**

$C$	cold side fluid
$f$	saturated liquid
$g$	saturated vapor
$fg$	difference in property for vaporization from liquid to vapor
$H$	heat source or alternatively hot side fluid
$o$	dead state (nominally 14.7 psia, 537°R)
$r$	rejected heat
$s$	isentropic process or point or also supplied heat

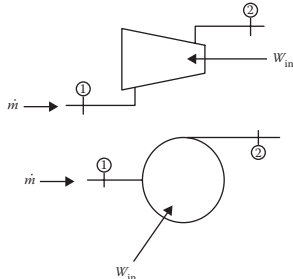
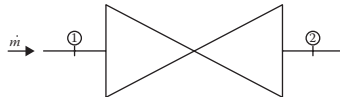
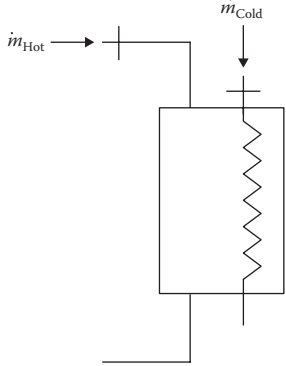
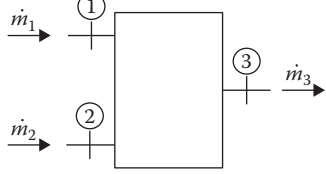
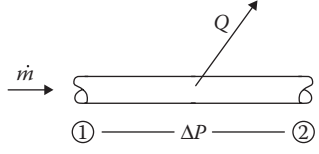
**Over dot**

$\dot{\square}$	rate or time derivative
-----------------	-------------------------

Appendix B—Analysis Summary

Device	Assumptions	Equations
<p>Heat engines</p> 		$\eta_I = \frac{W_{NET}}{Q_S}; \eta_{II} = \frac{W_{NET}}{\Phi_S} = \frac{\eta_I}{\eta_{CARNOT}}$
<p>Reactors, combustion chambers and boilers</p> 		<p>Energy: <math>q_s - q_R = h_2 - h_1</math>                      Exergy: <math>\phi_s - \phi_R - i = \psi_2 - \psi_1</math>  <math display="block">\phi = q \left( 1 - \frac{T_o}{T_{LIMIT}} \right)</math>  <math display="block">\eta_I = \frac{h_2 - h_1}{q_s}; \eta_{II} = \frac{\psi_2 - \psi_1}{\phi_s}</math></p>
<p>Nozzles and diffusers</p> 	<p>Adiabatic                      Insignificant potential energy</p>	<p>Mass: <math>\dot{m}_1 = \dot{m}_2</math>                      Energy: <math>\Delta ke = -\Delta h</math>                      Entropy: <math>s_{GEN} = s_2 - s_1</math>                      Exergy: <math>ke_2 - ke_1 = \psi_1 - \psi_2 - T_o (s_2 - s_1)</math>  <math display="block">\eta_I = \frac{h_1 - h_2}{h_1 - h_{2s}}; \eta_{II} = \frac{\psi_2}{\psi_1}</math></p>
<p>Turbines</p> 	<p>Adiabatic                      Insignificant potential and kinetic energy</p>	<p>Mass: <math>\dot{m}_1 = \dot{m}_2</math>                      Energy: <math>w = h_1 - h_2</math>                      Entropy: <math>s_{GEN} = s_2 - s_1</math>                      Exergy: <math>w = \psi_1 - \psi_2 - T_o (s_2 - s_1)</math>  <math display="block">\eta_I = \frac{h_1 - h_2}{h_1 - h_{2s}}; \eta_{II} = \frac{h_1 - h_2}{\psi_1 - \psi_2} = 1 - \frac{T_o (s_2 - s_1)}{\psi_1 - \psi_2}</math></p>

(Continued)

Device	Assumptions	Equations
<p>Compressors and pumps</p> 	<p>Adiabatic Insignificant potential and kinetic energy</p>	<p>Mass: <math>\dot{m}_1 = \dot{m}_2</math>                      Energy: <math>w_{IN} = h_2 - h_1</math>                      Ideal pump work for incompressible fluid:  <math>w_{IDEAL} = v(P_2 - P_1)</math>                      Entropy: <math>s_{GEN} = s_2 - s_1</math>                      Exergy: <math>w_{IN} = \psi_2 - \psi_1 + T_o(s_2 - s_1)</math>  <math>\eta_I = \frac{h_2 - h_{1s}}{h_2 - h_1}</math>; <math>\eta_{II} = \frac{h_2 - h_1}{\psi_2 - \psi_1} = 1 - \frac{T_o(s_2 - s_1)}{\psi_2 - \psi_1}</math></p>
<p>Throttle valves</p> 	<p>Adiabatic Insignificant potential and kinetic energy</p>	<p>Mass: <math>\dot{m}_1 = \dot{m}_2</math>                      Energy: <math>h_1 = h_2</math>                      Entropy: <math>s_{GEN} = s_2 - s_1</math>                      Exergy: <math>\psi_1 - \psi_2 = T_o s_{GEN}</math></p>
<p>Noncontact heat exchangers</p> 	<p>Adiabatic, except for heat exchange between fluids Insignificant potential and kinetic energy</p>	<p>Mass: for each stream: <math>\dot{m}_{IN} = \dot{m}_{OUT}</math>                      Energy: <math>\dot{m}(h_{IN} - h_{OUT}) _{HOT\ STREAM} = \dot{m}(h_{OUT} - h_{IN}) _{COLD\ STREAM}</math>                      Entropy: <math>\dot{S}_{Gen} = \dot{m}_C(s_{C2} - s_{C1}) + \dot{m}_H(s_{H2} - s_{H1})</math>                      Exergy: <math>\dot{m}_H(\psi_{H1} - \psi_{H2}) = \dot{m}_C(\psi_{C2} - \psi_{C1}) + T_o \dot{S}_{Gen}</math>  <math>\eta_I = 100\%</math>; <math>\eta_{II} = \frac{\dot{m}_C(\psi_{C2} - \psi_{C1})}{\dot{m}_H(\psi_{H1} - \psi_{H2})}</math></p>
<p>Mixing chamber</p> 	<p>Adiabatic Insignificant potential and kinetic energy</p>	<p>Mass: <math>\dot{m}_1 + \dot{m}_2 = \dot{m}_3</math>                      Energy: <math>\dot{m}_1 h_1 + \dot{m}_2 h_2 = \dot{m}_3 h_3</math>                      Entropy: <math>\dot{S}_{Gen} = \dot{m}_3 s_3 - \dot{m}_1 s_1 - \dot{m}_2 s_2</math>                      Exergy: <math>\dot{m}_1 \psi_1 + \dot{m}_2 \psi_2 = \dot{m}_3 \psi_3 + T_o \dot{S}_{Gen}</math>  <math>\eta_I = 100\%</math>; <math>\eta_{II} = \frac{\dot{m}_3 \psi_3}{\dot{m}_1 \psi_1 + \dot{m}_2 \psi_2}</math></p>
<p>Duct or piping with heat transfer</p> 	<p>Adiabatic, ideal gases: <math>i = T_o R \ln\left(\frac{P_1}{P_2}\right)</math>                      Adiabatic, liquids: <math>i = v(P_1 - P_2) \frac{T_o}{T}</math></p>	<p>Mass: <math>\dot{m}_1 = \dot{m}_2</math>                      Energy: <math>h_1 - h_2 = q</math>                      Entropy: <math>s_{GEN} = s_2 - s_1 + q/T</math>                      Exergy: <math>\dot{I} = \dot{m}(\psi_1 - \psi_2) = \dot{m} T_o s_{GEN} + \dot{Q}_{LOST} \left(1 - \frac{T_o}{T}\right)</math>  <math>i = q_{LOST} \left(1 - \frac{T_o}{T}\right) + T_o (s_2 - s_1)</math></p>
<p>Closed pumped primary loop</p>		<p><math>i = w T_o / T</math></p>

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# 24

## *Economics of Nuclear Power*

Jay F. Kunze and Edward S. Lum

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## 24.1 History and the Promise of Inexpensive Energy

The economics of nuclear power for generating electricity is determined by factors that are nontraditional with respect to comparisons with electricity derived from fossil energy. These economic differences are the result of a huge difference in energy release per atom, nuclear energy versus chemical. The former involves changes in the nucleus while the latter involves changes in the orbital electrons surrounding the nucleus. A brief discussion of these differences follows.

In 1938, Hahn, Strassman, and Meitner proved that they had fissioned the uranium atom using neutrons from an artificial source (a mixture of radium and beryllium). Scientists throughout the world used the values that were then known for the masses of the fission products versus that of the starting atom of uranium and concluded that a tremendous amount of energy is released in the fission process. The fissioning experiment was reproduced in about 100 different universities in the United States within the next year. The conversion of the mass lost in the fission process indicated that the fission of a uranium atom would release approximately 200 MeV (million electron volts). This is in sharp contrast to most chemical processes, such as combustion, which release only about 3–5 eV (electron volts) per combustion of a carbon or hydrogen atom. The ratio on an atom-to-atom basis is the order of 50,000,000 to 1.

Obviously, the immediate question is why this tremendous difference in energy release per atom. The reason is simply the difference between the binding energy of the electrons "orbiting" around the nucleus of the atom and the binding energy of the nucleons in the very tiny nucleus. The electrons in the atom are bound by energies in the range of a few electron volts for the outer shell of electrons to thousands of electron volts in the inner shell of the heavy nuclei. It is these latter energies that, when released, comprise what are

normally termed x-rays. The nucleons (protons and neutrons) in the nucleus are bound by various forces, mostly a somewhat mysterious nuclear binding force. The overall effect is a binding energy of the order of 8 MeV for each nucleon. Despite this huge ratio in the difference between nuclear and chemical energy, the question is how much uranium exists in the earth's crust compared to carbon (coal) or hydrocarbons (oil and gas). The ratio is of the 330 ppm by weight of carbon to ~12 ppm by weight for uranium plus thorium (the latter can be converted to the fissile uranium 233 isotope). This then represents a 28:1 ratio in favor of fossil fuels. Furthermore, on an atom basis *by weight*, uranium's mass is 20 times that of carbon. Hence, the final comparison of total energy available in the earth's crust is approximately

$$50,000,000[1/28][1/20] = 89,000/1 = \text{nuclear fission energy/fossil energy}$$

Obviously scientists were excited about this apparently inexhaustible supply of nuclear fission energy, which would last perhaps five orders of magnitude longer time than the fossil energy (coal, oil, and gas), on which the world at that time was fully dependent for its energy sources.

But it was the ratio of 50,000,000 to 1 of the energy per atom of fission compared to chemical that excited those who build the power plants. It indicated that even if uranium was 100 or 1000 times more expensive than coal, and if the power plants converting nuclear energy to electric energy were 10–100 times more costly, the cost of electricity generated by nuclear power would be very low, perhaps trivial, as the often quoted “too cheap to meter” expression implies.

Unfortunately, the immediate attention of this outstanding energy source was for a weapon of war, a nuclear bomb. The two “atomic bombs” dropped on Japan brought an end to World War II where the first the public knew about nuclear energy. Following the war, the attention of most engineers and scientists regarding nuclear power was directed to designing and building electricity generating power plants to provide relatively inexpensive electricity.

What has happened since leaves many in doubt about this unusually abundant and powerful form of energy. Initially, several demonstration power plants were built in the late 1950s, funded by the Atomic Energy Commission. The success of those initial nuclear power plants resulted in a fast growing interest in nuclear-electric power plants. The first power plant that was not funded by the government, the Oyster Creek Boiling Water reactor plant, was contracted for in 1965 and was placed in operation 4.5 years later. The contract price for the plant was less than a competitive bid for a coal-fired plant. This success resulted in a fast growing number of orders for nuclear power plants. By 1975, there were approximately 200 large plants (500–1200 MW electric output each) contracted for construction in the United States. Nuclear power plants did not achieve “too cheap to meter” status for the electricity they produce, and the hazardous spent nuclear fuel by-product (containing fission products) created fear and represented political issues that have since stalled nuclear development in the United States. The Three Mile Island incident in the United States in 1979 and the Chernobyl accident in 1986 near Kiev, Ukraine, in the Soviet Union largely convinced the public in the United States that nuclear power had lost its appeal and was no longer an acceptable source for generating electricity. However, a number of other countries had moved ahead with construction of plants, notably France, South Korea, India, and Japan. More recently, after the turn of the century, China launched a major nuclear construction program. However, following the terrible tsunami at Fukushima, Japan, the public fear, created by the media attention to the four damaged

nuclear plants, has had a major effect on the nuclear program of many countries throughout the world. Germany, despite its success with nuclear power, has decided to shut down all of its nuclear plants when they reach the end of their current licenses. Italy has decided to build no more plants, and France has planned to reduce its nuclear electric generation from the current 80% of its electricity to 50% or less.

At the time of this writing (2014), in the United States, the concerns of global climate change caused, at least in part, by the huge and growing amounts of carbon dioxide being emitted into the atmosphere have caused government officials, business leaders, and even the public to reconsider the advantages of nuclear power. Today, the cost of power generated at the existing nuclear power plants is very competitive with the other forms of generating electricity, averaging slightly lower in cost than coal, wind, solar, or natural gas. In Table 24.1, comparisons of the operating cost to produce a kilowatt-hour (kWh) of energy using these various types of energies are made. Operating cost only reflects the cost operating an existing power plant and includes the fuel cost (which is essentially the cost of the “fuel” delivered to the plant) plus the fixed costs of maintaining the plant in working order. In most cases, this latter cost is primarily that of employee wages and benefits, but also includes fixed costs of fees, taxes, and materials. Table 24.1 lists these nominal costs based on the most modern of these power plants used for base-load supply of electricity and assumes that these have capacity factors of 90%, that is, each plant operates at the equivalent of full power for 90% of the hours in each year, or equivalent combination of power times hours throughout the year.

Two figures of note in Table 24.1 are the very high fuel costs of plants operating using petroleum and the relatively high fixed costs for the nuclear plants. The former makes petroleum plants highly uneconomical, and as a result only 0.4% of the electricity generated in 2013 was from such plants. The competitiveness of nuclear plants depends heavily on the employee costs, which account for 80%–90% of the total fixed cost.

The omission of the “renewable” energies—hydroelectric, wind, solar, and geothermal energies—from Table 24.1 is noteworthy because the “fuel” costs of these plants is essentially zero. The construction (capital costs) divided by the capacity factors (giving an effective construction cost per unit output) of these plants is quite high compared to the conventional fossil fuel and nuclear plant construction costs. Hence, comparisons of actual total costs of the various forms of generating electricity must include capital costs as well as operating costs and the effective capacity factors.

The following sections examine the current costs of nuclear power and the projections into the future as new power plants are being built. The examination will initially focus on the cost of light water reactor plants, which represent 90% of the nuclear power plants

**TABLE 24.1**

Nominal Average Production Costs in Cents/kWh for Plants Operating in 2013 in the United States. Does Not Include Capital Recovery Costs on Investment

Type of Plant	Fixed Cost (cents/kWh)	Fuel Cost (cents/kWh)	Total Cost (cents/kWh)	Basis – Plant Characteristics
Coal	0.6	1.6	2.2	\$40/ton, 11,000 Btu/pound, 40% thermal efficiency
Natural gas	0.3	2.9	3.2	\$5 per million Btu, 60% thermal efficiency
Petroleum	0.3	15.25	15.6	\$100 per barrel, 40% thermal efficiency
Nuclear	1.4	0.7 + tax 0.1	2.2	Current fleet of light water reactors



operating in the world in 2014. The costs related to the heavy water moderated power plants from Canada are quite similar to those of the light water reactors. Newer power plant designs such as small modular reactors (SMRs), Generation IV reactors (of which there are six basic designs) will be discussed in general terms, because there are not even “first of a kind” (FOAK) costs available for these new designs.

In the 75 years since the discovery of nuclear fission, other applications of nuclear power have been considered—for naval propulsion, aircraft propulsion, and for land transportation (automobiles, trucks, and trains). Nuclear power directly applied to land transportation was quickly considered impractical because of the need to shield a nuclear power plant to protect those in or near the vehicle from harmful levels of radiation. The mass of the shielding made the use of such power plants impractical.

Nuclear power for aircraft was a focus of a US Air Force project from 1954 to 1961. The testing of high-temperature air-cooled nuclear reactors was carried out for many of those years at the Idaho Test Station, part of the National Reactor Testing Station in Idaho. The nuclear reactor replaced the combustion chamber in a standard jet engine. This program achieved developing high temperature, air cooled reactors that could supply heated air to the turbines that was hotter than jet engine turbines could tolerate in that era. Designs of installing one of these reactors with the jet engine compressor and turbine in existing large military aircraft (the B-36 bomber) along with necessary shielding to protect the crew were carried out. Also, a small water moderated research reactor was flown and operated while the aircraft was in flight, so as to take radiation measurements throughout the aircraft. Economics was never a significant issue for this project, nor was the production of radioactive argon (from the 1% argon in air) that would be emitted by the exhaust. However, despite its success, the project was canceled in 1961 in favor of intercontinental ballistic missiles for the nation’s offensive posture needed to assure that World War III would never occur. In the early 1980 period, NASA did revive interest in cargo nuclear-powered aircraft and conducted tests of the survivability of the reactor containment vessel in a high-speed crash. However, the fear of radiation inherent in the minds of the media and public would make serious consideration of constructing a nuclear-powered airplane quite unacceptable in the present irrational fear of radiation by the media and the public.

The program for the application of nuclear power for ocean-going vessels was launched in the late 1940s in the United States and has resulted in a very successful array of nuclear-powered naval vessels, primarily submarines and aircraft carriers. These vessels are considered economical and practical for their military operations, especially with the cost of oil in the \$100 per barrel range. The principal military advantage is never having to refuel these vessels, because they are now designed for a reactor lifetime of about 30 years without refueling. The Nuclear Navy of the United States has been highly successful, with over 80 nuclear-powered submarines and 10 nuclear-powered aircraft carriers in service.

The United States also built and operated one nuclear-powered cargo vessel, but abandoned it as uneconomical compared to the cost of oil-powered vessels, in the 1960 period, when the cost of a barrel of oil was in the range of about \$10. The Soviet Union built and launched two nuclear-powered ice breakers, and after the end of the Cold War were put into service primarily for tourist-associated business involving travel to the poles. The economics of construction and operation of new nuclear-powered commercial ships has not drawn much attention in the last several decades, despite the cost of petroleum hovering around the \$100 per barrel range. Fuel costs for shipping of goods and materials is a relatively small fraction of the wholesale costs.

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## 24.2 Basics of Nuclear Power Costs for Electric Plants

This section will deal with the costs associated with the present types of nuclear reactors that are built and operating, or which are about to be built. The current plants that are being built are classified as Generation III plus light water reactors. (Generation I are those that were built in the 1950 and 1960 era, with electric power outputs in the 50–500 MW range. Generation II are the larger reactors ~1000 MW electric designed and built in the 1970 and early 1980 period, which basically borrowed on the Generation I designs, but were larger.) The Generation III power plants were designed in the 1980 and 1990 periods, with more passive (rather than active) emergency core cooling systems and simplified designs that had fewer valves and pumps. The current plants being built have power ratings in the 1100–1700 MW electric range.

### 24.2.1 Design Permitting and Legal

In the United States, the regulation of nuclear energy and other nuclear applications is done by the Nuclear Regulatory Commission (NRC), which operates quite independent of the Department of Energy (DOE). It has a budget of approximately \$600 million/year, mostly funded by fees assessed to the 100 operating nuclear-electric power plants, plus fees assessed to commercial firms licensed to use nuclear materials. The NRC performs thorough review of all new reactor designs, including modifications to existing reactor designs. Typical design documentation involves about a dozen or more large volumes (in printed form, though most documentation is now provided in electronic format only). Such reviews generally take several years and involve numerous requests for additional information from the NRC to the prospective licensee.

Engineering design activities for obtaining approval and licensing from the NRC will cost the nuclear manufacturer submitting the design many millions of dollars. In addition, the NRC charges the firm submitting the design for the salaries plus overhead of NRC personnel performing the review. Total upfront costs for a new reactor design are probably in the range of several hundred million dollars. These costs are generally included in the FOAK costs for the first such plant built. However, every new plant, built to the approved design, must undergo a thorough review for early site permitting, which can result in costs in the tens of million dollar range for each site.

These engineering, legal, and administrative costs are significant, though they represent a few percent of the total construction costs. Nevertheless, these need to be included in the accounting for purposes of thoroughness of capital cost estimates.

### 24.2.2 Plant Construction Costs

The construction cost of the power plant is a major factor (perhaps the major factor in most cases) in determining the cost of electricity from a nuclear power plant. In using costs for construction of existing plants as a basis for establishing current construction costs, one must account for the variations in the value of the dollar from the time those plants were built to the present.

In the case of the United States, the Consumer Price Index has varied as shown in Table 24.2 where the value of 100 occurred in the 1982 period.

**TABLE 24.2**  
The US Consumer Price Index by Year

Year	Index	Change Per Year Since Last Value (%)
1970	38.8	
1975	53.8	6.9
1977	60.6	6.7
1980	82.4	12.5
1985	107.6	3.8
1987	113.6	4.4
1990	130.7	6.1
1995	152.4	2.5
1997	160.5	1.7
2000	172.2	3.4
2005	195.3	3.4
2007	207.342	4.1
2010	218.056	1.5
2012	229.594	1.7

#### 24.2.2.1 Experience to Date

An example of a Generation II plant is shown in Table 24.3. It is a four-loop Westinghouse PWR, nominally rated at 1200 MW net electric output. Construction began in 1976, and the plant went commercial at the end of 1984. This was one of the last of the plants to be completed in the United States nominally on schedule, and which has had a reasonably successful operating history of nearly 30 years to date. The plant has an operating license extension to 2043.

The construction of the above-mentioned plant occurred at an unfortunate time, experiencing three very severe economic conditions that greatly increased the cost of construction:

- Construction took place at a time when the nation experienced some of the highest inflation rates in its history, as much as 10%/year. The state in which the plant was constructed passed legislation that prohibited a regulated utility from paying for any portion of the plant construction out of earnings (which are regulated by the state's Public Utility Commission) until the plant goes into commercial operation. The rationale for this legislation was that nuclear power is a difficult technology, and the success of constructing and operating such a plant was, hence, considered to be uncertain. Rate payers therefore should not be expected to incur any costs for such a plant until it was proven to operate successfully. Neither was the utility permitted to pay the interest on the construction loans using company earnings.
- The Three Mile Island incident occurred when the plant was approximately half completed. As a result of that accident, the NRC stopped construction on all plants, for a period of time until new regulations could be implemented which would require changes in design and construction that would assure safer conditions for all future plants. The mandated holdup in construction was between six and nine months, depending on how the delay in various aspects of the mandated changes is evaluated. During the delay period, which was of unknown length

TABLE 24.3

Costs of a Typical Generation II PWR

Component of the Cost	Cost in Thousands of \$	Fraction (%) of Total Cost	Fraction (%) Total, Not Including Interest During Construction	Cost in Thousands of \$ Adjusted for Inflation, to 2013
Nuclear steam supply system*	78,000	2.60	4.04	176,791
Turbine and generator	43,800	1.46	2.27	99,275
Other equipment and major materials	559,000	18.63	28.97	1,266,999
Construction labor	364,200	12.14	18.87	825,476
Construction management and O.H.	369,000	12.30	19.13	836,355
Engineering management and overhead	133,200	4.44	6.90	301,904
Legal, regulatory, and related business costs	231,600	7.72	12.04	524,932
Engineering and overhead costs for startup	81,900	2.73	4.25	185,630
Land, insurance, miscellaneous	35,700	1.19	1.85	80,916
Taxes during construction	31,800	1.06	1.65	72,076
Interest paid during construction	1,070,700	35.69	–	2,426,790
Total	3,000,000	100	100	6,797,144

Source: Hitachi, Japan, 2005, pri. comm.

\*Reactor, pumps, heat exchangers, pressurizer, etc.

when imposed, the plant owners did not know whether to lay off construction workers and construction management, and for how long. Once personnel are laid off, there is no guarantee that they would be available when construction could resume.

- The overall effect of these adverse economic situations probably cost not just most of the more than a billion dollars in interest paid during construction, but an estimated \$150 million dollars paid for unproductive labor during the forced moratorium on construction. Furthermore, once the NRC completed its required retrofitting requirements, the cost of implementing these mandated changes cost an estimated \$250,000,000.

Thus, the estimated overall actual net construction cost of the plant was approximately \$1,500,000,000, in the early 1980 decade, nominally. If this cost is inflated to the present (2014) using the Consumer Price Index, the EPC (Engineering, Procurement, and Construction) cost needs to be multiplied by a factor of about 2.9, giving approximately \$4.35 billion, or \$3600 per kW electric.

Experience to date with the eight Westinghouse/Toshiba new design (AP 1000) reactors being built, a typical *Generation III-plus* reactor, suggests that both construction time and

construction costs could be significantly reduced. If that reduction amounts to a saving of 30%, then the estimated construction costs (not including interest during construction) would be approximately \$2500/kW. However, in the next section it is shown that for the four new AP1000 reactors in South Carolina and Georgia, as well as the four being built in China, the construction cost is significantly higher than \$2500/kW.

Estimates from a major Japanese nuclear reactor supplier and constructor (Callaway Nuclear Power Plant, 1989, *pri. comm.*) have indicated that the construction costs of the plants completed in Japan at the turn of the century (2000–2001) were in the range of \$2200/kW (EPC). Inflation to the present would add about 50% to these numbers, and bring these values to \$3300/kW for EPC costs.

Because of the inflated prices for strategic materials (copper and concrete in particular), it is probably appropriate to add 20% to the above-mentioned construction cost estimates, which would put the costs per kW electric at about \$4000. Nominal variation on this cost would be about 30% on the high side, or about \$1200/kW as a deviation range on the high side.

#### **24.2.2.2 New Construction—Construction and Operation Licenses Granted in 2013**

At present, there are two new nuclear plants under construction: one in South Carolina and the other in Georgia. Their cost estimates are shown in Table 24.4.

The data in Table 24.4 is sparse in detail compared to the table of the plant that went operational in 1984 (see Table 24.3). The reason for this dramatic difference comes from the competition now faced in the nuclear industry, as vendors and buyers keep almost all capital-cost data proprietary. The only data that are available are what are mandated to be released to the shareholders of these companies. These data only list total cost and overruns. The interesting part about the information in Table 24.4 is the number that is associated with the financing cost. It is a significant part of the total investment of the power plant.

Because of a current lawsuit between Southern California Electric (SCE) and Mitsubishi Heavy Industries, the cost of four steam generators has been made public. The San Onofre Nuclear Generating Station needed to have its four main steam generators replaced by Mitsubishi Heavy Industries. After operation began, unprecedented wear was found on the steam generator tubes, which ultimately led to the plants being closed. SCE had filed suit against Mitsubishi for an alleged design flaw. During the course of the lawsuit, it was made public that back in 2004, the four original new steam generators cost \$137 million, for each plant.

##### **24.2.2.2.1 Overall Summary**

EPC costs, estimated from all of the above-mentioned sources, would appear to be in the range of \$4200/kW, with an uncertainty, or variation, of  $\pm$ \$500, a nominal standard deviation.

##### **24.2.2.3 Methods of Financing**

In the United States, most large production facilities, whether power plants or manufacturing facilities, are financed through a variety of methods in the free enterprise economy. Among these are

1. Sale of additional stock—Stockholders expect returns in the 10%–15% range, partially in dividends, partially in appreciation. In general, electric utility stocks do not experience much appreciation.

**TABLE 24.4**  
Costs of New Construction

Vogtle Units 3 & 4			Virgil C. Summer Units 2 & 3		
Power output	2234 MWe		Power output	2234 MWe	
Information date	2011		Information date	2010	
<b>Georgia Power Report</b>			No Individual breakdown of cost was available.		
Transmission cost	\$41 million				
Capital cost for construction	\$4.539 billion				
Cost of financing	\$1.662 billion				
Ownership Breakdown			Ownership Breakdown		
Owner	% Owned	Investment in billions of \$	Owner	% Owned	Investment in billions of \$
Georgia Power	45.7	6.201	SCE&G	55	5.536
Oglethorpe	30.0	4.002*	Santee Cooper	45	4.529*
MEAG	22.7	2.9238	Total	100	10.065*
Dalton Utilities	1.6	0.213*			
Total	100	13.34*			
Cost/kW	\$5,970		Cost/kW	\$4,505	

Notes: The bottom line costs in the table represent not only the costs of EPC but, in addition, all of the related financing costs. The EPC portion is in the range of \$4000 to \$4800/kW.

\*These data were calculated based on the assumption of equal investment according to percentage of ownership and were developed from known data of the other companies invested. They are not officially published values from the companies represented (SCE&G, MEAG, Georgia Power).

2. Sale of bonds—Commercial bonds generally require competitive interest rates, which are in the range of 5%–7% in recent years, depending on the bond rating of the company.
3. Borrowing from lending institutions—Highly rated companies can borrow at rates in the range of 5%–6% interest.

Overall, the plant may have financing at an effective interest rate in the 7%–8% range.

A typical large electric power plant (whether fossil or nuclear) is built by a variety of these financing methods.

**24.2.2.4 Amortizing the Construction Costs**

If 6% or 8% is used as a nominal average interest rate on the construction cost investment, and the period in which this is to be amortized involves a significant fraction of the lifetime of the plant, an average annual contribution to paying of the construction cost debt can be figures. For simplification, it will be assumed that the amortization will be structured so that it involves equal payments by the owner over a period of years. For simplification, let's assume 30 years for the amortization period, on a nuclear plant which is expected to have a lifetime of 40–60 years.

The formula for calculating the annual payment is

$$\text{Payment} = [\text{Amount of Loan}] \frac{i}{1 - (1 + i)^{-n}}$$

where “ $i$ ” is the annual interest rate (as a decimal) and  $n$  is the number of years over which the borrowing is to be amortized uniformly.

For example, if  $i = 0.08$ , and  $n = 30$ , the formula shows that the annual payments are 8.88% of the principal borrowed. Based on \$4200/kW, these amortization costs amount to \$373/year/kW, or \$448 million/year for a 1200 MW plant. The factor for a 6% loan over 20 years is quite similar, 8.72%.

Once the loan is paid off, after the 30-year period, for the remainder of the plant lifetime there will be no payments to the bank. During those latter years of operation, the only costs that the plant operator will incur will be normal operating costs, including the cost of fuel. Many of the 104 operating nuclear power plants in the United States now are in this situation, being free of the burden of the amortization costs. However, major maintenance and repair costs, such as steam generator replacements, may also be amortized, adding capital costs later in the plant life.

#### 24.2.2.5 The Economy of Size

The economy of size is a well-known principle of economics in the engineering and construction field. Normally, one would expect that the cost of a plant would be proportional to the output capability of the plant:

$$\frac{\text{Cost A}}{\text{Cost B}} = \frac{\text{Size A}}{\text{Size B}}$$

This equality is rarely true, and a more realistic equation is

$$\frac{\text{Cost A}}{\text{Cost B}} = \left\{ \frac{\text{Size A}}{\text{Size B}} \right\}^n$$

The exponent ( $n$ ) is less than 1 for size ranges that are practical to build. The exact value of  $n$  varies depending on the types of items or facilities being compared, but a value in the range of 0.6–0.8 seems to apply to most situations.\*

(As an understandable practical example, consider the mass of steel in two barrels of different size. Consider barrel A that is 1 meter high by 0.5 meter in diameter. Barrel B is 1 meter high by 1 meter in diameter. The surface area of A is  $(5/8)\pi$  while that of B is  $1.5\pi$ , an increase of 2.4 in quantity of material (assuming it is the same thickness) while the volume that the barrels can hold is four times larger.

One might argue that if the two barrels had to hold the same pressure, the hoop stress, which is the larger stress, would require that the thickness be doubled, resulting in 4.8 times the amount of material for four times increase in volume. However, the weld length would only be 75% larger, for a 400% increase in volume, thus significantly affecting labor costs in favor of the larger size. Some of the major economies of size are in the savings in the costs of auxiliary structure, such as shops, offices, and storage facilities. These are generally the same size and cost to build regardless of the output capacity of the reactor facility.)

\* The economic data presented herein is representative of what were analyzed to be appropriate when the text was written in 2013. To make the data more representative of the economics in 2016, 12% to 15% should be added to all of the cost data, except for nuclear fuel supply and fuel costs of natural gas and oil. Nuclear fuel costs probably escalated between 5% to 7% in those three years, but cost of natural gas and petroleum have plummeted by nominally a factor of 2.

In the case of nuclear power plants, reactor sizes kept increasing to take advantage of the economy of size, until the size reached what were deemed physical manufacturing, construction, or shipping limits. Even then, improvements continued to be made in the heat removal capabilities, resulting in higher volume power densities from the same size plants.

### 24.2.3 Plant Operating Costs, Not Including Fuel

Operating costs include all fixed costs (i.e., those independent of how effectively the plant is operated) plus fuel costs (which are nominally directly proportional to the output electric energy from the plant). Aside from fuel costs, employee costs (salary and benefits) are typically the major component, plus costs of materials and supplies, as well as normal office and facility expenses, light, heat, and air conditioning. Other fixed costs include taxes, contractor costs, and licensing costs.

Employee costs will change little regardless of whether the plant is operating or not, because employees are normally salaried. Typical nuclear power plants in the United States today employ nominally 600–800 full-time employees per plant. Those sites which have more than one reactor plant of the same design accomplish the operation with an economy of scale, because some of the employee's functions serve both plants. For example, a two-plant site might employ 600 employees per plant, while a single unit plant would need 800.

These fixed operating costs total in the range of \$100 to \$120 million per year per plant. However, there is a great variation on this value, perhaps as high as 50%. One particularly significant occurrence that would greatly add to these costs is the expense of replacing major plant components, such as steam generators in PWRs. A majority of the PWRs in the United States have replaced, or are planning to replace, steam generators. The cost of such a replacement is at least \$100 million per plant.

### 24.2.4 Fuel Costs

Fuel costs are composed of the cost of mining uranium, the cost of converting it into uranium hexafluoride, the cost of enriching the uranium hexafluoride, the cost of converting the enriched material to  $\text{UO}_2$ , and the cost of manufacturing the fuel assembly.

#### 24.2.4.1 Mining of Uranium

Uranium is found in a wide variety of ores, in various chemical forms, usually with valence of +4 or +6. After some refining near the mine, the most common form of uranium is known as "yellowcake," and it consists of a nitrate and sulfate of uranium. This is then refined to become  $\text{U}_3\text{O}_8$ , which is the form that is sold on the market exchanges.

Uranium prices have fluctuated significantly over the last several years. From 2003 to 2006, uranium prices averaged about \$20/pound of  $\text{U}_3\text{O}_8$  (\$23.60/pound of uranium). The spot price market peaked at nearly \$90/pound early in 2007, with the average spot price in the 2012–2013 being in the \$50/pound ( $\text{U}_3\text{O}_8$ ) range.

The  $\text{U}_3\text{O}_8$  is put through a chemical conversion process to produce  $\text{UF}_6$  (uranium hexafluoride) gas, which sublimates from a solid to a gas at 53°C (127°F) at one atmospheric pressure. The hexafluoride forms the feed for both the gaseous diffusion and the gas centrifuge enrichment processes. Cost of the conversion process is in the range of \$10/kg of uranium product.



**TABLE 24.5**SWUs Required for a Given Enrichment with a Tails Fraction of  $X_w = 0.002$ 

Final Product Enrichment (%)	Ratio of Feed to Product	Equivalent Units of Separative Work, Per kg of Product
3.0	5.384	4.254
3.5	6.346	5.352
4.0	7.307	6.472
4.5	8.269	7.609
5.0	9.231	8.760

**24.2.4.2 Enriching the Uranium to Higher Content of U-235**

Natural uranium has an isotopic content of U-235 of only 0.72% (i.e., fraction = 0.0072). Present-day light water reactors are designed to use uranium with an enrichment of between 3% and 5% (see Table 24.5).

Enriching the uranium was initially done by electromagnetic separation (equivalent to a mass spectrograph device). These devices consume much electricity to create high magnetic fields, and thus are a very expensive method of enriching the product, primarily because of being very energy intensive. Each separator uses ion beams bent in a magnetic field, and only a few grams could be produced by a single unit in a day. The electromagnetic separators were built in large numbers, inserted into a huge iron core magnetic ring at Oak Ridge National Laboratory beginning in 1943, during World War II. In 1944, construction began on a gaseous diffusion process facility at Oak Ridge. The gaseous diffusion method has been the method that has been used for enriching uranium by the United States, Russia, and United Kingdom over the last 50 years. In this process, the gas is forced (by pumps) through a ceramic membrane. The U-235 hexafluoride migrates slightly faster than does the U-238 (a theoretical enrichment factor of 1.0043). When flow through and product withdrawal are considered, the enrichment factor per stage drops closer to 1.00. In optimum design, the enrichment factor per stage is only about 1.0022. It requires more than 200 stages to reach the enrichment of 4.5% typical of modern PWR plants. It takes many thousands of stages to reach the enrichment needed to make U-235 of sufficient purity to make an effective nuclear weapon (nominally 93% or more U-235).

At one time, the United States had three gaseous diffusion enrichment plants, at Oak Ridge, TN; Portsmouth, OH; and Paducah, KY. All have been closed down and are being decommissioned. The European Union, with the consortium known as URENCO, built a number of gaseous centrifuge enrichment plants in the 1970s in the Netherlands and the United Kingdom. This technology is much more efficient and uses considerably less energy than the gaseous diffusion process.

One centrifuge facility has been built and began operation in 2012 in the United States, Louisiana Energy Services plant in Lea, NM. This plant uses the URENCO technology.

The U.S. Enrichment Corp. (USEC) is planning on using its own newly developed centrifuges for a pilot plant at Pikeston, OH, with the full-scale plant to be built at the existing Portsmouth, OH, site of the shutdown diffusion plant.

AREVA has announced plans to build an enrichment plant near Idaho Falls, ID, using the URENCO technology.

The cost of enrichment is calculated in terms of separative work units (SWUs). The number of SWUs needed to enrich to a certain level depends on the waste product that is

permitted. This waste is termed the “tails” and the enrichment of the tails is known as the “tails assay.” Typically, this waste product is run at 0.2% U-235.

The equation to determine the ratio of feed to product is

$$\frac{F}{P} = \frac{X_p - X_w}{X_f - X_w} \quad (24.1)$$

where:

$X_p$  is the product fraction of U-235

$X_w$  is the waste fraction

$X_f$  is the feed fraction (0.0072)

The equation for number of SWU required per unit of product is

$$\frac{SWU}{P} = (2X_p - 1) \ln \frac{X_p}{1 - X_p} + \frac{X_p - X_w}{X_f - X_w} (2X_w - 1) \ln \frac{X_w}{1 - X_w} - \frac{X_p - X_w}{X_f - X_w} (2X_f - 1) \ln \frac{X_f}{1 - X_f} \quad (24.2)$$

Using Equations 24.1 and 24.2, the feed required and the number of SWUs can be determined.

The average cost of an SWU in 2012–2013 was in the range of \$140.

Thus, the requirement for producing 1 kg of product enriched to 4.5% U-235 is to supply 8.269 kg of  $U_3O_8$ , costing about \$50/pound (\$110/kg) and paying for 7.609 SWU. This equates to a cost obtaining 1 kg of enriched uranium with a waste stream of 0.002 fraction, is  $\$110 \times 8.269 = \$910$  for the natural uranium, approximately \$10 for converting it to uranium hexafluoride, plus  $\$140 \times 7.609 = \$1065$  for the enrichment service for a total of approximately \$1985 for the 1 kg of product.

The cost for an SWU has tracked fairly well with inflation during the last decade and is not expected to vary significantly in the near future. This is because there will be very few new reactors going into operation in the next three years. Once one or both of the proposed new gaseous diffusion plants go into operation in the United States, it is expected that the cost of an SWU might decline because of competition.

#### 24.2.4.3 Manufacturing of the Fuel Assembly

After leaving the enrichment plant, the hexafluoride is converted to uranium dioxide ( $UO_2$ ), at a nominal cost of \$10/kg, and sent to the fuel manufacturing plant. There the small fuel pellets are manufactured to approximately 0.3 inches (7.5 mm) in diameter, about 0.5 inches (12.5 mm) long, and sintered to give them handling integrity. The pellets are loaded into long thin zirconium alloy tubes which are then assembled into a fuel assembly. The typical PWR fuel assembly consists of a  $17 \times 17$  square array of rods, 264 of which contain uranium. Of the remaining 25, the central one is for instrumentation and the other 24 contain either control rods or are vacant. The manufacturing cost for making one of these assemblies is an item of negotiation for each fuel contract. It is estimated that the nominal cost for manufacture is \$500,000 per assembly.

The total cost of a typical fuel assembly that contains about 450 kg (about 1000 pounds) of uranium (in the form of the oxide) is as follows. To obtain the 450 kg of 4.5% U-235 fuel requires 3721 kg of natural uranium feed material (about 9652 pounds  $U_3O_8 = 8186$  pounds of uranium) for approximately \$410,000 at \$50/pound, and utilized 3424 SWU, \$480,000 at \$140/SWU. The conversion to the hexafluoride and back to the oxide probably accounted

**TABLE 24.6**

Cost of a Single PWR Fuel Assembly

Cost of One PWR 17 × 17 Fuel Assembly (450 kg of U)		Cost/kg of Fuel	% of Total
Cost of raw materials	\$410,000	\$911	28
Cost of enrichment	\$480,000	\$1067	33
Cost of conversion to UF <sub>6</sub> and then to UO <sub>2</sub>	\$50,000	\$111	4
Cost of manufacture of the assembly	\$500,000	\$1111	35
Total cost of the assembly	\$1,440,000	\$3200	100

for \$90,000. The manufacture of the assembly from the enriched oxide costs in the range of \$500,000. In addition there are shipping costs (see Table 24.6).

#### 24.2.4.4 Cost of Fuel per kWh

To determine the cost per kWh, it is necessary to know how many kWh of energy the fuel assembly will generate before it is discharged. In the early days of Generation II PWR plants, the average burnup that a fuel assembly would experience was 33,000 MW per metric ton of heavy metal (uranium and produced plutonium). More recently, the assemblies have been run to burnups exceeding 60,000 MWD/MTHM.

If a 60,000 MWD/MTHM value is used, for a PWR with 34% thermodynamic net efficiency, then the fuel cost for an assembly of 450 kg delivered for approximately \$1.44 million is

$$\$1.44\text{E}6 \text{ to produce a total of } \left( 0.34 \times 60\text{E}6 \frac{\text{kWD}}{\text{ton}} \times 24 \frac{\text{h}}{\text{D}} \right) \times 0.450 \text{ tons} = 225\text{E}6 \text{ kWh}$$

⇒ \$1.44E6/225E6 kWh = 0.64 cent/kWh.

One needs to add to this the 0.1 cent/kWh tax imposed by DOE for disposal of the used fuel, that is, 0.74 cent/kWh.

This cost is representative of what to expect in the future, for fuel procurement contracts being contracted with the above-mentioned economic conditions. Much of the fuel being used currently was purchased when the cost of uranium was ~\$10/pound of U<sub>3</sub>O<sub>8</sub>, about 25% of the average price in 2007. The enrichment prices when the fuel was enriched were nominally in the range of \$80 to \$100/SWU. These two factors alone would reduce the price of the fuel assembly to \$1.05 million, reducing the fuel cost to (0.56 + 0.1) = 0.66 cent/kWh.

#### 24.2.4.5 Total Cost of Nuclear Power per kWh

Consider a nuclear plant (PWR) with net 1200 MW electric power output, operating at 90% capacity factor. It will produce 9.46E9 kWh/year:

The amortized capital cost (from Section 24.2.2.4) will be \$448,000,000/9.46E9 kWh = 4.73 cents/kWh.

The fixed operating costs (not including fuel, see Section 24.2.2) will be approximately \$120,000,000/9.46E9 kWh = 1.27 cents/kWh.

The fuel costs would be 0.74 cent/kWh (see Section 24.2.4.3).

**Total costs would be 6.7 cents/kWh.**

Operating costs, including fuel, would be 2.0 cents/kWh.

The average operating costs, including fuel, for nuclear power in the United States in 2007 was 1.7 cents/kWh. This figure has drifted upward and in 2012 was in the range of 2 cents/kWh, if the plants that were disabled for most of the year are excluded from the average.

Obviously, the capital cost is the major contributor. Once the plant is amortized, only the operating costs would apply, reducing the effective costs by 60%. Because most of that plants have been fully paid for (costs have been amortized off), the majority of the plants had net cost per kWh in the operating cost range of 2.0 cents/kWh. Plants undergoing major refurbishment, such as steam generator replacement, would need to add in the amortization of these replacement capital costs.

The above-mentioned analyses apply to economic conditions in the United States, in the 2013 time frame. Capital cost amortization can be much different in other countries, depending on the effective interest rate. Japan, for instance, has economic arrangements which result in an effective interest rate in the 2% range. Countries with high inflation rates might have much higher interest rates than those currently applying in the United States.

In 2013, wholesale power rates in the United States have been in the 3 to 5 cents/kWh range, except during peak power periods. Thus, nuclear power, from existing plants, would appear to be quite competitive economically, with the current power rates. Obviously new plants, with the high capital cost, if amortized over a 30-year period, do not appear economical in the present electrical supply market. However, it is quite likely that the average wholesale electrical prices will increase in the next several years, to be more in line with the prices in the prerecessions days (before 2008).

#### **24.2.4.6 The Government Involvement**

The High Level Nuclear Policy Waste Policy Act was passed in 1980 and amended in 1987 to designate Yucca Mountain as the nation's underground waste storage site. Under this act, all nuclear power was taxed at 0.1 cent/kWh, and these taxes have been collected by the government for the last three decades. Currently, the collections amount to about \$780,000 million per year, which were being applied to the development of the Yucca Mountain high-level waste storage site. The act called for the US DOE to take possession of the used fuel beginning in 1998. Because this has not occurred, and the spent fuel pools at the various operating plants approached full capacity, the plant operators had no choice but to begin storing the oldest fuel outside in dry storage casks that could also be used for shipping when (if?) the Yucca Mountain site is ready to receive used fuel.

Because of the expense of building dry storage casks, a number of plant operators brought law suits against the DOE asking for some relief on these extra costs, and such has been granted by the courts. There is a move in Congress to completely suspend the 0.1 cent/kWh tax.

It is estimated that Yucca Mountain will cost a total of at least \$52 billion to be put into operation and operated through to 2050. However, the specifications for the fuel stacking density in the current law for the repository, the fuel from the current plants through to 2010 will be enough to fill the facility. There have been suggestions that the law could be amended to permit denser storage of fuel, thus extending the lifetime of the facility. If this is not done, then 0.1 cent/kWh will not have been enough to cover the costs of building the Yucca Mountain facility and operating it through to 2050, plus the expense of planning for an additional repository.

In summary, the future of the 0.1 cent/kWh tax and the method of disposition of the spent nuclear fuel is uncertain in the United States. What is certain is that only 7% of the

nuclear energy in the current fuel assemblies has been utilized when the fuel is discarded. It is not environmentally prudent to “dispose” of this fuel. It should go into “temporary” storage until the government and industry decide that a closed fuel cycle needs to be implemented to utilize a larger fraction of the nuclear energy in the fuel before permanent disposal. Ideally, the only permanent disposal should be of those fission products for which no useful purpose exists.

#### **24.2.5 The Effects of Capacity Factor**

The capacity factor represents the fraction of the energy generated by the facility divided by the maximum energy that could be developed if the plant ran at 100% capacity over a given period of time. Both amortization costs and most of the operating costs (except for fuel) are a constant regardless of whether the plant operates or not. Hence, it is essential that the highest capacity factor possible be achieved with these plants, because amortization and nonfuel operating costs represent over 80% of the above-mentioned cost analysis results. Fuel costs per kWh are not dependent on the capacity factor.

During the early days of operation of most of these plants, capacity factors were in the 50%–60% range. In the 1980s, a great effort by plant operators was put into increasing the capacity factor, and those efforts were successful. During the last decade, the average capacity factor for the nation’s 104 nuclear power plants has been approximately 90%. This is approaching the practical limit, because refueling generally takes at least 1 month of every 18 months, amounting to an automatic 6% loss in capacity. It can be expected that plant owners and operators will attempt to increase the capacity factor close to the maximum achievable 94%.

#### **24.2.6 The “Used” Power Plant Market**

With deregulation, most of the nuclear plants are now merchant plants, dedicated to selling power on the open market at as high a price as possible. Some plants have minimum commitments to the parent company and its wholly owned utility.

Beginning in the mid-1990s, a number of plant owners put their plants up for sale. That was a period when nuclear power was thought to have little future, and the first of these plants sold at an unexpectedly low price. Recent selling in 2006 saw three plants (the two Point Beach plants and the Palisades plant) being sold for about \$700 million per 1000 MW (\$700/kW). This is about 30% of the cost of new plants. All three of these plants began commercial operation in the 1970–1972 period, and hence have seen nominally the full 40 years of their original license. All three have filed or will file for and receive 20-year life extensions. Hence, the price at which these plants were exchanged would seem fair, because they then only had a bit more than one-third of a total lifetime expectancy remaining (nominally 22 years of 60 years).

#### **24.2.7 Decontamination and Decommissioning**

Present operating nuclear power plants are generally required to create a sinking fund for eventual decommissioning of the facility. A number of power plants in the United States have been decommissioned, with the procedures and costs reasonably well documented. However, many of these decommissioning projects were learning projects. In the United States, 28 plants have been shut down and most of these have been decommissioned and the area put to reuse. Only eight of these have been of 500 MW electric class or larger. Perhaps the most meaningful of these projects was the 1095 MW Trojan pressurized water

reactor plant, which ceased operation after a successful 16 years of power production. The decommissioning effort began in 1995 and was completed in 2003. Total cost for decommissioning the facility and returning it for reuse for a nonnuclear purpose was approximately \$450 million. However, many believe that this was a most auspicious case, and shouldn't be used as a typical case, because most of the other decommissioning cases have involved much larger sums. However, because it was accomplished with \$450 million, hopefully the experience with Trojan can be applied with similar success in future decommissioning projects.

If the \$450 million is discounted by 2% a year for the next 60 years for inflation, the value needed for decommissioning in 60 years will be approximately \$1.5 billion. The annual contribution to a sinking fund to reach the \$1.5 billion in 60 years at an interest rate of 6% will be \$2.8 million per year, or only about 0.03 cent/kWh, the equivalent of two days' worth of power sales from the plant. Thus, the sinking fund approach for a decommissioning fund has little impact on the cost of power from the plant.

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## 24.3 Issues That Will Affect Future Costs

In recent years, prices for many commodities and materials have been escalating at a very rapid rate. Two of these materials most relevant to new power plant construction are copper and concrete. Obviously, these cost increases in basic materials will greatly impact the costs of a variety of finished products, including the construction of nuclear plants and the manufacture of the fuel assemblies. In the following section, an attempt has been made to discuss the impacts of a number of items and issues on the present and future economics of electricity generated by nuclear power.

### 24.3.1 Water Availability and Cost

All thermal generating systems require cooling, and the most common method of cooling large facilities such as nuclear power plant is by evaporative cooling. The second most prevalent method is by direct once-through cooling. This method has been subject to some environmental restrictions on allowable increase in temperatures of the body of water a short distance from the discharge location. A recent requirement from the EPA is that all new plants use evaporative cooling instead of once-through cooling.

Evaporative cooling is accomplished typically in one of three ways:

1. Use of natural draft cooling towers, typically about 170 meters (555 ft) high
2. A number of small, low-to-the-ground fan forced cooling towers
3. Large man-made lakes, built for the purposes of providing cooling water for the power plant

In all of these cases, there is evaporative loss of water to the atmosphere, typically equal to about 24,000 acre feet (30 million cubic meters) of water per year for a 1200 MW electric plant operating at a 90% capacity factor during the year. In areas where water is scarce, and in high demand for agriculture irrigation, this amount of water represents that which would be used for growing approximately 12,000 acres of crops.

In terms of costs that must be paid for the water, this is usually trivial in terms of other costs, assuming that water rights can be bought in the region. The main issue is the regulatory environment that may make it impossible to obtain the needed water rights, or subordinate those rights to more senior water right holders during period of drought. Consequently, in regions of the country where water is a scarce commodity, it may be essentially impossible to operate large power plants such as the current light water reactors.

This water requirement is generally trivial for areas with major rivers and where there is abundant rainfall. For instance, the Missouri River, with a typical flow of about 70,000 cubic feet per second near its mouth where it meets the Mississippi, has 2000 times the flow rate compared to the evaporative cooling needs of a 1200 MW electric light water reactor plant.

Dry cooling can be considered but is usually not reasonable for the current Generation III light water reactor plants, which have a thermodynamic efficiency of about 34%. Dry cooling has been planned for or is being used for some reasonably large gas-turbine/combined-cycle plants. These have thermodynamic efficiencies of 60%, and hence have heat rejection requirements only about one-third as much for the same electrical output nuclear plant. Whether or not dry cooling will even become feasible for large nuclear installations will depend on the need for establishing such facilities in water-poor regions, currently (2014) where natural gas is unavailable at a reasonable cost.

### 24.3.2 Uranium Supply and Demand

The highly fluctuating market for uranium yellow cake during the 2007–2009 period is largely the result of reaching the end of importation of highly enriched fuel from Russia that was being down blended to make the 3%–5% enriched fuel for current light water reactors. Many uranium mines in the United States had been closed for the last two decades because of the low demand for yellow cake. Once these mines are reactivated, and prospecting begins again for new uranium, it is possible that the price for the raw material may drift back from its highs of \$130/pound (spot price) to its previous somewhat stable range of \$20 to \$30/pound. In 2013, the average price of uranium on the world market was in the \$50/pound range. Significant new solution mining operations have been put into production in the Wyoming and adjacent areas. As these operations begin to produce more uranium, at a cheaper production price than that from traditional mining, it is likely that the raw material cost will trend down more toward the \$30/pound of yellow cake, reducing the current order price of a fuel assembly by about 15%.

### 24.3.3 Enrichment Services

All three of the gaseous diffusion enrichment plants that served the United States for over half a century have been shut down. The last of the highly enriched Russian uranium for down blending was received in the United States in 2013. In recent years, three new centrifuge enrichment plants were being planned for the United States. One of these (Louisiana Enrichment Services, plant in New Mexico) has been in operation since 2012. Construction on plants in Idaho Falls (AREVA) and in Portsmouth, OH (USEC), has not yet begun. There appears to be adequate enrichment capacity available in Europe and from the one plant in the United States to handle the world's needs for the next few years, until at least one of the two new proposed plants in the United States are built and operating. Hence, it is likely that eventually the price of an SWU will become more competitive, and should decline somewhat from its current level. However, changes in the price from the current \$140/SWU are not likely to be very significant, but should trend down rather than up, in terms of an equivalent value of the US dollar.

#### 24.3.4 Reprocessing versus Open Fuel Cycle

The current once through, open fuel cycle results in actually fissioning only about 1% of the uranium mined. Approximately 93% of the mined uranium currently exists as low-enriched tails from the enrichment process, and 7% consists of low-enriched material in the spent fuel assembly, which are currently being stored at the plant's (in the United States).

It has long been recognized that reprocessing of fuel is a necessity if the resource is to be used effectively for a long period of time (centuries). The cost of reprocessing, once a plant has been built, in the United States is open to question. The process that will be utilized in the United States in this first (new) reprocessing plant has not yet been established. It is unlikely to be the old tried and tested PUREX process. For that process, still in use in France and United Kingdom, estimates of the cost for reprocessing are in the range of \$2000 to \$3000/kg of mixed oxide fuel, ready for remanufacture into fuel assemblies. This is above the cost of fresh uranium fuel supplied to the fuel manufacturer for today's contract cost estimates (Table 24.2).

A pyrometallurgical reprocessing method was pioneered at the Argonne National Laboratory at the Idaho National Laboratory in the 1980–1994 period. This method is also being developed in Korea, China, and Russia. Hopefully, whatever new process is decided upon, the costs of reprocessing will be less than the above-mentioned estimates for the tried and proven PUREX method (which generates a huge amount of radioactive liquid waste). Furthermore, the fuel manufacture from the reprocessed fuel may require that the manufacturing be done remotely, depending on whether plutonium or minor actinides are left in the reprocessed material. Remote handling for the manufacture process will add to the costs of the fuel manufacture, now approximately one-third the cost of the finished assembly.

#### 24.3.5 Radioactive Waste Storage Issues and Costs

As discussed in Section 24.2.4.6, the Yucca Mountain high-level radioactive waste storage facility is estimated to cost over \$52 billion to finish and operate for the next 40 years. At the current legislative limit of 70,000 metric tons of heavy metal, the cost would be nominally \$750,000/ton. The 70,000 tons of commercial fuel will have generated, on average, only about 13,000 MWD of electricity per ton. At the current tax of 0.1 cent/kWh, this fuel represents an income from tax of \$312,000/ton. Hence, the current tax will have covered less than half the cost of the facility. There is talk about raising the legislative capacity to double the current 70,000 tons, which would not only allow the facility to accept fuel beyond that being discharged up until about 2010, but would provide nearly full cost for the facility from the collected tax.

Whatever finally occurs, Congress will have a say in the tax to be charged on nuclear-generated electricity. However, the fuel cost for the generated electricity is less than 12% of the current wholesale power costs, so even doubling of the current 0.1 cent/kWh would have a rather insignificant impact on the cost of the electricity generation. (However, current legal action is occurring to have the DOE suspend collection of the spent fuel tax until such time that the US DOE can actually accept delivery of spent nuclear fuel from the operating power plants.)

If the fuel were to be reprocessed so that only the fission products were to be disposed off, encapsulated in a suitable glass-like material, the volume that this would occupy in the storage facility would be an order of magnitude less than the volume of the entire fuel assemblies. Thus, the facility lifetime would be significantly extended.



Thus, it would appear that taxing the fuel more than that is done at present could also provide funding for the first of a new generation reprocessing facility, and could also be used to provide some of the operating costs so that using reprocessed fuel could be done at a competitive cost with the once-through cycle involving only fresh uranium.

### **24.3.6 Public Perceptions and Radiation Health Effects**

The prime issue that the public has for nuclear power is the intense fear of radiation. The paradigm in the health physics profession has been the Linear No Threshold hypothesis, which states that the risk of contracting cancer and the risk of death versus dose is a linear function from high doses (where the risks are known) back to zero dose (and zero risk). In the last 20 years, a vast amount of demographic data have accumulated that show that low doses of radiation are not just harmless but actually beneficial to health. The health benefits include less incidence of cancer and longer lifetimes. The doses that seem to have beneficial effects involve levels of as much as 30 times normal background. The rational explanation for this effect is that radiation stimulates the immune system, making the body less susceptible to invading viruses and bacteria, and is better able to attack cancers and other pernicious internal attacks to the body.

Should regulatory agencies recognize this “hormesis” effect of radiation, then the ALARA (As Low As Reasonably Achievable) regulatory policy would be eliminated, reducing regulatory cost, and the costs of site decommissioning would be eased once the current EPA levels of 15 mrem/year (0.15 mSv/year) were increased to more reasonable levels in the range of natural background radiation. Furthermore, if the public recognizes that low doses of radiation are not harmful (but actually beneficial), the irrational fear of radiation that now prevails would disappear, and legal challenges to siting new power plant facilities would be considerably reduced, thus reducing legal costs and planning and construction delays.

### **24.3.7 Economic Infrastructure on Plant Financing**

As previously discussed, the completely free and open market financing arrangements in the United States for the building of new plants is in sharp contrast to the provisions in some other countries with more centrally managed financial markets. Hence, financing for the building of nuclear plants in the United States may not occur as easily as it does in many other countries in the world. Currently (2013), the US Congress has provided loan guarantees for the first few new plants to be built, which lessens the risk associated with the planned financing and construction. Whether or not these incentives will be extended has yet to be determined.

### **24.3.8 Future Issues and Developments**

#### **24.3.8.1 Generation IV Reactors and GNEP**

The new reactors proposed in both the Generation IV plans and the GNEP (Global Nuclear Enterprise Partnership) plans both have higher thermodynamic efficiency and involve reprocessing of the fuel and eventually burning up of the minor actinides. Thus, the requirements for cooling water will be reduced, and there will be enhancement of the move toward reprocessing and reducing the volume to be occupied by the waste sent to a national repository. The goal of both of these new programs is reduction in waste, hence overall reduction in costs associated with waste. The following are the six competing designs, the first three designs are reactors that utilize a thermal spectrum and the last three utilize a fast spectrum:

#### *24.3.8.1.1 Very High Temperature Reactor*

The very high temperature reactor uses a graphite core that permits a very high exit temperature of the helium coolant. High-temperature industrial applications, such as glass making and more efficient generation of hydrogen, are some of the proposed applications, as well as generation of electricity with a high thermal efficiency (exceeding 45%).

#### *24.3.8.1.2 Supercritical Water-Cooled Reactor*

The supercritical water-cooled reactor is a normal light water reactor (LWR) that has increased temperature and pressure. The purpose is to place the water in the primary coolant loop into a supercritical state, which would dramatically increase operating efficiency.

#### *24.3.8.1.3 Molten-Salt Reactor*

The molten-salt reactor uses molten salt as either the primary coolant or fuel. In either case, both are in motion around the core. One such prototype reactor was built and operated at the Oak Ridge National Laboratory in the 1960 decade. The concept is primarily focused on the thorium/U-233 fuel cycle. The reactor primarily operates near atmospheric pressure, allows for continuous removal of fission products, and offers natural proliferation resistance characteristics.

#### *24.3.8.1.4 Gas-Cooled Reactor*

The gas-cooled reactor (GCR) has a gas as the primary coolant, usually helium. With a fast-neutron spectrum, the GCR has the ability to breed fertile uranium and to consume actinides.

#### *24.3.8.1.5 Sodium-Cooled Fast Reactor*

The sodium-cooled fast reactor (SFR) uses liquefied sodium as the primary coolant. The major advantage is that the reactor can operate at atmospheric pressure in the primary coolant loop. The ultimate inherent safety of this reactor, when using metallic (not oxide) uranium fuel, was demonstrated in 1986 with the Experimental Breeder Reactor II at the Idaho National Laboratory, under the direction of Argonne National Laboratory.

#### *24.3.8.1.6 Lead-Cooled Fast Reactor*

The lead-cooled fast reactor uses either lead or lead–bismuth eutectic in the primary coolant loop. This gives similar advantages as with the SFR in terms of operational safety. Several of these reactor designs were built and operated on Russian submarines.

### ***24.3.8.2 Underground Siting of Nuclear Power Plant Parks***

An old concept of siting nuclear power plants underground has been resurrected, and is being discussed in terms of large nuclear parks. The proposed parks would involve half a dozen or more large reactor electric production facilities, as well as reprocessing, fuel remanufacturing, and waste storage facilities. The underground siting offers many other advantages such as enhanced and less expensive security, being virtually immune from terrorist attacks, and placing the disposed waste in the same region that benefits from the production of the electricity. Such activities will require new standards and rules from the NRC and new approaches to financing large nuclear complexes. The net effect has been shown to be reduced costs for the plant construction, as well as reduced cost and elimination of transportation costs for fuel reprocessing and radioactive waste storage.

The main reason for building nuclear plants underground is security from both internal and external threats: access to the plant can physically be limited to a few well-guarded openings. This reduces the threat of an armed force gaining access to the reactor. External threats are reduced as well because there is no way to determine exactly where the reactor is located within the complex. Flying a plane into the complex would prove fruitless because the target cannot be seen. In addition, being located underground there would be layers of dirt 100 meters or thicker to act as a shield, also an advantage against bombs located in vehicles on the ground level.

If constructed in granite or similar competent rock, the underground cavity would serve as the containment, eliminating the cost of building the containment structure. Operating costs would require additional air supply and exhaust fans. However, the additional operating cost would be trivial, and would easily be offset by the reduced operational security expense. The references have shown that using new underground construction techniques, involving huge tunnel boring machines, typically 15 meters in diameter, plus new large diameter vertical hole cutting machines, makes consideration of underground siting economically practical compared to surface siting of these nuclear facilities and complexes.

However, such a huge complex of nuclear facilities in one location underground would involve a massive investment. Such a huge investment would probably need government assistance in terms of loan guarantees. Furthermore, location of such a facility would require selecting an appropriate underground site. A competent hard rock would be preferred, rather than a sedimentary rock. This requirement would significantly limit the number of locations in the United States that would be favorable to such underground siting. However, there are enough locations to accommodate many hundreds (perhaps many thousands) of GW of electrical power.

#### **24.3.8.3 Superconducting National Grid**

The late Chauncey Starr proposed a superconducting national grid. With the underground siting of nuclear power plants, as discussed above, and the desire to have the ability to tie the entire nation together via high-voltage DC transmission lines, the idea of a superconducting grid becomes more essential. The technology for manufacture and use of high-temperature superconducting “wire” made from the ceramic materials that are superconducting at liquid nitrogen temperature makes superconducting transmission much less costly than the use of liquid helium for the well-developed superconducting technology that has been in use for several decades in magnetic resonance imaging (MRI) machines and for high magnetic fields needed in many high-energy physics devices.

#### **24.3.9 Small Reactor Systems**

Some of the Generation IV concepts as well as other developments involve small reactor systems, in the range of a few hundred MW thermal (~100 MW electric). These are being designed for extremely long life operation between refueling. Because of their small size, the units would be built and assembled in a factory, and delivered completely assembled to the reactor site. Hence, these are given the acronym SMRs (small modular reactors). The well-established “economies of scale” appears to be a detriment to the competitiveness of the concept, but there are several considerations that could overcome the “economies of scale” principle, resulting in these SMRs being competitive in the nuclear generation market. Several of these considerations are discussed below:

1. A distributive systems in an electrical grid

- It appears that SMRs could have a definitive role in distributive generation on large area electrical grids, such as providing generation near the end of lengthy sectors extending many hundreds of miles from the main generation facilities. Currently, solar is being considered for such applications, and the difference between generation costs for solar (capital investment amortization and maintenance) is substantially higher than the total costs of new nuclear generation. Even if the SMR generation costs were higher than present large light water nuclear plants, the SMR would appear to be less expensive in long range, and more reliable alternative, than is solar for maintaining electric grid requirements at the end of long-extending sectors.

2. Areas where population is sparse and electrical requirements are small

- These smaller systems are also proposed for isolated areas that do not have the demand for huge amounts of power in the 1000 MW range. Such venues as in Alaska and Hawaii in the United States, and in many of the developing countries in the world, would be candidates for such nuclear facilities. Despite the higher cost of electricity from these smaller reactors, the cost will generally be less than the current cost of electricity in these isolated areas.

3. Less long-range commitment to construction investment will be required

- The generation company would have to only commit, in the near term, to an ~100–2000 MW electric expansion, rather than the huge capital costs of a 1000–1500 MW electric unit. Even if the SMRs were all colocated, these would be added incrementally over a period of time, with the first units producing income long before the final units are completed. In contrast, a large conventional LWR will not produce any income until the entire plant is completed, and construction funds will incur interest charges until the plant is completed (AFUDC — allowance for funds used during construction), and only then will start producing income.
- The following is an economic analysis of the comparison of two such (hypothetical) situations. Both analyses are based on the EPC cost. That assigned to the large LWR is \$4200/kW, which is nominally in the range of what appears competitive in 2014. A factor of 1.42 increase, to \$6000/kW, has been assigned to the SMR, because of the economy of size theory, which is that

From the economy of size formula earlier in this chapter,

$$\frac{\text{Cost}(1)}{\text{Cost}(2)} = \left[ \frac{\text{Size}(1)}{\text{Size}(2)} \right]^n$$

It was noted that *n* is generally in the 0.6–0.7 range. It is recognized that this simple equation needs to be modified because there are economies to be derived from factory production versus field production, in favor of the SMR. In the case of this example, 0.85 has been used for the exponent *n*, to account for the advantage of the manufacture and assembly in a factory setting.

Large LWR Example	Equivalent Multiple SMR Example
One conventional 1500 MWe PWR	Ten 150 MWe SMR units
EPC cost = \$4200/kW	EPC cost = \$6000/kW
Total EPC cost = \$6.3 billion	Total EPC cost = \$9.0 billion

Large LWR Example	Equivalent Multiple SMR Example
Time for construction = 5.0 years	Time for construction = 2.0 years
Operating capacity factor = 90%	Operating capacity factor = 95%
Annual interest rate on AFUDC = 6%	Annual interest rate on AFUDC = 6%
Long-term financing 6% interest for 20 years	Long-term financing 6% interest for 20 years

The following are the results of the analysis of the interest charges during the construction process, and the partial earnings that result from the operation of the first SMRs, the first of which goes on line at 2 years and 8 months from the start. A new SMR goes on line every 4 months thereafter. Each has a construction period of 2 years.

	Large LWR (1500 MWe)	10 Small 150 MWe SMRs
Total interest during construction	(\$972 million)	(\$380 million)
Net earnings on power sales during construction	0	\$591 million

Net earnings are based on a wholesale electricity price of 7 cents/kWh, and a production cost of 2 cents/kWh, for a net gain of 5 cents/kWh.

The net overall effect is a credit of \$1184 million in savings for the SMR during the construction period. This can be used to offset the construction cost disadvantage at the end of 5.7 years.

Long-Term Financing Comparison	Large LWR	10 Small SMRs
EPC	\$6,300 million	\$9,000 million
Net AFUDC interest	972	(221) net earnings
Total long-term financing principle	\$7,272 million	\$8,779 million

When these amounts are factored into the amortization formula, using 6% annual interest for 20 years (8% for 30 years gives virtually the same result), with 90% overall capacity factor for the large LWR and 95% for the fleet of 10 SMRs, obtains 5.36 cents/kWh for the LWR and 6.13 cents/kWh for SMRs.

This difference of about 0.8 cent/kWh in the capital cost component is probably minor compared to and offset by the lessened risk of being able to make a \$6–\$9 billion investment incrementally over a period of many years, without having to commit to the entire investment “upfront.” Furthermore, the above-mentioned example implies that long-term financing eased, and return on investment begins at about 2.7 years when the first SMR goes into operation delivering power to the grid. Consequently, long-term financing would begin to conclude sooner for the SMR case, an additional advantage.

One of the problems facing the nuclear industry today is the uniqueness of every plant in the United States. Virtually every nuclear power plant has unique pumps, pipes, and various other components. This is a problem because of cost perspective on new construction. There is little mass production in nuclear plant components, so this drives up the cost. The SMR designs are meant to lower the cost of construction by making multiple units that have mass-produced parts.

For the same overall installed capacity, there will be ~10 times more of each of the components, allowing for more data on reliability, and eventual component improvements.

**TABLE 24.7**  
The mPower SMR Specifications

mPower SMR	
Thermal output	530 MWt
Electrical output	155–180 MWe
Vessel size	Diameter: 13 ft (4 m) Height: 83 ft (25.3 m)
Vessel weight	628 tons (w/o fuel) 716 tons (w/fuel)
Reactor coolant	
Pressure	2050 psi (14.1 MPa) – nominal
Core inlet temp	567°F (297°C)
Core outlet temp	608°F (320°C)
Core flow	30 million lb/h (3780 kg/s)
Steam conditions	
Pressure	825 psi (5.7 MPa)
Superheat	50°F (28°C)
Feed water temp	414°F (212°C)

### 24.3.9.1 Status of SMR Design and Production (2014)

#### 24.3.9.1.1 “m-Power”

On April 15, 2013, a Cooperative Agreement was signed between Babcock and Wilcox (B&W) “mPower” and the DOE for the further development and licensing of B&W “mPower” SMR. The key specifications of the mPower design are that the steam generators and core would be the same single unit, the new reactor would be sited underground, there would be passive safety systems, and that it would have a 4-year operating cycle. Each of the key aspects of the mPower SMR is responses to current issues within the nuclear industry. The specifications of that reactor are shown in Table 24.7.

The steam generator and the core being one enclosed unit removes many valves and components that would normally be needed to route the water and steam around the plant. It also keeps all the major contaminated structures as one piece that can be removed with relative ease.

Passive safety systems have been around since the birth of nuclear power plants, but it has only been since Three Mile Island and Fukushima that their role as a “must have” in nuclear construction has been popular and even mandated. With any nuclear reactor, the issue has always been how to remove decay heat when no active cooling can take place. In current LWRs, active cooling is required at all times, otherwise the core would melt down in a matter of days just from the decay heat. The purpose of passive systems is to provide a means of decay-heat removal with no need for any powered pumps or systems that require power. This extends the melt-down time of a reactor for so long as the passive system can remain in operation. The most common means for this in LWRs is to have gravity-fed water flowing through the core. This means usually having a large pool of water above the reactor. Nominally, such pools can keep a reactor cool for about two weeks before running out of water. There are other passive safety systems that help to shut down the neutronics of the reactor so that an uncontrolled excursion is impossible.

A 4-year operating cycle means that refueling no longer has to take place every 18 months. The main benefit is capacity factor increase: more money for the generator and

less down-time. The latter is important if these reactors are to be used in remote locations. Their purpose is to provide power where little other options exist. They could also be used to bolster voltage at the extreme ends of large-area electrical grids.

#### 24.3.9.1.2 NuScale

In December 2013, the US DOE announced an award to the NuScale Power LLC, for its SMR design. The US DOE would cost share with private industry, the cost of development, on a 50%/50% basis. This pressurized water reactor design is as follows:

- Capsule containing the reactor and power conversion equipment: 80 ft high, 15 ft in diameter
- Thermal power: 160 MW
- New electric power output: 45 MW
- Fuel: Similar to Westinghouse standard PWR 17 × 17 fuel assemblies, but only 2 meters in length

The company has estimated that a group of 12 of these modules (producing 540 MW electric) would cost \$2 billion to manufacture. In addition, there would be site preparation costs, providing for in-ground installation of these modules, plus all related control room, engineering, and waste heat removal facilities. If these latter add another \$1 billion to the cost of this 540 MW electric complex, then the cost per kW for EPC costs would be about \$5500/kW.

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## 24.4 Costs of Competitive Power Sources

In this section, some brief comparisons will be made with competitive sources of electricity.

### 24.4.1 Coal and Coal Gasification

Coal-fired power plants are the largest source of carbon dioxide emissions to the atmosphere. Hence, governments are likely to eventually limit carbon dioxide emissions, or place a significant tax on such emissions. This is the major concern when a coal-fired power plant is being evaluated for upgrade to comply with new emission regulations, or when considering construction of a new plant.

EIA data show that the fraction of generation of electricity by coal has dropped from about 50% of that generated in the prerecession period of 2005 to 39% for 2013. Capital costs of pulverized coal plants with complete environmental controls are less than for new nuclear plants, in the range of 75% of a new nuclear plant. But the uncertainty concerning the potential of a carbon tax on emissions has resulted in a reluctance to upgrade old plants, and to avoid considering new plants in preference to natural gas-fired GTCC plants, which emit only one-third as much carbon dioxide as a modern coal plant of the same net electrical output.

The cost of coal is of some concern, more so in places other than in the United States. It has been reported that the world-wide average price of coal increased from about \$60/ton at the beginning of 2007 to \$120/ton at the end of 2007. That tremendous increase has not been the case in the United States. Still at \$30 to \$40/ton (typical cost for western coal delivered to the Midwest or east), the fuel cost for generating power in coal-fired plants in the

United States is in the range of 1.5 cents/kWh, nominally two times the cost of nuclear fuel. The personnel labor costs, not including fuel costs, are substantially less than for nuclear plants. The overall cost in the current markets for electricity from coal plants is quite competitive with nuclear plant costs. Thus, maintaining operation of existing coal plants that need no significant emission control upgrades is a viable business decision.

Coal gasification is rather new in commercial applications. The overall process, demonstrated in about 4 plants in the United States and about 10 elsewhere in the world, is designated as Integrated Coal Gasification Combined Cycle. Coal gasification offers a method of making it much easier to capture carbon dioxide if the gasification process is carried out to produce hydrogen as the resulting gas. Pure oxygen is used in the gasification process, and the CO<sub>2</sub> produced can be physically separated and captured. The gasification process requires a significant fraction of the electricity from the resulting plant output. Adapting the gas turbine to burn hydrogen instead of natural gas (primarily methane) requires some redesign of the gas turbines. The resulting combustion product is water, but with nitrogen oxides produced, a significant pollutant requiring control methods. Separation of oxygen from air is a necessity for the gasification process, in which 0.25 moles of pure oxygen are required for every mole of hydrogen produced.

- First reaction:  $C(\text{solid}) + O_2 + H_2O \longrightarrow 3CO + H_2$
- Second reaction (“water gas shift”):  $3CO + 3H_2O \longrightarrow 3CO_2 + H_2$

However, the physical separation method, needed to separate the CO<sub>2</sub> from the hydrogen, uses an organic gas and requires high temperatures.

The cost of electricity from the few demonstration plants is uncertain, but is most certainly higher than the cost of electricity from pulverized coal plants.

Hence, the future of carbon emission allowances and credits imposed by governments will determine the long-term competitiveness of coal for generating electricity, despite the huge amount of coal reserves that exist in the United States and elsewhere in the world.

#### 24.4.2 Natural Gas-Fired Systems

The newest gas-turbine combined cycle (GTCC) plants have a thermodynamic efficiency of nominally 60%. This is a result of a combined efficiency of approximately 35% for the gas turbine and 38% for the bottoming Rankine cycle. In 2013, the GTCC natural gas-fired plants were the overwhelming type of new electric generating plants being built. With natural gas wholesale prices in the range of \$4 per million Btu, the heat rate for these plants is 5692 Btu/kWh and the fuel cost for the generated electricity is 2.3 cents/kWh. These plants can be contracted and built in a 2- to 3-year time frame.

However, natural gas prices have a record of being rather volatile. For instance, in the 1998–2000 period, natural gas prices dropped to a low of \$1 per million Btu, and was the driving factor in the decision to shut down and decommission the Trojan Nuclear Power Plant in Oregon. However, orders on new installations were reached a backup of 3 years, and within a few years, the natural gas prices peaked at a high of \$14.

GTCC plants have a capital cost which is about half the cost of nuclear plant, amounting to an amortization cost of about 2 cents/kWh amortized over a 20-year period. Operating costs (not including fuel) will be less than for a similar sized coal plant, or about 0.5 cent/kWh.

GTCC plants do have a significant advantage over coal, for the GTCC plants need only about one-third as much cooling water as a nuclear plant, and emit only one-third as much carbon dioxide as does a pulverized coal plant. Another major concern for new capacity



additions is that the gas transport infrastructure in the United States would become inadequate until new pipelines are built.

### 24.4.3 Solar Power

Solar energy is plentiful, and it would take only a fraction (about 10%) of the area of Arizona to generate all the electricity we are now using. Unfortunately, there would be no generation at night or on cloudy days, so the capacity factor would be no better than about 45%. (The IEA reported, for 2012, overall 30% capacity factors for all solar installations in the United States.) Thus, until economical energy storage systems are developed, utilities would need to have back up generation capability for all of their solar installations.

Capital cost is the major contributor to the cost of solar power. With photovoltaic power systems presently costing about \$4 per peak watt, the capital cost of \$4000/kW would be equivalent to \$8888/kW at 100% capacity, equating to 9 cents/kWh when amortized over 30 years (at 8% interest rate). Obviously, this is well above the nuclear capital cost of nominally \$4200/kW with a 90% capacity factor (equivalent to \$4667/kW at 100% capacity factor), and an amortized cost of 4.7 cents/kWh.

Many solar engineers are projecting that photovoltaics will eventually come down to \$1.5/W, putting the equivalent cost at \$3333 for 100% capacity factor, or about 3.4 cents/kWh for amortization costs (30 years at 8%). However, even with such a significant cost reduction, solar will require backup reliable generation capability or energy storage.

### 24.4.4 Wind Power

Wind power is essentially free of fuel costs (except for royalties to the land owners), and has rather low operating costs, compared to nuclear power. Capital costs are in the range of half the capital costs of nuclear power. However, wind has a typical capacity factor of only about 30% (there are a few locations where one might expect as high as 40%). Installed capital costs for wind turbine systems have seen a significant increase during the last year, because of inflated copper and reinforced concrete prices. A nominal installation cost of \$2000/kW is typical. This is equivalent to \$6667/kW at a 100% capacity factor.

Fuel costs are essentially zero, except for royalty payments to land owners. These would generally be about 10% of the selling price of the power, which currently is nominally at avoided costs, in the 7 to 12 cents/kWh range. Thus, the royalty cost would be between 0.7 and 1.2 cents/kWh. (This royalty cost is actually more than nuclear fuel costs.)

Both wind and solar represent favored green energies, which under the PURPA act of 1978 receive favorable treatment from utilities, which must buy these at the avoided cost, even though these sources of power are unreliable because of the weather. In addition, units built before December 31, 2013, receive an additional 1.9 cents/kWh subsidy from the government. Thus, for merchant plants, both solar and wind have very attractive economics presently, and it is likely that Congress will make certain that these advantages are continued.

### 24.4.5 Green Energy and Electric Grid Reliability

One of the main issues today is the effect of greenhouse-gas emissions on global climate. This is a very controversial subject. However, one issue that cannot be argued is that the current base-load power production is primarily fueled by coal and natural gas, both of

which produce CO<sub>2</sub>. Such fuels are not inexhaustible. If we are to change this trend, a change in how we produce our base-load generation must happen.

A power plant is used for base-load power generation when it fulfills certain criteria. The first and most important: the power plant must be able to provide 24/7 power without interruption (not to include scheduled maintenance outages). This is vitally important because consumers pay for 24/7 power, and it is even unlawful for utilities not to provide power without interruption. It is for this reason alone that many renewables are not suitable for use in base-load power. Wind and solar both fall into this category: wind only provides power when wind blows and solar only during daylight hours. Hydroelectric dams are a source of renewable energy, but in North America. As demand increases, all the possible or allowable places that are suitable for a dam have already been established. Such limitations are not necessarily true in much of the rest of the world.

To significantly lessen the growth in greenhouse-gas emissions, the new plants need to use fuel that has a near-zero net impact on CO<sub>2</sub> emissions. Solar cells, while not directly producing emissions for power generation, have a significant adverse environmental footprint in their construction. Furthermore, the manufacture of storage batteries also has a similar chemical waste footprint, which is the current method used to provide the required energy needed at night or in dim daylight hour conditions. Wind energy also has zero CO<sub>2</sub> emissions. Wind installations are similar in price to the building of coal and natural gas plants but have the disadvantage of rarely exceeding a capacity factor of 30%. Furthermore, wind energy production is highly erratic, and largely unpredictable, creating considerable difficulties for electric grid management.

Finally, power plants must be economically competitive. Operators will not invest in more expensive power plants if they do not expect an adequate return on their investment.

A nuclear power plant is the only one that fulfills all criteria. The current fleet in the United States has had a 90% capacity factor over the last 12 years, which demonstrates that nuclear power is capable of providing 24/7 power to consumers. (The reason the capacity factor is not 100% is due to the required one month outage needed for refueling and maintenance activities. Refuelings are scheduled for times of minimum power demand, typically spring and fall.) The waste from a nuclear power plant is solid and not emitted into the atmosphere. The net carbon footprint for the construction, operation, and demolition for a nuclear power plant is almost nothing compared to any other industrial application. Finally, nuclear power has already proven to be highly cost-competitive with other sources of power.

#### 24.4.6 Summary of Cost Comparisons

Table 24.8 compares the factors involved in total costs for these various methods of generating electricity. The construction costs are the most uncertain, or have the potential for the greatest variation, primarily because of the serious inflation that has occurred with construction material prices in the last two years. As for fuel costs, the most volatile are the natural gas costs for the GTCC plants. Construction costs are likely to vary because of site differences, regulatory approval delays, weather abnormalities, and legal challenges during construction.

The construction costs are only EPC costs, and do not include AFUDC, otherwise known as interest paid on the construction funding, up to the time the plant goes into commercial service. These are usually included in stating the costs of nuclear construction, but are not typically included in determining the total construction costs for other types of plants.

**TABLE 24.8**

Comparison Costs of Generation of Electricity

Plant Type	Constr. Cost, \$/kW	Capacity Factor (%)	Amortization ¢/kWh (a)	Operating ¢/kWh(b)	Fuel Costs ¢/kWh (c)	Total Cost ¢/kWh
Nuclear	4200	90(d)	4.2	1.2	0.8	6.2
Coal (1)	3000	95	2.8	0.7	1.94 (e)	5.3
GT/CC	2000	95	1.8	0.5	3.4 (f)	5.7
Solar PV	4000	40	8.7	0.4	0	9.1
Wind	2000	30	6.6	0.3	0.9	7.8

(a) Amortized at 6% over 20 years.

(b) Operating costs, not including costs of fuel.

(c) Fuel costs include ~10% royalty payments to the land owner on which the wind turbines are located.

(d) In 2013, the Nuclear Energy Institute quotes a 92% capacity factor if the three shutdown plants (SONGS 2 and 3 and Ft. Calhoun) are not included.

(e) Based on \$50/ton, 11,000 Btu/pound heat of combustion, 40% net thermodynamic efficiency.

(f) Based on \$6 per million Btu, 60% efficiency.

A recent tabulation by the Nuclear Energy Institute (February 2014) of electricity “generation costs” from the plants in service in 2012 showed that overall costs were 3.94 cents/kWh for multiunit plants and 5.05 cents/kWh for single unit plants. These included debt servicing costs, at least half of which were for the debt incurred from licensed power upgrades, and applications for 20-year license extensions. Other debt service costs (capital cost amortization) were the result of debt refinancing and for major equipment replacements, such as condensers in ocean water cooled plants, and steam generator replacements in pressurized water plants. (The same report quoted overall costs for GTCC natural gas plants as 5.01 cents/kWh.)

## 24.5 Conclusions About the Future Costs of Generating Electricity

### 24.5.1 Available Resource and Known Reserves

Nuclear power represents a long-term significant advantage over other conventional forms of nonrenewable resources. Even using only the present nuclear fuels used in LWR technology and nominally economically recoverable amounts of uranium, the nuclear resource is more than the known fossil resources, coal, gas, and oil. However, if the nuclear industry can achieve an economical fast reactor technology and environmentally acceptable fuel recycle capability, the nuclear resource is virtually unlimited, far exceeding (by orders of magnitude) the coal, oil, and natural gas reserves.

### 24.5.2 Reliable Electric Energy Production

With great environmental pressure to utilize the renewable resources, the necessity of reliable base energy load, and reliable load following capability dictates the need for the always available thermal power plants, utilizing energy conversion by use of fossil fuels or nuclear power. Regulations throughout the United States and the world generally now aim for current technology renewables (all of which are unreliable except geothermal) to be the order of 20% of the available generating capacity within the next 10–15 years. Higher goals

are being considered for the future. However, without energy storage schemes (such as with hydroelectric power), continuing to demand more renewables as a fraction of installed electric capacity virtually requires reliable thermal energy backup of equal capacity.

Once the subsidies for renewables are eliminated, these will become much too expensive forms for new generating capacity compared to fossil fuels and nuclear power, based on current technologies. Breakthroughs in economic competitiveness could occur, such as reduced cost of solar collectors and an economical method of energy storage (other than hydroelectric). But until such developments occur, fossil and nuclear fuels will be required for most of the electricity that is to be generated.

### **24.5.3 Environmental Legislation and Concerns**

#### **24.5.3.1 Fossil Fuels**

The main concern from the use of fossil fuels is the emission of the greenhouse gas, carbon dioxide. Burning of natural gas for producing electricity can be done with the release of only about one-third as much carbon dioxide as is released from current technology coal-fired electricity generation. The electric power industry in the United States is quickly turning to natural gas to replace coal plants. Conversion of existing coal-fired generation with natural gas-fired generation is occurring quite significantly in the United States. In the first decade of the 21st century, coal was producing nominally 50% of the electricity generated in the United States. Now nearing the middle of the second decade of the 21st century, coal is accounting for only about 40% of the electricity generated.

Much tighter emission controls on fossil fuels are being considered in the United States, impacting coal in particular, with regard to emissions of minor pollutants such as mercury. These are further reasons why the industry in the United States is slowly converting its coal electricity generation to “clean burning” natural gas.

#### **24.5.3.2 Nuclear Power**

The perceived environmental effects of nuclear power are the waste disposal issue. Regulations have dictated that nuclear waste disposal must achieve absolute containment for thousands of years, because of the long half-lives of the transuranic elements. However, if the fast spectrum reactor technology and fuel recycle capability can be developed for economical commercial implementation, these transuranics would be burned up in fast reactors, and the waste storage issue would be dictated by the fission products, and could be limited to total absolute containment on the order of only hundreds of years. Political and public understanding and acceptance of this fact needs to be understood, and that will require education concerning the facts. This could take a generation or more to accomplish, especially the understanding that low levels of radiation are not harmful, and actually appear to be beneficial to health.

#### **24.5.3.3 Renewables**

Wind seemingly has no adverse environmental effects until wind generators begun to “pollute” the visible landscape. Solar appears to have no environmental effects except for the chemical wastes from production of solar collectors. Whether these environmental concerns will become issues is uncertain.

Hydroelectricity is an ideal renewable, because it is reliable as well as “dispatchable” (available when needed). However, in the United States environmental concerns have

basically prevented any future conventional hydroelectric facilities requiring new large reservoirs for energy storage. Run-of-the-stream low-head hydroelectric generation has potential, but the total effect of all new developments is likely not to exceed perhaps even 1% of our national electric generation.

Geothermal energy is limited to those few areas in the United States where high-temperature hydrothermal resources exist. Even though current geothermal energy production is accomplished without any significant environmentally observable effects, reservoir depletion has become an issue. That fact, plus the geographical limitation to geothermal energy, make it unlikely to be a significant contributor to electric energy production in the United States or the rest of the world (except in a few unusual locations, such as Iceland).

#### **24.5.4 Economics**

##### **24.5.4.1 Fossil Fuels**

“Clean coal” is by far the least expensive fossil fuel for generating electricity. But the uncertainty concerning the impact of future environmental legislation is causing coal generation of electricity in the United States to become a lower fraction, though still the leading fraction, of the total electricity generation. Even in China, where environmental concerns from virtually unregulated coal generation of electricity have resulted in severe effects on human health, there is serious discussion concerning emissions control. Even with such, coal will still be the least expensive fossil fuel.

Natural gas prices in the United States have been hovering in the well-head price range of \$3.00–\$5.00 for most of the second decade of the 21st century. This low fuel price, the short lead time for GTCC plants from time of order of a plant to construction and placing into commercial operation, and the economic viability of new plant installations of any size have made natural gas the overwhelming choice for new plant construction in the United States.

One seldom mentioned issue, however, is the infrastructure capability for delivery of the natural gas to the growing number of new users. Pipeline capacity is virtually at its limit, and may be the forcing function that will raise the wholesale price of natural gas well above the well-head production price.

##### **24.5.4.2 Nuclear Power**

Although nuclear power is shown in Table 24.3 to be slightly less economical compared to natural gas, the latter has the uncertainty of an unstable fuel price compared to nuclear fuel. However, one of the main disadvantages to nuclear power is the huge commitment of funds to build the current nominal 1100–1700 MW light water reactor plants. It has been shown (Section 24.3.9) that the SMRs offer long-term economic advantages. The long-term commitment of huge funding is eased. However, the true capital cost (in \$ per kW) of these SMR plants has yet to be determined. Such information is not likely to be available until FOAK plants are built and put into operation.

##### **24.5.4.3 Renewables**

As shown in Table 24.3, solar and wind, without the subsidies (incentive payments), are far from being competitive with the thermal fuel generation options. There needs to be a significant cost reduction in solar collectors for solar to become competitive without subsidies. Wind has essentially reached its optimum capital cost for the foreseeable future.

Hence, once the subsidies (economic incentive regulations) that currently apply to most renewables expire, solar and wind will no longer be as highly favored as they currently are.

#### **24.5.5 The Nuclear Energy Situation in 2014 and Aspects That Effect Economics and the Industry**

In the United States, with 99 nuclear plants operating and virtually all having had license renewals (or will be applying for those renewals) for an extra 20 years (60 years total), the original investment in these plants has been fully amortized. They are producing power at a cost of approximately 2.0 cents/kWh. Nominally, 30% of that figure is the cost of the fuel and of waste disposal. This is much less than the overall average wholesale price of base-load power during 2013, which was in the range of 3 to 5 cents/kWh over much of the year. However, much higher wholesale prices generally prevailed in the New England and West Coast areas. Only nuclear and coal plants that have been amortized can operate economically in the current depressed wholesale market. The most efficient GTCC natural gas plants have a fuel cost alone, in today's low prices of natural gas, that is barely competitive in these low wholesale electricity price markets. If the nation continues to recover from the recession, wholesale electricity prices will most certainly rise, making it clearly profitable to operate the GTCC plants with gas costing up to \$6 per million Btu (resulting in a fuel cost of 3.4 cents/kWh). Because these plants can be built and put into operation in about 2 years, in the present economy merchant electricity generators strongly favor additional capacity being from GTCC units, which can be economically added in unit sizes from 200 to 500 MW.

In the United States, the present operating nuclear plants are profitable under these current wholesale electricity price levels. However, new nuclear plants (or new plants of any type) cannot operate profitably under the current low wholesale prices for base-load, firm power. However, new electrical capacity is needed to replace old fossil-fired plants, mostly coal-fired, that are being retired. The principal competitor to nuclear power is natural gas with the GTCC plants. Because these two methods of generating electricity are nominally equivalent in short-to-medium-term economics, the near term future for nuclear power in the United States will depend on many factors besides economics. Will the public be satisfied with whatever long-term solutions are found for the high-level nuclear waste? Will the government promote closing of the fuel cycle, so that the nuclear resource can be used more effectively than merely consuming 6%–7% of the resource mined? Will Generation IV plants, required in closing the fuel cycle, receive government assistance in building the FOAK? Will adequate infrastructure capacity be available to supply the developing needs to use natural gas? Will carbon dioxide emissions become of such great worldwide concern that even the low emissions from natural gas combustion become a major issue? Will the public and the governing officials that the public elects show a realistic attitude toward nuclear energy?

Concerning the rest of the world, can the irrational fears of nuclear radiation be ameliorated? Will concern for carbon dioxide emissions put sufficient pressure on the major energy growth countries of China and India, so that their only option will be to provide most of the rapidly growing demand with nuclear power?

Answers to all of the above-mentioned questions will have a greater effect than the EPC costs on decisions concerning the growth (or decline) of nuclear power relative to the alternatives for generating electricity. These answers will have major effects on the developments in the free-economic nations, where nuclear power and other forms of power generation are heavily regulated by the central governments. And quite certainly the answers to these questions will determine the future in nations where economic decisions are made by the central government.

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# 25

## *Radiation Protection*

Mark R. Ledoux

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Health physics is the science of measuring and controlling the exposure of people and the environment to radiation. Within the context of nuclear power reactors, the tasks of controlling radiation dose to people and to the environment are usually separate tasks performed by different groups. This separation of duties is driven largely by the distinctly different rules and regulations that apply to each function and by the means of measurement and control employed. This chapter is limited to consideration of radiation dose to people. Primarily, this involves controlling radiation dose to workers, both the power plant operators and subcontractors, but can also include members of the public. Acceptable dose limits and exposures may vary depending on the type of work being performed, worker classification, or special circumstance (e.g., occupationally exposed, general public, declared pregnant female, or planned special exposure).

The understanding of health physics begins with an understanding of the underlying fundamental physical characteristics of the radiation that is being measured; understanding the different types of radiation, how each type of radiation interacts with matter, and what effects they might have. Based on how radiation interacts with matter, measurements can be made to determine how much radiation an individual has been, is being, or may be exposed. This quantity is then compared with the applicable regulatory limits to determine if past operations have been, or proposed operations would be, in compliance with those regulations. In many instances, it is necessary to limit, control, or modify how activities are conducted to remain in compliance.

The following section is a brief overview of health physics. The reader is referred to Cember (1996) or Shleien et al. (1998) for additional information and detailed tables of values discussed.

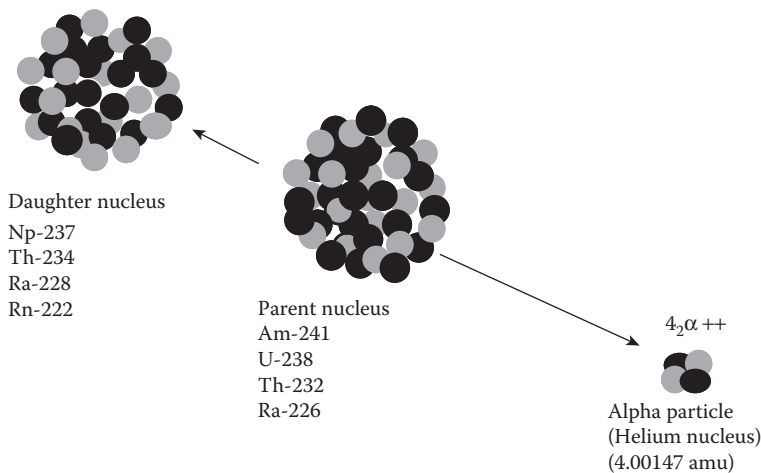
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## 25.1 Radiation Fundamentals

Radiation is the emission or propagation of energy through space. As such, it includes not only alpha and beta particles, gamma and X-rays, and neutrons but also visible light, radio waves, microwaves, infrared, and ultraviolet light. The first five types of radiation are generally emitted during the process of nuclear decay or fission, events involving the nucleus of atoms with the exception of X-rays, and are what we usually mean when we refer to radiation. The last five are different in that they do not typically originate with nuclear events and they are considered non-ionizing radiations, that is, they do not possess sufficient energy to knock an electron out of its orbit in an atom, ionizing the atom.

### 25.1.1 Alpha Particles

Alpha particles are composed of two protons and two neutrons and are ejected during alpha decay of heavy nuclei. The loss of mass leads to a nucleus that is more stable. Common alpha emitters are isotopes of uranium, radium, polonium, and plutonium. An alpha particle can also be thought of as a fully ionized helium-4 nucleus. Figure 25.1 depicts



**FIGURE 25.1**  
 Alpha decay.

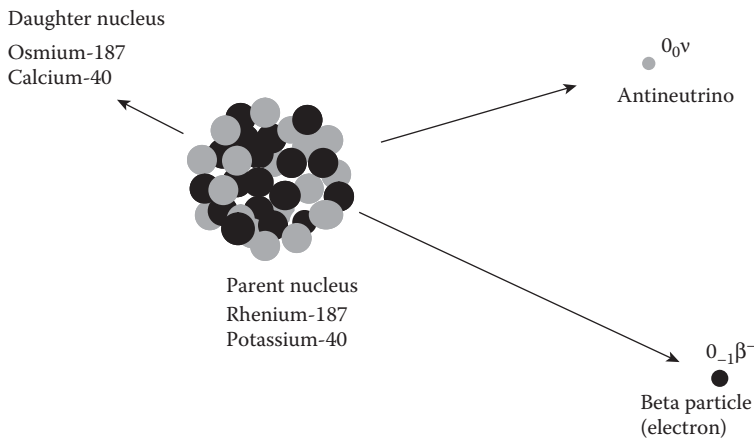
a schematic of alpha decay. A parent nucleus such as americium-241 or uranium-238 spontaneously emits an alpha particle that transforms it into a daughter nucleus such as neptunium-237 or thorium-234, respectively.

This process of decay results in the release of energy that is emitted in the form of kinetic energy of the alpha particle and the recoil, or daughter, nucleus. Conservation of momentum and energy dictates that the alpha particle and daughter nucleus move in opposite directions, and that the alpha particle has the majority of the kinetic energy. For the alpha decay of uranium-234, the alpha particle has >98% of the kinetic energy released in the decay.

Once emitted, the alpha particle is positively charged because it has two protons and no electrons. Because of this +2 positive charge, it immediately begins interacting with all other charged particles in its vicinity. These interactions are called “Coulombic interactions” because it is the electrical attraction and repulsion that causes these interactions. Most of these interactions are with the electrons of the material through which the alpha particle is traveling. As the alpha particle interacts with the electrons, some of the kinetic energy of the alpha particle is transferred to the electrons, slowing down the alpha particle. Each interaction transfers only a small amount of energy due to the large mass difference between the alpha particle and the electrons. Alpha particles are approximately 4000 times as heavy as an electron. However, the large charge and relatively low velocity of the alpha particles (compared with beta particles) result in many Coulombic interactions, causing the alpha particle to stop in a short distance. In air, an alpha particle will travel only a few centimeters before coming to a stop, and it travels only a few microns in solids. The alpha particle tends to travel in a straight line from the point of emission until it stops.

### 25.1.2 Beta Particles

Beta particles are essentially electrons that are emitted from a nucleus during nuclear decay. This type of decay occurs in nuclei that have a neutron to proton ratio that is too high. A neutron is effectively converted into a proton and an electron, lowering the neutron to proton ratio. Many isotopes are produced during radioactive decay and nuclear fission decay by beta emission. Figure 25.2 depicts a schematic of beta decay. A parent



**FIGURE 25.2**  
Beta decay.

nucleus such as rhenium-187 spontaneously emits a beta particle and an antineutrino that transforms it into a daughter nucleus such as osmium-187.

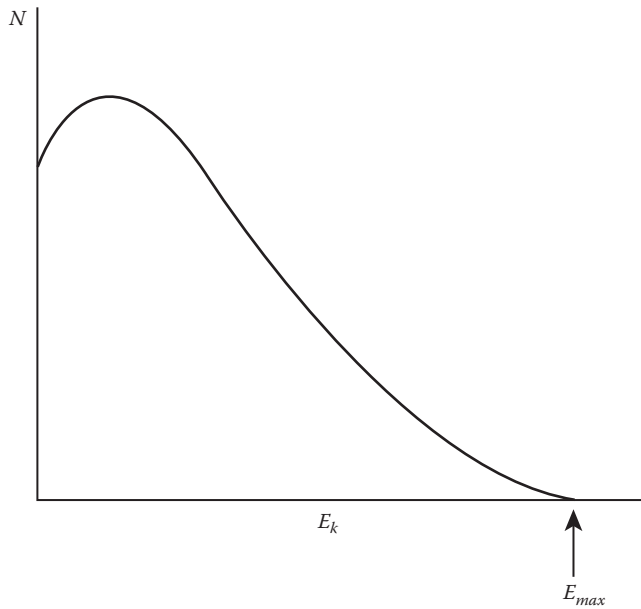
Unlike alpha decay where two particles are emitted, three particles are emitted during beta decay: the beta particle itself, an antineutrino, and the recoil, or daughter nucleus. These three particles share the energy emitted during the radioactive decay in the form of kinetic energy, although the large mass difference (a difference of four or five orders of magnitude) between the beta and antineutrino relative to the recoil nucleus means that essentially all of the energy is carried off by the beta particle and antineutrino. On average, the beta particle has about one-third of the available energy, with the antineutrino having the remainder. However, the beta particle may have any energy between zero and total amount available. Figure 25.3 plots the spectrum,  $N$ , of possible beta particle energies,  $E_k$ , where  $E_{\max}$  is the maximum possible beta particle energy. The antineutrino is ignored because it does not interact with matter.

Once emitted, the negatively charged beta particle immediately begins having Coulombic interactions with all other charged particles in its vicinity. As with the alpha particle, most of these interactions are with the electrons of the material through which the beta particle is traveling. As the beta particle interacts with the electrons, some of the kinetic energy of the beta particle is transferred to the electrons, slowing down the beta particle. Unlike the alpha particle, the amount of energy transferred can be quite large, up to half of the total kinetic energy in billiard ball-like collisions, because the electrons and the beta particle have the same mass. These same billiard ball-like interactions can result in large changes in direction of travel of the beta particle.

The large charge and relatively low velocity of the alpha particle (compared with beta particles) results in many Coulombic interactions, causing the alpha particle to stop in a short distance. In air, a beta particle can travel up to a few meters, and in solid materials and tissue it travels up to a couple centimeters.

### 25.1.3 Gamma and X-Rays

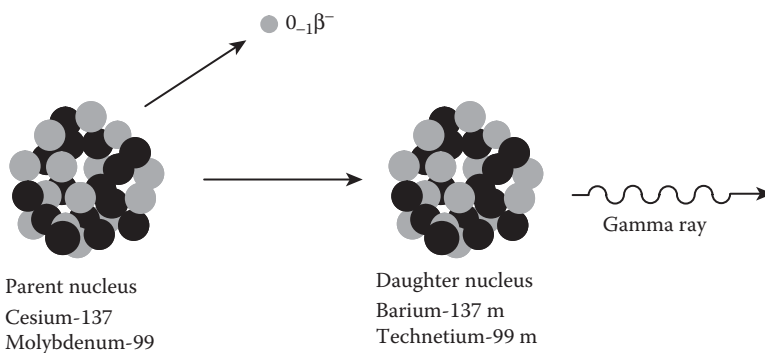
Gamma rays are electromagnetic radiation, that is, the release of excess energy from the nucleus of an atom retained after radioactive decay. For instance, americium-241 decays by emission of an alpha particle as discussed above, but it also emits a gamma ray at the



**FIGURE 25.3**  
Beta decay energy spectrum.

same time. The gamma ray emission can be simultaneous or delayed. Similarly, X-rays are electromagnetic radiation that release excess energy from the orbital electrons of an atom. Physically, gamma and X-rays are identical, with the only difference being the point of origin. Gamma rays are also emitted during fission reactions. Figure 25.4 depicts the radioactive decay of cesium-137 by beta decay to barium-137m, which subsequently decays by gamma ray emission to barium-137. Because the half-life of barium-137m is only two and a half minutes, the gamma ray it emits is commonly associated with the cesium-137 decay.

Unlike alpha and beta particles, gamma rays have no charge or mass; only kinetic energy. Therefore, their method of interaction with materials is markedly different. Instead of



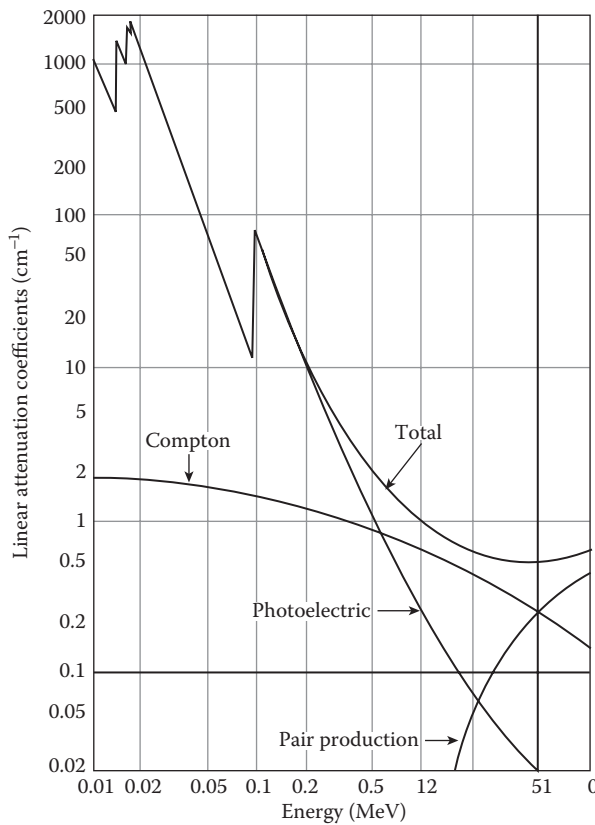
**FIGURE 25.4**  
Gamma ray emission.

having Coulombic interactions with all the electrons in the material around them, gamma rays only interact intermittently. There are three main types of gamma ray interactions:

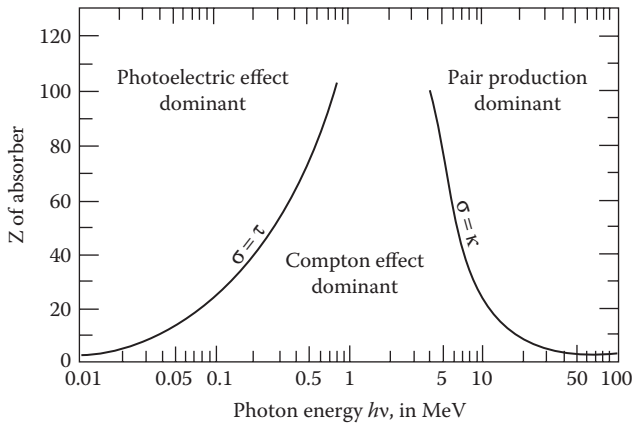
- Compton scattering
- Photoelectric effect
- Pair production

Each of these interactions is a probabilistic occurrence. When a gamma ray passes near an atom or an electron, there is a finite probability that it may or may not interact with that atom or electron. Gamma rays can theoretically travel infinite distances without interacting at all. Figure 25.5 plots the linear attenuation coefficient for lead for each type of interaction. The linear attenuation coefficient is the probability per unit distance that a gamma ray will have an interaction. As can be seen, the lower the energy of the gamma ray, the higher the probability of interaction for energies below about 5 MeV.

The probability of these interactions depends not only on the energy of the gamma ray but also on the atomic number,  $Z$ , of the material. Figure 25.6 plots the relative probability of Compton interactions,  $\sigma$ , with photoelectric effect,  $\tau$ , and pair production,  $\kappa$ . Higher  $Z$  materials, such as lead are more likely to have photoelectric effects than Compton scattering. From the standpoint of shielding gamma rays, photoelectric effect is preferable as discussed later.



**FIGURE 25.5**  
Photon interaction probability in lead.



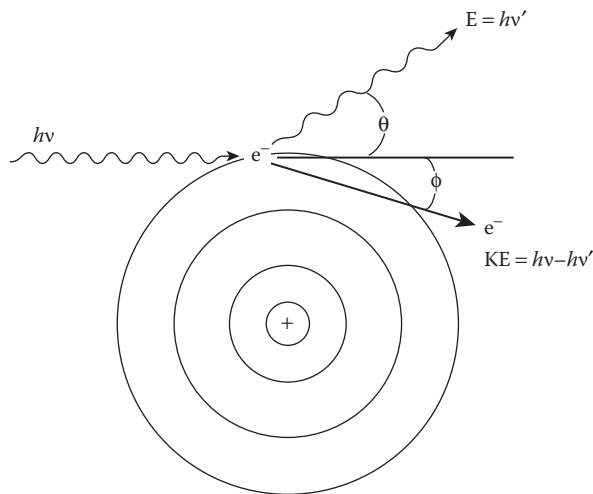
**FIGURE 25.6**  
Relative photon interaction probabilities.

**25.1.3.1 Compton Scattering**

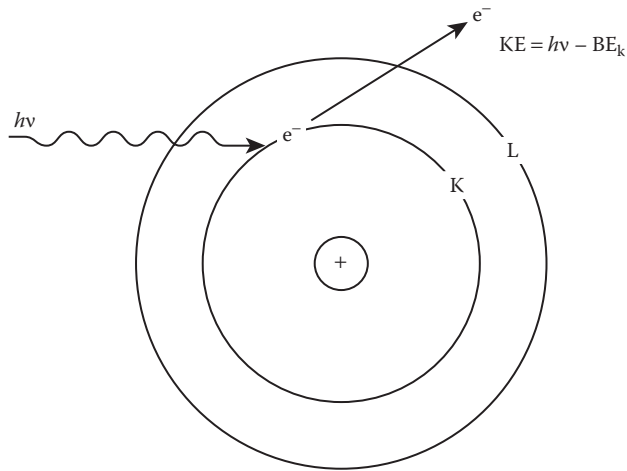
Compton scattering occurs when a gamma ray collides and scatters off of an electron. In the collision, a portion of the gamma ray’s energy is transferred to the electron. Figure 25.7 depicts an incoming gamma ray,  $h\nu$ , colliding with an electron and departing with reduced energy,  $h\nu'$ , and the electron has the remainder of the energy as kinetic energy. The original gamma ray is reduced in energy but continues on until it has another collision.

**25.1.3.2 Photoelectric Effect**

The photoelectric effect occurs when a gamma ray is absorbed by an electron and the kinetic energy of the gamma ray is transferred to the electron. Figure 25.8 depicts an incoming gamma ray,  $h\nu$ , being absorbed by an electron that then shoots off with a kinetic



**FIGURE 25.7**  
Compton scattering.

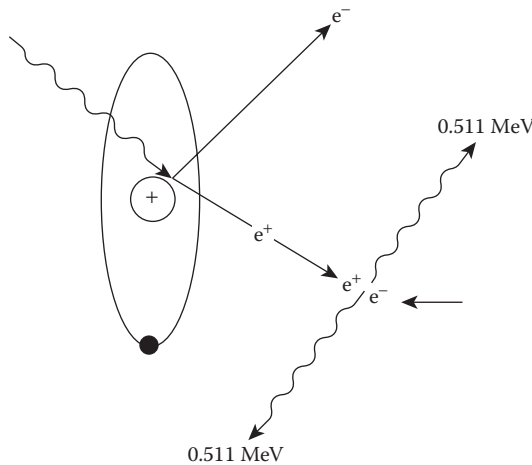


**FIGURE 25.8**  
Photoelectric effect.

energy equal to the incoming gamma rays energy minus the binding energy of the electron to the atom. The original gamma ray has been stopped and eliminated and replaced by an energetic electron that behaves exactly the same as a beta particle.

**25.1.3.3 Pair Production**

At energies above 1.022 MeV, an effect known as “pair production” can occur. This is when a gamma ray is converted into a positron and an electron in the Coulombic field of a nucleus. The positron and electron behave like beta particles and slow down. The positron will eventually collide with another electron, annihilating both particles and producing two more gamma rays, each of 0.511 MeV. While the original gamma ray has been stopped, two more gamma rays are produced, albeit of lower energy. Figure 25.9 depicts this chain of events.



**FIGURE 25.9**  
Pair production.



### 25.1.4 Neutrons

Neutrons are most commonly encountered as a result of fission reactions or  $\alpha$ -n reactions where an alpha particle collides with a nucleus resulting in the emission of a neutron. Beryllium or fluorine is used with an alpha emitter to create neutron sources such as Pu-Be, Po-Be, Am-Be, or PuF<sub>4</sub> sources. Because neutrons have no charge, they like gamma rays, do not have Coulombic interactions, and interact only intermittently. Each of these interactions is a probabilistic occurrence. There are three main types of neutron interactions:

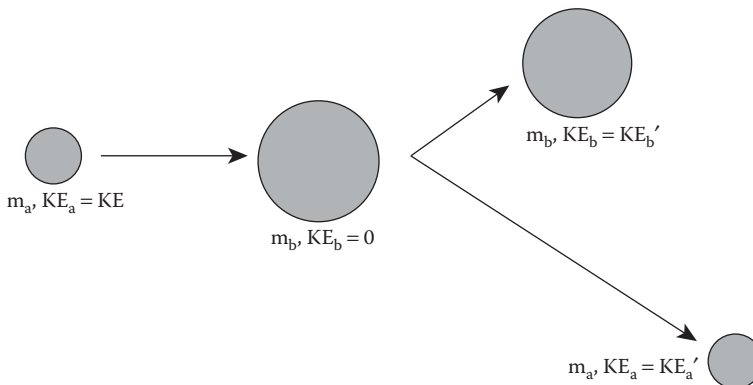
- Elastic scattering
- Inelastic scattering
- Absorption

#### 25.1.4.1 Elastic Scattering

Unlike alphas, betas, and gammas, neutrons do not typically interact with electrons. Neutrons interact with the nuclei of the atoms of the material they are in. One type of interaction is elastic scattering. This type of scattering is a classical physics type of scattering where momentum and energy is conserved. Figure 25.10 depicts this process. The amount of energy transferred depends on the mass of the atom that is struck and the scattering angle. If a lightweight atom such as hydrogen is struck head-on, 100% of the neutron's kinetic energy could be transferred, whereas a glancing blow could transfer very little. A head-on collision with a heavy atom such as uranium would result in very little energy transfer and the neutron would simply bounce off in a simple reversal of direction. The process of neutrons slowing down is a series of scattering collisions, resulting in the flight path of neutrons through materials being very random, similar to that for beta particles.

#### 25.1.4.2 Inelastic Scattering

At higher energies, neutrons undergo inelastic rather than elastic scattering. During inelastic scattering, momentum and kinetic energy is not classically conserved. A portion of the kinetic energy of the incoming neutron is absorbed by the nucleus of the atom, exciting



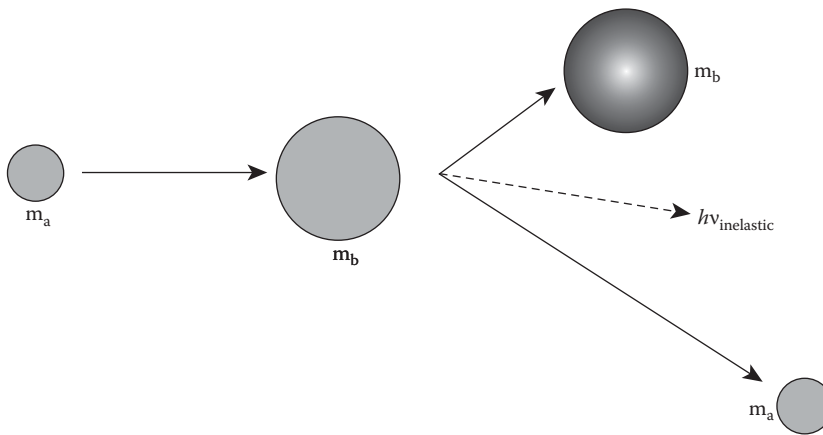
**FIGURE 25.10**  
Neutron elastic scattering.

the nucleus. This excitation energy is then released in the form of a gamma ray,  $h\nu_{\text{inelastic}}$ . Figure 25.11 graphically demonstrates this process.

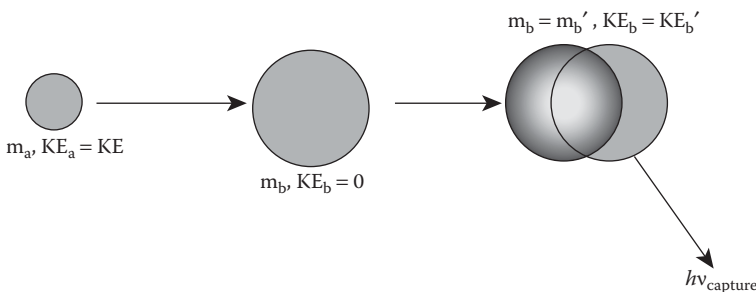
**25.1.4.3 Neutron Absorption**

Neutron absorption is the absorption of the neutron by the nucleus of an atom. Rather than being scattered off, the neutron is actually absorbed by the nucleus. This process may occur at any energy, although it is more common at lower energies after a neutron has been slowed down by inelastic and elastic scattering. Once the neutron has reached thermal neutron energies, it will eventually be absorbed, whether by uranium-235 to initiate another fission reaction, or by shielding materials or other structures of the reactor.

The absorption reaction can also result in a nuclide that is radioactive that will eventually decay itself, releasing more radiation. A common example of this at nuclear power reactors is neutron absorption by cobalt-59, a stable element, to create cobalt-60, which decays by beta decay and also releases two relatively high-energy gamma rays. Figure 25.12 graphically demonstrates this process.



**FIGURE 25.11**  
Neutron inelastic scattering.



**FIGURE 25.12**  
Neutron absorption.

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## 25.2 Regulations

As stated earlier, the purpose of health physics is to control the radiation dose to people, including workers, both the power plant operators and subcontractors, and also members of the public. The governing regulations are Title 10 Part 20 of the Code of Federal Regulations, or 10 CFR 20. In 10 CFR 20, the U.S. Nuclear Regulatory Commission (USNRC) has set radiation dose limits for various categories of individuals. These categories are as follows:

- Occupationally exposed individuals
- General public
- Declared pregnant females
- Embryos/fetuses

### 25.2.1 Occupationally Exposed Individuals

Occupationally exposed individuals are commonly known as “rad workers” and the training courses one takes to become a rad worker are usually referred to as “rad worker I” and “rad worker II” depending on the level of training provided. To be considered an occupationally exposed individual, a worker’s job duties must involve exposure to radiation or a radioactive material. They must also be trained as required by 10 CFR 19.12 on how to properly handle radioactive material or work in areas where radiation exposure is possible and the proper precautions and procedures to follow. Because risks are involved, it is necessary that those exposed be knowledgeable about those risks and, more importantly, know how to minimize them to perform their jobs in a safe and effective manner. Dose limits for occupational workers apply for all occupational dose received in a given year, even if it was from multiple employers.

### 25.2.2 General Public

The general public is everyone that is not a “rad worker.” This includes not only members of the public in the normal sense of the term but also anyone employed or working at the site that is not specifically trained as a rad worker. Commonly, this includes members of the administrative staff and personnel who work exclusively on the secondary side of the plant.

### 25.2.3 Declared-Pregnant Females and Embryos/Fetuses

The dose limits for embryos and fetuses are lower than those for occupationally exposed individuals. The licensee or employer may have to take additional action to ensure that those dose limits are not exceeded. However, as a matter of law, an employer can only take those actions after the pregnancy has been declared. A “declared-pregnant female” is “a woman who has voluntarily informed her employer, in writing of her pregnancy, and the estimated date of conception.” Simple visual indication or verbal notification that a woman is pregnant is insufficient.

### 25.2.4 Dose Limits

There are two sources of dose that must be considered when calculating a person's dose: external dose and internal dose. External dose is commonly measured with a dosimeter worn on the torso. Section 25.3.1 provides more information regarding calculating external dose. External dose is received for discrete intervals of time, such as when a person enters and exits a radiation area. When the person exits the radiation area, they are no longer receiving any external dose. Three different types of external dose are typically measured: deep dose equivalent (DDE), shallow dose equivalent (SDE), and the dose equivalent to the lens of the eye (LDE). The SDE and the LDE are measured or calculated specifically to track the dose to the skin and the lens of the eye. For all other body parts, the DDE is used.

In contrast, internal dose is caused by radioactive materials that have been taken into the body, typically through inhalation, ingestion, or dermal absorption. In this case, the radioactive material will continue to contribute radiation dose to the person until it fully decays away or is eliminated from the body by physiological processes. The dose may be delivered for days, months, or even years. Because dose limits are specified as allowable annual doses, the concept of the committed effective dose equivalent (CEDE) is used when calculating internal doses. The CEDE is the dose that will be received over the next 50 years from an intake of radioactive material. This CEDE is the dose that must be kept below the annual dose limit. Similarly, for calculating the dose to individual organs, the committed dose equivalent (CDE) is used, which is also the dose that will be received over the next 50 years. The dose for the next 50 years is assigned in the year of intake as a conservative estimate and to ease the record-keeping requirements of calculating the dose in a given year from intakes that occurred in earlier years, perhaps when an individual was employed by someone else.

The external and internal doses are then summed to evaluate compliance with the radiation dose limits. Table 25.1 lists these limits. For the general public, there is an additional dose limit for unrestricted areas that requires that the dose received in any 1 h must be  $<0.002$  rem. This limit is not a dose "rate" but a limit. The 0.002 rem could be reached in 1 min, a "rate" of 0.120 rem/h, but if no other dose is received in the remainder of the hour, then the limit has not been exceeded. This requirement is worded

**TABLE 25.1**

Dose Limits

Radiation Dose Limits	Rem Per Year
<i>Occupational Limits</i>	
Whole body TEDE (DDE + CEDE)	5
Organ dose, Deep dose + CDE	50
Eye dose, CDE + LDE	15
Shallow dose, SDE + CDE	50
<i>General Public</i>	
Whole body TEDE	0.1
Unrestricted area dose	0.002 rem in any 1 h
<i>Embryo/Fetus</i>	
Total dose	0.5

this way to permit intermittent events with high dose rates but short durations to occur without exceeding the limit. For embryos/fetuses, not only should the total dose be kept under the limit, but the monthly exposure rate should be kept uniform. In other words, the dose should not be received all at once.

### 25.2.5 Planned Special Exposures

In exceptional situations, doses above those listed in Table 25.1 for occupational workers may be allowed. These situations are referred to as “planned special exposures.” For the latter to be authorized, alternatives that would avoid the exposure must be unavailable or impractical; the individuals who will be exposed must be informed of the purpose, their estimated dose, any special conditions that might be involved, and measures that can be taken to minimize doses. The authorization of the planned special exposure must also be in writing.

The dose limits for a planned special exposure may be up to five times the normal occupational dose limits. However, the dose from planned special exposures is cumulative. An individual cannot exceed these higher limits during their lifetime. Therefore, dose incurred during a previous planned special exposure reduces the allowable dose for any subsequent planned special exposure that individual may participate in. In addition, any dose received in excess of the dose limits in any year, even if not part of a planned special exposure, must also be subtracted from the allowed dose. Once the planned special exposure is complete, both the individuals involved and the U.S. NRC must be informed of the dose received within 30 days.

As a practical matter, planned special exposures are rarely used. The reporting requirements and negative public perception of any event that exceeds the “normal” dose limits dissuade licensees from considering using planned special exposures. In addition, most tasks can be accomplished without resorting to the use of planned special exposures. Between the use of robotics, remote manipulators, and the use of multiple personnel each working for a short duration, planned special exposures are not needed.

### 25.2.6 As Low As Reasonably Achievable (ALARA)

One of the most important aspects of radiation protection requirements, and one that distinguishes them from other personnel safety requirements required by Occupational Safety and Health Administration (OSHA), is the concept of ALARA; 10 CFR 20.1101 requires that “the licensee shall use, to the extent practicable, procedures, and engineering controls based on sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA.” This is in sharp contrast to most OSHA standards where it is acceptable to allow exposures, such as breathing air concentrations of contaminants, to go right up to the limit as long as that limit is not exceeded.

Implementation of ALARA is quite subjective due to the differences in possible interpretation of what is “reasonable” or “achievable.” When applying the ALARA philosophy, one should take into account:

- The state of available technology: What can we do?
- Economic costs versus benefits: Is it worth doing?
- Socioeconomic/societal considerations: What are the down sides in terms of expense and impact on the public or large populations?

The cost–benefit comparison is a consideration that each radioactive material licensee and user must make for themselves. There is a large gray area in the middle where the benefit of a reduction in dose to the individual(s) must be weighed against the cost to the enterprise to implement it. When it comes to reducing external doses in accordance with ALARA principles, there are three major factors: time, distance, and shielding. For reducing internal doses in accordance with ALARA principles, simple contamination control measures should be used.

### **25.2.7 Time, Distance, and Shielding**

To limit external dose below regulatory limits and in accordance with ALARA principles, three factors are used: time, distance, and shielding. These concepts are used separately and in concert to limit radiation doses.

#### **25.2.7.1 Time**

External dose is the product of the radiation dose rate in a given area and the length of time spent in that area. Therefore, minimizing the time spent in the area directly reduces the total dose received. Jobs should be planned before entering a radiation area such that they can be conducted in the most time-efficient manner possible.

This is also one way in which planned special exposures are avoided. If the radiation dose rate in an area is extremely high, a worker may be given a single task, for instance, turn one valve or loosen one bolt, to perform in that area such that their stay time in the radiation area is very short, perhaps as little as a few seconds.

#### **25.2.7.2 Distance**

The further away from a radiation source one gets, the lower the radiation dose rate. Staging materials or conducting operations further away from the radiation source will lower the total dose. This concept can be implemented with administrative measures and engineered devices. For very high radiation areas, one can maximize the use of remote manipulators or robots to keep the operator at a distance from the source of the radiation. In more common situations, keeping materials at arm's length or simply taking a few steps back from the radiation source will reduce the total dose received.

#### **25.2.7.3 Shielding**

By placing an appropriate shield between the radioactive source and the worker, radiation is attenuated and exposure may be completely eliminated or reduced to an acceptable level. The type and amount of shielding needed will vary with the type and quantity of radioactive material used. The shielding may be permanent (e.g., built-in concrete walls), temporary (e.g., lead blankets or movable barriers), or part of the worker's personal protective equipment (PPE), such as a lead apron or plexiglass face shield. More information regarding radiation shielding is discussed in Section 25.4.1.

#### **25.2.7.4 ALARA Internal Dose Precautions**

For materials where internal exposure is a possibility, steps must be taken to reduce or eliminate that dose pathway. To limit internal dose below regulatory limits and in accordance

with ALARA principles, simple procedures and controls are typically implemented, the key of which is to prevent or minimize any intakes of radioactive material. To this end, eating, drinking, chewing gum, smoking, or storing food where radioactive material is used or stored is prohibited. To reduce the possibility of absorption of radioactive materials or contamination on your skin, proper PPE is used. Commonly used PPE consists of coveralls, gloves, and booties. To reduce the possibility of inhalation, a respirator or other sort of respiratory protection can be used. More information regarding PPE and respiratory protection is discussed in Section 25.4.2.

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### 25.3 Radiation Dosimetry

Radiation dosimetry is the measurement of a person's exposure, usually on an individual basis. USNRC licensees are required to individually monitor the occupational radiation exposure to any individual likely to exceed 10% of the dose limits from external or internal dose considered separately. Although, typically most licensees perform such individual monitoring for anyone classified as a rad worker regardless of the dose they are expected to receive. The discussions of external and internal dosimetry that follow are focused on the evaluation of doses to individuals.

For members of the public, including licensee employees who are not rad workers, generally no individual monitoring is performed. Compliance with dose limits is determined by conservative generic calculations. Area surveys are performed or area dosimeters are used to determine the external dose rates in locations accessible by members of the public. These dose rates are multiplied by conservative estimates of the length of time a member of the public may be in that area to determine the total potential annual external dose. For internal dose, estimates of the magnitude of intakes of material that might occur from atmospheric or effluent releases of radioactive material are performed to calculate hypothetical conservative internal dose estimates. The summed internal and external doses are used to demonstrate compliance with dose limits for members of the public.

#### 25.3.1 External Dosimetry

An individual's external dose is commonly directly measured with personal dosimetry. There are several devices that can be used to accomplish this, and they are generically divided into two categories: those that document the dose of record and direct-reading dosimeters that are used for real-time or near real-time measurement of radiation doses being received.

To determine dose of record, three types dosimeters are commonly used: film badges, thermoluminescent dosimeter (TLD) badges, and optically stimulated luminescence (OSL) badges. Film badges were the first type to be developed and provide a permanent record in the form of an exposed film that can be archived. However, their sensitivity and accuracy in the varied radiation fields at nuclear power plants are less than desirable, and they are seldom used in the nuclear industry although they are still common in medical settings. TLD badges were then developed with improved sensitivity compared with film badges, but the drawback was that once the dose received by the badge is measured it could not be remeasured because the measurement process resets the dosimeter. OSL dosimeters are a relatively recent innovation that improve on the sensitivity compared with TLD dosimeters and also permit measurements to be repeated for confirmation. Figures 25.13 to 25.15



**FIGURE 25.13**  
OSL badge. (Courtesy of Landauer, Inc.)



**FIGURE 25.14**  
Film badge. (Courtesy of MP Dosimetry.)





**FIGURE 25.15**  
TLD badge. (Courtesy of National Discount x-Ray Supply.)

show common appearances for dosimeters. Some facilities may incorporate the dosimeter into the site security or access badge.

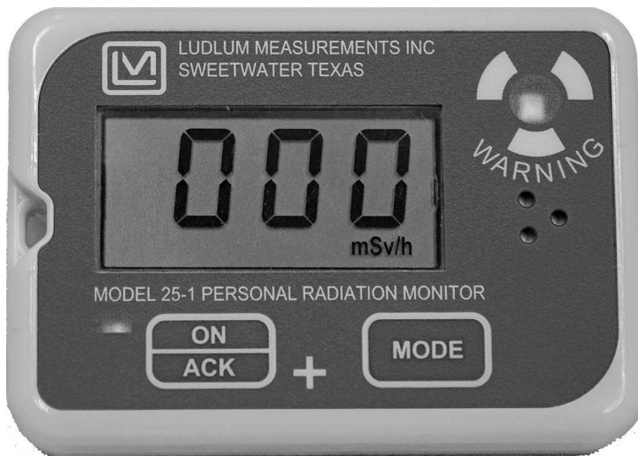
Currently, TLD and OSL dosimeters are the most commonly used dosimeters at nuclear power plants. These badges are exchanged and read on frequencies determined by the anticipated dose received by the wearer. For most workers not expected to receive doses near the limits, this frequency is quarterly or annually. For workers performing tasks in high radiation areas or expected to receive more significant doses, the frequency may be shortened to monthly. If information regarding a worker's dose is needed more often than that, direct-reading dosimeters are used in addition to provide the necessary real-time information.

Pocket dosimeters or electronic personal dosimeters (EPDs) are used to track an individual's dose on a per-day or per-task basis. The pocket dosimeter is about the size of a large fountain pen and is typically worn in a shirt pocket just as a pen would be and many are intentionally styled to appear similar, as shown in an example in Figure 25.16. Pocket dosimeters were developed in the 1920s and consist of an electrically charged quartz or carbon fiber inside a sealed tube that gradually discharges when exposed to radiation. Measurement of the change in the deflection of the fiber by the electric charge permits determination of radiation exposure and can be read by holding the dosimeter up to a light source looking down the tube. The use of pocket dosimeters has largely been superseded by the use of EPDs.

EPDs have supplanted pocket dosimeters for several reasons. They are not prone to measurement error due to rough handling, can be programmed to provide multiple functions, and are amenable to automation of record-keeping. Modern EPDs vary from the size of a pack of cigarettes (Figure 25.17) to devices no larger than a credit card (Figure 25.18). In addition to tracking total dose received, some models also provide a read-out of instantaneous dose rate and can alarm on total dose received or dose rate. They can also be inserted into charging stations that download the data from the EPD for automated data entry into dosimetry databases.



**FIGURE 25.16**  
Pocket dosimeter. (Courtesy of Paul Frame.)



**FIGURE 25.17**  
Thermo EPD. (Courtesy of Thermo Scientific, Inc.)

### 25.3.2 Internal Dosimetry

Unlike external dosimetry, an individual's internal radiation dose cannot be directly measured or monitored. Internal dose must be inferred from measurements of radioactive material concentration in excreta, in the body, or in the air a person is breathing. Measurement of radioactive material in the body or in excreta is referred to as "bioassay measurement." The purpose of bioassay measurements or air sampling is to determine what quantity of material has been taken into the body. Direct bioassay measurement of an individual is preferable to using air concentrations. The most common routes are by inhalation or ingestion, although dermal absorption is possible with some materials such as tritium and material may also be taken in through wounds.

The results of bioassay measurements or air sampling are used to calculate the magnitude of an intake, and the route of intake is determined from knowledge of the exposure scenario or can sometimes be inferred by the results of multiple bioassay measurements.



**FIGURE 25.18**  
Arrowtech EPD. (Courtesy of ArrowTech, Inc.)

Once the magnitude and the route of the intake have been determined, the internal dose, the CEDE and CDE for organs of concern can be calculated.

At nuclear power plants, the most likely radionuclides are

- Mn-54
- Fe-59
- Co-58, 60
- Zr/Nb-95
- Ru-103/Rh-103m
- Ru/Rh106
- I-131, 133
- Cs-134, 137
- Ba/La-140
- Ce/Pr-144

As with external dosimetry, the frequency of bioassay measurement is dictated by the expected dose the person may receive and the risk of nonroutine intakes. Physical and physiological factors also must be factored in. Isotopes with short radioactive half-lives or that are cleared from the body rapidly require more frequent bioassay measurement than for materials that remain in the body for longer periods of time.

Rad workers are normally given baseline bioassay measurements when they are initially hired to establish a baseline and determine if the individual already has a body burden of radioactive material from previous employment or from natural sources. These baseline measurements normally consist of a urinalysis measurement and sometimes a whole body count. If any material is present, this baseline measurement prevents subsequent measurements from being attributed to new intakes when it may be from a prior intake and thus should not be considered as part of the individual's dose.

Bioassay measurements are subsequently taken at periodic intervals and at termination to detect if unsuspected intakes have occurred. If an intake is suspected, additional special monitoring should be used. An intake may be suspected if the individual has entered an airborne radioactive material area, has contamination on the face or nose, other skin contamination, or other unusual event.

### 25.3.3 Bioassay Measurement

There are four main types of bioassay measurements: urinalysis, fecal analysis, whole body counting, and lung or chest counting. The first two constitute *in vitro* analyses, and the second two are *in vivo* analyses.

Urinalysis is perhaps the most common form of bioassay measurement. It is routinely performed as part of periodic bioassay program even when intakes are not suspected due to the relative cost and quickness of this type of bioassay. Fecal analyses are generally only performed when an intake is suspected because the excretion of radioactive material in the feces is highest in the first few days immediately following the incident. After that, urinary excretion is more common and more likely to detect an intake.

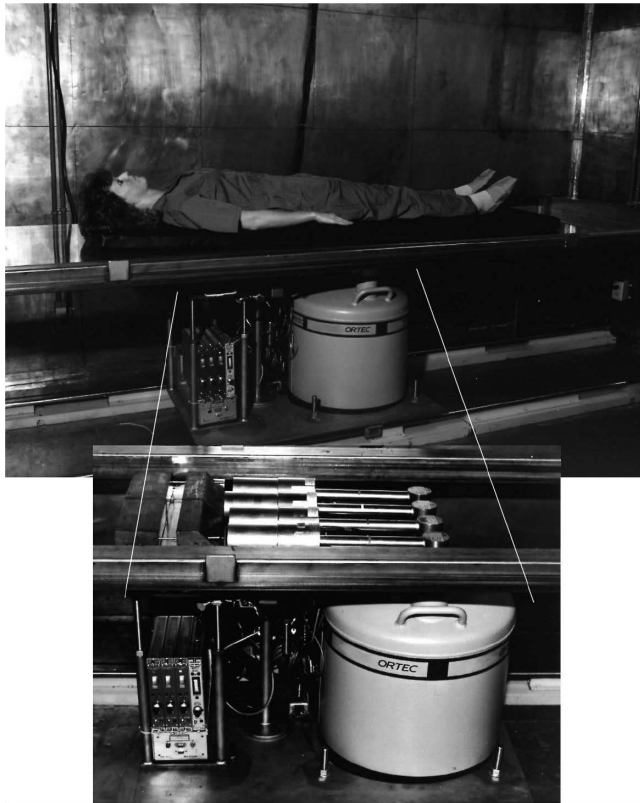
Samples for both types of *in vitro* bioassays are best taken when the risk of sample contamination of the sample is lowest. This means that the samples should be collected at the individual's home or at the beginning of the work shift before the individual has entered areas containing radioactive materials.

Urine and fecal samples can be analyzed for essentially any type of radioactive material. Depending on the degree of precision needed and the radioactive material of concern, analyses for gross gamma, gross beta, gross alpha, or radionuclide-specific analyses may be performed. For periodic bioassay measurements, gross analyses may be performed with a portion of the sample and additional analyses performed later if positive results are obtained from the first analyses.

Whole body and lung counting is performed to detect radioactive materials still in an individual's body that emit gamma rays; pure alpha and beta emitting radionuclides cannot be detected external to the body. A lung count exclusively measures radioactive material that is in the lungs, whereas a whole body count measures radioactive material anywhere in the body, including the lungs. Whole body counting is more common than lung counting, but many counting systems are configured to do both as necessary. Figure 25.19 depicts a whole body counting system that utilizes a sliding bed that passes over a detector to scan the entire body.

### 25.3.4 Intake Calculation

Once a bioassay measurement has been taken and determined to be a true positive result, that result can be used to calculate the associated intake of radioactive material. Additional information beyond the bioassay data is needed to complete this calculation. The most important factors are the solubility type of the radioactive material, the date of the intake,



**FIGURE 25.19**  
Whole body counter. (Courtesy of Ortec.)

and the intake route. The intake route is presumed to be inhalation because this is most common unless it is known to be some other route.

The date of the intake may be definitely known if an incident occurred that resulted in the bioassay measurement(s) being made. Otherwise, a review of records and an interview of the individual should be performed. A review of radiation work permits may indicate when an individual had the potential for an intake of the material of concern even if no intake was suspected at the time, or an interview of the individual may reveal a particular occurrence that could have resulted in the intake. If multiple bioassay measurements are made over a period of time, it may be possible to fit a plot of the data and back-calculate the date of the intake. If none of these are possible, the default assumption is to assume the intake occurred at the midpoint between the positive measurements and any previous bioassay measurements that would have detected the material. Use of the midpoint strikes a balance between conservatively estimating the intake and overconservatism.

The other piece of information needed is the solubility of the material. Radioactive materials are classified into three different solubility classes. Two different classification systems are used but both have the same purpose and in practice are used interchangeably. The first classification system is based on ICRP 26 (1977) and is also the basis for the tables contained in 10 CFR 20. This system classifies materials into solubility classes of D (day), W (week), or Y (year) according to the rate at which the material is absorbed out of the lungs and into the bloodstream. The second system is based on ICRP 66 (1994)

and uses classifications of F (fast), M (medium), and S (slow) that roughly correspond with the D, W, and Y of the previous system. Not all materials exist in all three solubility types. If the chemical form of the material is known or there are sufficient bioassay data to deduce the solubility type, then the calculations may be performed using the solubility. Otherwise, the most conservative type or the default type from 10 CFR 20 Appendix B should be used.

With knowledge of the bioassay result, intake date, solubility type, and intake route, the total magnitude of the intake can be calculated. This can be performed manually using the data contained in NUREG/CR-4884, Interpretation of Bioassay Measurements (Lessard et al., 1987). This document provides lists of intake retention functions (IRFs) for many different radioactive materials and multiple solubility types. The IRFs are given as the fraction of the initial intake that remains in a given location in the body (systemic organs, lungs, nasal passages, gastrointestinal tract, or total body) or that is excreted (fraction excreted in a 24-hour period in the urine or feces and total excreted by each pathway) as a function of the number of days after the intake. Dividing the bioassay measurement, for instance, total radioactivity in a 24-hour urine sample, by the IRF for 24-hour urine for the correct number of days after the intake yields the total magnitude of the intake, as shown in Equation 25.1. NUREG/CR-4884 is based on the older bioassay models from ICRP 26. Similar IRFs are available based on the newer model from ICRP 66 in inhalation intake retention fractions from current ICRP models published, for example, in the *Health Physics Journal* (Potter, 2002).

$$\frac{\text{Bioassay}_{\text{Day}_X}}{\text{IRF}_{\text{Day}_X}} = \text{Intake} \quad (25.1)$$

There are also computer programs available that perform these analyses and permit the rapid analysis of multiple intakes and many bioassay data points. Two programs used are the Computer-based Internal Dosimetry (CINDY) system, available from Canberra, and the Integrated Modules for Bioassay Analysis (IMBA), available from the United Kingdom Health Protection Agency. These programs also calculate the resulting internal dose both to the whole body and to individual organs.

### 25.3.5 Internal Dose Calculation

Computer programs such as CINDY and IMBA will calculate the dose based on an intake automatically, or it can be done manually using data from several sources; 10 CFR 20 Appendix B Table 1 contains listings of allowable limits on intake (ALIs) from which internal dose can be calculated. The intake of one ALI of a material results in a dose equal to the occupational limit. The most limiting occupational limit is given, which is usually the dose to the whole body, or if the dose to a particular organ is most limiting, that value is given along with the ALI corresponding to the whole body dose limit (CEDE). The intake divided by the whole body ALI multiplied by 5 rem yields the internal dose (Equation 25.2).

$$\frac{\text{Intake}}{\text{ALI}} * 5\text{rem} = \text{Total}_{\text{Dose}} \quad (25.2)$$

Federal Guidance Report No. 11 (Eckerman, Wolbarst, and Richardson, 1988) provides dose conversion factors (DCFs) also based on ICRP 26 that allow one to calculate the whole body

CEDE dose and dose to several organs simply by multiplying the intake by the appropriate DCF for a particular radioactive material and solubility type. Similar DCFs based on ICRP 66 can be obtained from various ICRP publications or via the computer program ICRPDOSE2 (ICRP 2001) available from the ICRP.

### 25.3.6 Use of Air Sampling Data

In lieu of individual bioassay data, data from air sampling can be used; 10 CFR 20 Appendix B Table 1 also contains values called derived air concentrations (DACs). This is the air concentration that would result in a dose equal to the occupational limit if a worker were exposed to it for 2000 hours, or a full work year. Exposure, that is, inhaling with no respiratory protection, such a concentration for 1 h is referred to as a DAC-hour and is often used as a means of tracking a worker's total exposure. A DAC-hour is equivalent to a dose of 2.5 mrem. This method is usually employed only for workers not expected to reach a significant fraction of the dose limit or that are not part of the routine bioassay program. If a worker does reach a significant fraction of the dose limit when calculated in this manner, then they are added to the bioassay program or the bioassay sampling frequency is increased to determine a more accurate estimate of their internal dose.

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## 25.4 Exposure Control

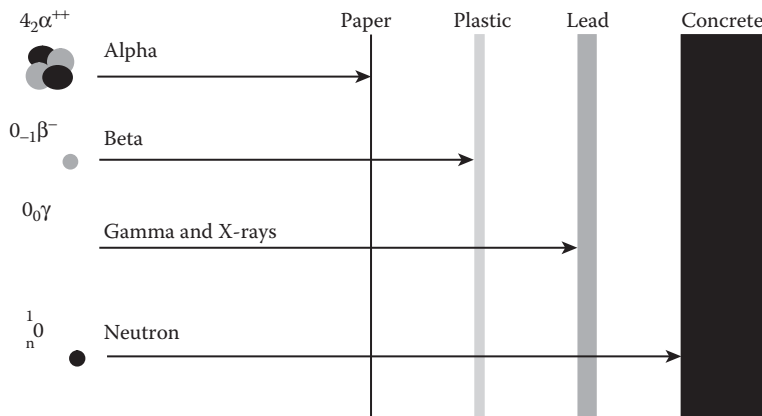
Radiation exposure control can be achieved by engineered controls, administrative controls, or the use of PPE. Engineered controls are the preferred method of control because an engineered control is less likely to fail. Administrative controls can be violated and PPE can be ineffective. Engineered controls are more prevalent for the control of external dose through the use of radiation shielding. This shielding may be permanent and part of the design of the facility or temporary barriers erected for specific tasks. Internal dose uses PPE and administrative controls more than engineered controls.

### 25.4.1 Radiation Shielding

There are four types of radiation, alpha, beta, gamma, and neutron, but of these four, alpha radiation is not a concern from an external dose and radiation shielding perspective. Alpha particles will not penetrate through the outer dead layer of the skin epidermis or even through a piece of paper. Therefore, only the other three radiation types are a shielding concern, and of these three, gamma and neutron radiation are the most difficult to shield. Figure 25.20 depicts the relative penetrability of the various radiation types through materials.

#### 25.4.1.1 Gamma Shielding

When shielding gamma rays, the objective is to stop the gamma ray by having it transfer all of its energy to electrons. Once this energy is transferred to electrons, it can be considered to be deposited at the point of transfer, where the energy eventually manifests itself in the form of heat. The amount of energy being transferred, while significant from a radiation dose perspective, is rarely a matter of concern. The exception to this being spent



**FIGURE 25.20**  
Radiation penetrability.

fuel pools where the total heat input to the shielding material, in that case the water of the pool, can be significant.

Recall from the discussion of gamma ray interactions earlier that there are three types of interactions: photoelectric effect, Compton scattering, and pair production. Of these three, only the photoelectric effect completely “stops” the gamma ray. Compton scattering produces another lower energy gamma ray and pair production produces two lower energy gammas. Therefore, it is desired to maximize the occurrence of the photoelectric effect, which is done by maximizing the density of the shielding material, which increases the number of electrons available for the gamma rays to interact with per unit thickness, and to increase the atomic number of the shielding material. This is why lead, with a density of  $11.3 \text{ g/cm}^3$  and an atomic number of 82, is a commonly used shielding material, especially in areas where space constraints require the use of thin shielding materials. Where space constraints are not an issue, concrete is commonly used. While the concrete itself it does not have a very high atomic number, iron shot can be used as an aggregate to increase the average atomic number. In addition, concrete is relatively cheap, nontoxic, and can be incorporated into the facility design as a load-bearing material, unlike lead.

#### 25.4.1.1.1 Exponential Attenuation

When passing through air, the shielding effect of the air is negligible and is ignored as having less effect than the geometric attenuation effects discussed later. When passing through solid materials, gamma rays will be exponentially attenuated. The radiation field intensity,  $I$ , will be reduced from the initial intensity,  $I_0$ , after passing through a shield of thickness “ $X$ ” in cm as given in Equation 25.3, where  $\mu$  is the linear attenuation coefficient and is a function of the shield material and the energy of the incident gamma rays. Figure 25.5 plots  $\mu$  as a function of energy for lead shielding:

$$I = I_0 e^{-\mu x} \quad (25.3)$$

A useful shield thickness to know is the half-value thickness, that is, the thickness of shielding required to reduce the incident radiation intensity to one half of its previous value. If radiation surveys have been performed to determine the dose rate in a given area, quick estimates of the amount of shielding needed to reduce the dose rate can be made



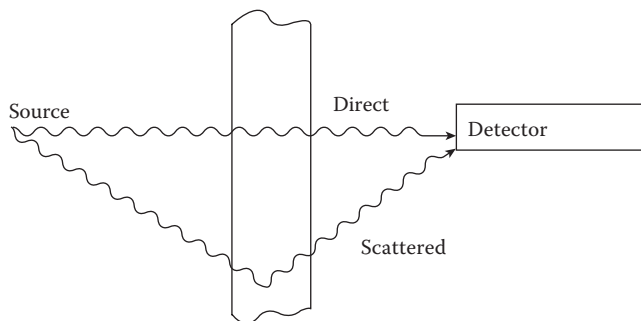
by adding half-value layer thicknesses of shielding until the halved dose rate reaches an acceptable level.

For cobalt-60 gamma rays with an average energy of 1.25 MeV, this thickness is approximately 1 cm of lead or 5.2 cm of concrete. Even though the concrete shield is five times as thick as the lead shield to provide the same amount of reduction, the difference in mass is much smaller. A 1-cm-thick lead shield has a mass of 11.3 g/cm<sup>2</sup> of shielding surface area, whereas the 5.2-cm-thick concrete shield has a mass of 12.0 g/cm<sup>2</sup> of shielding surface area, less than a 10% difference. Cobalt-60 has relatively high-energy gamma rays compared with most gamma-emitted radioactive materials encountered at a power reactor and thus are harder to shield due to the lower linear attenuation coefficient at the higher energies. Once shielding thick enough to sufficiently attenuate the cobalt-60 gammas has been calculated, gamma radiation from anything else will have been sufficiently attenuated as well.

The attenuation calculation above is based on any gamma interaction, including Compton scattering and pair production, as removing the gamma ray from the radiation being transmitted through the shield. The validity of this approximation is reduced as the shielding becomes thicker because gamma radiation deflected at other locations may contribute to the total gamma radiation intensity. This is called “buildup” (Figure 25.21). For more discussion of building, see Cember (1996).

#### 25.4.1.1.2 Geometric Attenuation

The further you move away from a gamma-radiation source, the lower the radiation dose rate. For a point source of radiation, the rate of change of the dose rate is inversely proportional to the square of the distance. Double the distance and the dose rate decreases by a factor of 4. This is the basis for the “distance” portion of the time–distance–shielding ALARA principles. This strict inverse square relationship is only true for point sources. In actuality, most radiation sources are not a point source. Common radiation source geometries are a line, such as a pipe with radioactive contents, or a cylinder, such as a waste container or filter cartridge. When close to such a source, the inverse square relationship is not true. However, as one moves further away, the inverse square relationship becomes a closer approximation. Table 25.2 lists the percent error from assuming an inverse square relationship where a positive value indicates an overestimate of the actual value. As can be seen in the table, if the distance away from the source is less than the length of the object itself, the approximation is quite poor, but if separation distance is three times the source length, the percent error in this approximation is less than 1%, and at five times the distance, the approximation is very good.



**FIGURE 25.21**  
Gamma ray “buildup.”

**TABLE 25.2**

Percentage Error in Assuming Point Source Geometry

Distance as % of Longest Dimension	Sphere	Cylinder	Line
50	41.3	16.1	-22.6
100	4.5	1.8	-7.9
150	1.7	0.6	-4.1
200	-2.1	0.0	-2.1
300	0.0	-0.2	-0.8
400	-0.8	-0.8	-0.8
500	0.0	0.0	0.0

### 25.4.1.1.3 Rules of Thumb

There are two quick calculations or rules of thumb that can be used to calculate what gamma dose rates are likely to be in an area. The first is the 6CEN rule:

$$D = \frac{6CEn}{r^2} \quad (25.4)$$

where:

$D$  = dose rate in rad/h

$C$  = activity in curies

$E$  = average gamma energy in MeV

$n$  = radiation yield

$r$  = distance in feet

When the energy and yield are unknown, a value of 1 can be used as approximations. Note that this rule of thumb is based on the inverse square relationship, but that relationship is reasonably accurate in many situations as discussed above.

The second quick calculation uses the specific gamma ray constant as given in Equation 25.5. Table 25.3 lists the specific gamma ray constant for some common radionuclides.

$$D = \Gamma \frac{A}{r^2} \quad (25.5)$$

**TABLE 25.3**

Specific Gamma Ray Constants for Common Radionuclides

Nuclide	$\Gamma$ (R/h-Ci at 1 m)
Cs-134	0.999
Cs-137	0.319
Co-60	1.370
Fe-59	0.662
Mn-54	0.511
I-131	0.283
Ba/La-140	1.253

where:

$D$  = dose rate in rad/h

$\Gamma$  = radionuclide-specific gamma ray constant

$A$  = activity in curies

$r$  = distance in meters

### 25.4.1.2 Beta Shielding

Shielding of beta rays is relatively easy. Intense beta radiation is normally accompanied by substantial amounts of gamma radiation as well, which is harder to shield. Shielding sufficient to lower the gamma dose rates to acceptable levels automatically lowers the beta dose rates as well. In situations where beta radiation is especially intense, one modification to the shielding design that may be employed is to use a multilayered shield with an inner layer specifically designed to shield beta radiation. The reason for this is that the best material to use to shield gamma rays, lead, is one of the worst to use to shield beta rays.

When betas pass through high-atomic-number materials such as lead, they can emit what is known as “bremsstrahlung” or “braking radiation.” This happens when a beta particle passes near a high-atomic-number nucleus and its direction of travel is bent by the electric attraction between the positively charged nucleus and the negatively charged beta particle, and it gives off an X-ray that much be shielded like a gamma ray. This effect is much more prevalent in high-atomic-number materials and with higher energy beta particles. Therefore, the best materials to use for shielding betas have low atomic numbers, such as plastics, water, and concrete. If lead shielding is being used to shield a mixed beta–gamma radiation source, then a plexiglass or other plastic shield is placed in front of it to stop the beta particles without emitting bremsstrahlung X-rays that would increase the gamma shielding requirements. A centimeter of plastic shielding is usually sufficient to totally stop the beta particles or reduce their energy to the point where bremsstrahlung is no longer a concern.

### 25.4.1.3 Neutron Shielding

Designing shielding for neutrons is much more complicated than shielding for gammas or betas. The objective of a neutron-shielding material is to slow the neutrons down and eventually absorb them. However, neutrons do not attenuate exponentially as do gammas. This is due in main part to the many scattering interactions that occur for neutrons before they are absorbed. In essence, the neutron “rattles” around in the shielding medium until it escapes or is slowed down and absorbed. Once the neutron is absorbed, the absorption reaction often produces a gamma ray that must also be shielded.

Neutrons may be absorbed at any energy, but for most materials the probability of absorption is inversely proportional to the velocity of the neutron and thus is much more likely at lower neutron kinetic energies and thus lower velocities. The easiest way to do this is to use light-atomic-weight materials and especially hydrogen in the shielding material. Neutrons scattering off of hydrogen can lose significant fractions of their kinetic energy in a single collision, up to 100%, whereas neutron scattering off of a uranium nucleus would have a maximum energy loss of less than 1%. On average, a neutron scattering off of hydrogen will lose half of its energy.

The primary objective of a neutron shield is to deflect the neutrons from heading outward, slow the neutrons down as quickly as possible, and absorb them. This is accomplished by the use of a laminate or multilayered shield. The inner layer of the shield is

constructed using relatively high-atomic-weight materials such as iron. This layer will tend to reflect neutrons back inward toward the reactor core or deflect their trajectory from heading directly through the shield.

The purpose of the second layer of the shield is to slow down and absorb the neutrons. Water makes the best neutron shield for this purpose followed by plastics and then concrete due to their relative hydrogen contents. This is the reason spent fuel pools are deep pools and why the reactor vessel is submerged underwater during refueling. The water serves an important role in cooling the fuel rods as well, but the depth of the pools is dictated by the need for shielding.

When a hydrogen atom absorbs a neutron, it gives off a 2.26-MeV gamma ray in an (n- $\gamma$ ) reaction. This high-energy gamma ray requires additional shielding to control the overall external dose rate outside of the shield. The additional shielding required is included in the depth of the spent fuel and refueling pools.

Although exponential attenuation of neutrons through a shielding material is not a physical phenomenon, in the case of laminate shield containing a high proportion of hydrogen in the outer layers, it is a close approximation. For gammas, there is the linear attenuation coefficient that describes how fast gammas are attenuated. For neutrons, this value is called the "macroscopic removal cross section,"  $\Sigma$ . Table 25.4 gives a few example values of  $\Sigma$ . The effective attenuation of the neutron fluence through the shield by the first inner shield layer is exponential. In actuality, the attenuation is not exponential, but the neutron scattering reactions in the first layer either scatter the neutrons back into the core where they are of no concern or scatter them through sharp angles such that the following layer(s) of hydrogenous materials can finish slowing and absorbing the neutrons before they make it through the shield.

Calculation of neutron shielding is best handled by computer codes. The calculations required are too involved to permit effective computation by hand except for very simple geometries and situations.

#### 25.4.1.4 Shielding Computer Codes

There are many computer codes available to assist with radiation shielding design. They fall into two generic categories: deterministic and probabilistic. The deterministic codes replicate what a person could calculate by hand but with increased speed and the ability to handle additional complexity. A common example of this sort of code is Microshield™ (Grove, 2005). Microshield calculates gamma shielding for simple geometries (slabs, cylinders, spheres) and one or more layers of shielding taking into account shielding attenuation, buildup, and geometric attenuation. Although limited to simple geometries, many real-world situations can be modeled as one of these simple geometries reasonably well. The codes are relatively easy to use and give quick results.

**TABLE 25.4**  
Example Macroscopic Removal Cross Sections

Shielding Material	Macroscopic Removal Cross Section (cm <sup>-1</sup> )
Water	0.103
Iron	0.168
Concrete	0.089
Uranium	0.174

The second type of codes, probabilistic codes, can also be referred to as Monte Carlo codes. One of the most widely used Monte Carlo codes is MCNP (LANL, 2003). Fundamentally, radiation interactions with matter are probabilistic events that are described by statistical quantities such as linear attenuation coefficients. Deterministic codes use the statistical quantity, or the average behavior, to perform their calculations. Monte Carlo codes replicate the underlying probability distributions and “roll the dice” using random number generators to determine how far a particle goes before interaction, what type of interaction it has, and what angle it scatters through. These codes are extremely powerful and can model any geometric configuration and any combination of radiation fields. The downside of these codes is that they are cumbersome to use, require laborious input, can be slow to yield answers, and require expert users to ensure valid results.

#### 25.4.2 PPE

PPE has two primary purposes: to prevent internal dose and to prevent the spread of contamination. It also can prevent or lower the dose to the skin. Occasionally, PPE is designed to reduce external dose such as lead aprons worn in hospitals, but this is not a common practice. In power reactor settings, radiation fields tend to be more isotropic, that is, coming from all around, rather than from a discrete point source, reducing the effectiveness of an apron work on only one side of the body. Aprons, skirts, or jackets that provide isotropic protection are quite heavy and not practical.

The PPE required is specified in a graded approach consistent with the hazards in a particular area and the tasks to be performed. A radiation work permit (RWP) is prepared by the facility health physics staff that specify what PPE is needed for particular areas and tasks. The PPE needed to perform a walk-through inspection of an area with no hands-on work being performed can be dramatically different from that required for performing cutting, grinding, and welding. When entering an area where PPE is required, an entrance/exit portal is set up where the PPE may be donned prior to entering the area and doffed on exit. The portal also has survey instrumentation to permit the individual to check for contamination after removing the PPE and prior to exiting the area.

To prevent or reduce internal dose due to inhalation of radioactive material, respiratory protection is worn (Figure 25.22). The type of protection worn is dependent on the amount of potential airborne radioactive material. Any type of respiratory protection places some degree of burden on the wearer from the weight of the device, reduced mobility, increased difficulty breathing, reduced field of view, and thermal stress. These burdens can negatively influence the length of time it takes to complete a task, increasing the person's external dose received during the job. Therefore, the objective is to optimize the respiratory (and other PPE) provided to minimize the total dose received. The types of respiratory protection that can be used, in order of increasing effectiveness and increasing burden, are half-face respirators, full-face respirators, self-contained breathing apparatuses, supplied-air respirators, and fully encapsulating supplied air suits (Figure 25.23).

Respiratory protection devices are rated as to the effectiveness of the protection provided. A typical protection factor for a full-face air purifying respirator, those with filter cartridges and no separate air supply is a factor of 100, implying that the contaminant concentration inside the respirator is no more than 1/100th of the concentration in the ambient air; 10 CFR 20 Appendix A contains a table of respiratory protection factors for different types of respiratory protection. As the degree of inhalation risk increases, the risk of skin contamination also increases. Therefore, half-face respirators, those without a covering for the eyes, are not commonly worn. In addition, some form of anti-contamination



**FIGURE 25.22**  
Air-purifying full-face respirator. (Courtesy of FRHam.)



**FIGURE 25.23**  
Fully-encapsulating suit. (Courtesy of FRHam.)

clothing (anti-C) is used in conjunction with the respirator to prevent skin contamination. In extreme cases, a fully encapsulating suit with either supplied air or an integral air filtration unit is worn.

The primary purpose of anti-C clothing is to prevent skin contamination and the spread of contamination. In most instances, the use of anti-C clothing precedes the use of respirators. It is more common for areas to be contamination areas, indicating that there are surfaces and structures in the area with removable contamination, than it is for an area to be an airborne radioactive material area. Any airborne radioactive material area is automatically a contamination area because it is not possible to have a high concentration of airborne radioactive material without having removable contamination on the surfaces and structures in that area.

The amount of anti-C clothing worn is specified by the RWP for a particular task and area. Walk-through inspections of an area may only require the wearing of gloves and shoe covers with or without company-issue coveralls. Light hands-on activities adds the wearing of anti-C coveralls over company-issue clothing and two layers of gloves, inner surgeons gloves, and outer gloves suitable for the task being performed. Joints between articles of clothing, especially between the gloves and the coverall sleeve and between the shoe cover/bootie and the coverall leg, are sealed with tape. A hood may or may not also be required. If light dust-generating activities are being performed, a minimum of a full-face respirator is also worn and the respirator is sealed to the hood with tape as well. Multiple layers of anti-C clothing are also sometimes used, although heat stress for the wearer becomes a concern at this point and must be balanced against the extra protection provided. In the most extreme conditions, a fully encapsulating suit is worn with or without additional clothing for abrasion resistance or additional protection.

### 25.4.3 Engineered Controls

As a best practice, engineered controls should be used in lieu of administrative controls or PPE. Engineered controls can be implemented to control both external and internal dose. Obvious engineered controls are shielding barriers built into the design of a facility.

The spent fuel pool itself and the transfer canal between the pool and the reactor refueling pool is one such obvious engineered control. Others are features such as placing filter media and ion-exchange media that accumulate moderate quantities of radioactive material in below-grade housings or within thick concrete shielding to reduce radiation dose rates. Valve banks can be equipped with handle extensions that penetrate through shielding walls to permit manual operation of the valves from a safer location. Such design features obviously require forethought before the facility is constructed.

In some areas, it is not possible to install permanent shielding and still permit access for inspection and maintenance. In these cases, portable shielding can be set up to provide a work area with lower radiation dose rates during a maintenance task. In this manner, only the workers who set up the temporary shielding are exposed to the unshielded radiation. If the temporary shielding can be set up quicker than the following task, the overall, collective, radiation dose received by all the workers is reduced. This sort of shielding can include lead blankets hung on frames or draped over equipment or plastic barriers filled with water that can also provide neutron shielding.

With regard to controlling internal dose, engineering controls can also be used. An additional benefit of using engineered controls is that they do not restrict the mobility of the worker, enabling tasks to be completed quicker, which also lowers the total dose received. Local ventilation is one common means. Providing increased ventilation or air filtration in

the vicinity of a job may allow the task to be performed without wearing respirators. The ventilation could be general area ventilation or filtration or task-specific ventilation such as vent hoods, or local suction to capture welding, cutting, grinding, or similar fumes.

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# 26

## *Health Effects of Low Level Radiation*

Jay F. Kunze

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### 26.1 Summary

Contrary to conventional understanding of radiation by the public, low levels of nuclear (or ionizing) radiation are not harmful to animals or humans but are actually beneficial to health. This chapter cites some of the vast amount of epidemiological and demographic studies that support this characteristic of low-level radiation and its positive effects on humans. Public understanding of this hitherto seldom recognized characteristic of radiation is crucial to the public having a more rational attitude toward nuclear energy utilization. The following is a brief summary of some of the most significant studies that show the hormetic effect of low levels of radiation, with the optimum for good health being nominally 50 times the average background radiation on earth.

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## 26.2 Historical Background

Nuclear radiation, emanating from interactions with the nucleus of the atom, generally represents the release of energy in the range of 1 million electron volts (MeV) to as high as 200 MeV, as is released in fission of uranium or plutonium nuclei. This compares with about 4 eV released in the combustion of one atom of hydrogen or carbon. Nuclear radiation was first discovered by Becquerel in 1896, and further investigated by Marie and Pierre Curie, who isolated and identified radium as one of these natural sources of nuclear radiation. The nucleus, extremely small in apparent size (compared to the size of the atom, defined as the outer "orbit" of its electrons), was not discovered until 1913, by Rutherford. In 1932, Chadwick identified the neutron as a basic particle included in the nucleus. In 1938, Hahn, Straussman, and Meitner split (fissioned) the U-235 nucleus using neutrons, identifying an enormous release of energy per fission.

The applications were initially for fashionable uses, such as luminescent dials on watches, the results of carefully painting numbers on the dials and on the hands of the watches, using radiant paint containing radium. It was tedious work, generally done with fine brushes by women, who would shape the point on the brush with their lips and tongues. Most developed mouth or tongue cancer. It is also believed that Marie Curie died of cancer initiated (and stimulated) by large amounts of radiation that she may have absorbed in processing many tons of uranium ore on the way to isolating radium. During these early periods of novel uses of nuclear radiation, methods of measuring doses had not been developed, and regulating doses for personnel or the public had not been adopted.

With the initiation of the Manhattan project to develop a nuclear weapon during World War II, the involved scientists developed methods of both estimating and measuring ionizing radiation, which is any radiation capable of displacing electrons in molecules, so as to modify the chemical composition, or even to destroy the molecule completely. Such modifications generally required energies greater than a few electron volts deposited per atom, which are small compared to the alpha, beta, and gamma ray energies emitted from excited nuclei. Hence, any of those nuclear radiations could likely damage many DNA molecules, and hence destroy or mutate many cells in the body.

### 26.2.1 Quantifying Radiation and the Amounts That Would Appear to Cause Cancer or Death

During the latter days of the Manhattan project, two criticality accidents occurred (1945 and 1946), resulting in the deaths of one scientist in each accident at Los Alamos National Laboratory, plus exposures of other in the laboratory to large amounts of radiation. These early experiences, and the radiation deaths of several other nuclear workers in U.S. Atomic Energy Laboratories in the next 15 years, provided estimates of acute radiation doses that would result in probable deaths. From these incidents, plus data from other countries, and experiments on animals (specifically dogs), regulations for safe amounts of radiation were developed. In the early 1950s, scientists from around the world came to an agreement that 5 rad (or REM)\* was a safe annual dose for "radiation workers." A one-time dose up to

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\* A RAD, the unit used in the 1950 to 1980 period, is 100 erg of deposited ionization energy per gram of tissue. The REM is the RAD Equivalent Man, allowing for multiplication factors to be applied to especially damaging radiation, neutrons, and alpha particles. The SI accepted units now are the gray and sievert, respectively. The gray is equal to 1 J of ionizing radiation deposited per kg of tissue. 1 Gy = 100 rad, and similarly 1 Sv = 100 REM.

10 rad (0.1 Gy) was considered tolerable and not likely to affect a person's longevity. Doses to the public were generally recommended to be the order of at least ten times smaller than those to a radiation worker, which implied that radiation workers had a very small risk of dying from exposure to radiation in their working environment, and the risks to the public were considered negligible.

### **26.2.2 The Advent of the Linear No-Threshold Hypothesis**

When the discipline of "health physics" developed in the 1950 decade, these rules were formalized into a statistical theory, known as the "Linear No-Threshold" (LNT) hypothesis. This theory stated that the long-term health effects of radiation on the human body were primarily statistical in nature, determined by the number of radiation particles (for instance, gamma ray photons) to which a person was exposed, and that any gamma ray penetrating the body could initiate conditions that sometime in the future would become a cancer. Because it was assumed that any gamma ray (or beta particle, or other ionizing radiation) could be detrimental to human life, it was concluded that there was no lower threshold of radiation that would be completely safe. This LNT hypothesis has been used for the last six decades to develop regulations for exposures of workers and the public, and to estimate the overall risks of the number of the public who would be expected to die (from cancer) should there be a release of radiation (intentional or accidental) to the environment. This theory (and its corollary, the Collective Dose Hypothesis—see Section 26.1.10) and the resulting method of regulating doses and calculating risks continues to be used throughout the world up to the present (several exception nations to be noted later).

### **26.2.3 Subsequent More Strict Regulations a Consequence of the LNT**

In 1978, the Nuclear Regulatory Commission (NRC) in the United States went a step further in developing regulations based on the LNT, and adopted a rule for all licensees that personnel doses should be kept as low as reasonably achievable (ALARA). Failure to implement an ALARA program effectively could result in violations and fines to a company having an NRC license. Consequently, workers in the nuclear industry have developed a deeply entrenched fear of radiation, because all have been well schooled in the LNT hypothesis.

### **26.2.4 The Appearance of Conflicting Data**

Shortly after the ALARA concept was mandated, demographic data began to appear indicating that the LNT may not always apply. One such source of data was from UNSCEAR (United Nations Scientific Committee on the Effects of Atomic Radiation), which had been tracking the survivors of the two nuclear bombs dropped on the Japanese cities of Hiroshima and Nagasaki in 1945. An anomaly appeared in the data for personnel doses from the nuclear explosions in the range of 0.1 Sv (10 REM), showing that persons receiving doses in that range not only had no increased risks of developing leukemia but actually had less risks than the average Japanese citizen exposed only to typical background radiation [1].

As a result of this apparent anomaly in the Japanese data, some other demographic studies were undertaken. The Society for Nuclear Medicine in the United States examined the health of the many U.S. military personnel who had been detailed to gather data where nuclear bombs tests had been conducted, both in Nevada and in the Pacific Islands. That study showed a definite trend that those exposed to radiation were healthier and were contracting less cancer than the average soldier.

In the early 1990s, the National Institutes for Health and the U.S. Department of Energy [2] undertook a study of nuclear navy civilian and military personnel who had received radiation in their work environment (all well below the upper limits allowed by the existing regulations) and found that those receiving radiation were healthier than their counterparts who did not receive radiation (for instance, office workers). Similar studies of radiation workers for Ontario Hydro (which operates most of the Canadian Nuclear Power plants) [3] and for the nuclear workers in the United Kingdom [4] showed similar results. In all of these studies, there was a clear trend of longer and healthier lives for those workers exposed to low levels of radiation compared to the workers not exposed to radiation.

### 26.2.5 The Plutonium-239 Concern

Because most of the nuclear weapons made in the United States used Pu-239 as the fissile material, personnel protection from the inhalation of this material became a high priority for the health physics profession. Plutonium tends to exhibit physiological behavior similar to calcium (though the chemical valences are not the same), and appeared to seek the bones. Consequently, bone cancer could develop from the ingestion of a few milligrams of Pu-239, with its 24,000 year half-life and its emission of very damaging alpha particles. Consequently, plutonium began to be known as one of "the most dangerous substances on earth."

In 1995, a follow-up study [5] was done on the 26 soldier machinists who did the machining on the plutonium bombs made at Los Alamos in the 1944–1946 period. The inhalation doses of these soldiers averaged 1.25 Sv (125 REM), with the highest being 7.2 Sv, from plutonium. These lifetime dose estimates included the multiplier of 20 for the damaging effects of the alpha particle radiation from the plutonium, a multiplier indicating its "relative biological effectiveness" or the "quality factor." The follow-up study in 1995 attempted to determine how these 26 had died. However, 19 were still alive and healthy (aged 69–76), and of the 7 who had died, only 3 had died of cancer, only 1 of those from lung cancer.

### 26.2.6 The Radon-in-Homes Saga

Radon is the noble gas element in the decay chain of uranium down to lead. Radon forms no chemical compounds, but remains a gas and seeps up through the ground, and finds its way into basements of homes through cracks in the concrete. It can accumulate to presumably dangerous levels over time if ventilation is poor in the basement. The discovery of levels of concern occurred in the early 1980s, in the northwestern part of the United States. The LNT hypothesis was applied to the levels measured, based on death rates from lung cancer for uranium mine workers in Czechoslovakia. Many of those workers developed and died of lung cancer, presumably the result of radon ingested into the lungs, which, while there, decayed into the solid Po-218. This atom would lodge itself in the lungs and decay to bismuth and eventually stable lead-206. The EPA proceeded to develop kits that could be placed in any home for a few weeks, and then be sent to a laboratory for analysis of radon in the home. In 1990, Dr. Bernard Cohen [6], who had urged the EPA to undertake these studies, examined the data obtained from some 1600 counties in the United States and compared that data with lung cancer death rates in those counties. The data fit a reasonably reliable straight line when lung cancer death rates were plotted versus radon levels in the homes. However, the slope of the line was inverted, showing that the highest radon levels had the fewest deaths, the lowest radon levels having the highest deaths.

These radon concentrations in the homes were orders of magnitude less than the levels exposing workers in the uranium mines in Czechoslovakia, and hence the results imply that low doses are not only harmless but beneficial to avoiding lung cancer. Nevertheless, to this day (2015) the EPA continues to ignore the results of their own data and places information spots on the broadcast media about the dangers of radon in homes, and what actions homeowners should take to reduce the radon concentrations.

**26.2.7 The Industry and Regulatory Response to the Hormesis Data**

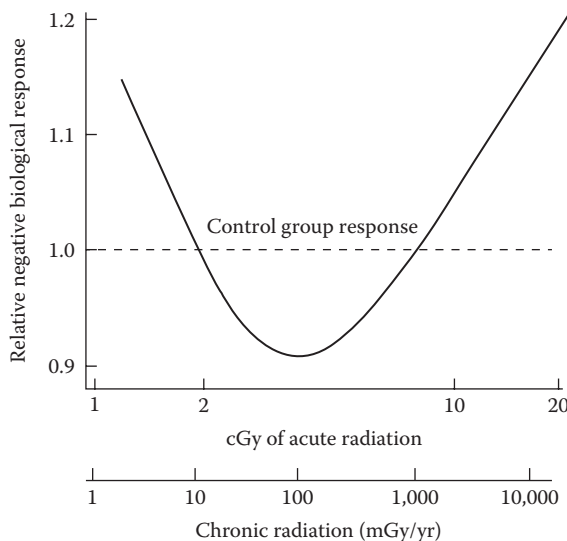
The above examples are just a few of the many studies that have shown that radiation health effects demonstrate a hormetic effect, a characteristic of most pharmaceuticals and even to many foods. Hormesis is the concept that large doses of a substance may be detrimental to health, but small doses are beneficial. The evidence for this knowledge was effectively presented in a text published in 1980, titled *Radiation Hormesis*, by Dr. T.D. Luckey, and in 1991 followed by the second edition [7]. This text included over 1000 references of data on radiation hormesis.

From about 1994 to 2008, a public interest organization known as Radiation Science and Health, headquartered at Worcester College in Massachusetts gathered and disseminated much of the vast amounts of data that then existed [8].

In 2005, the International Dose Response Society was formed at the University of Massachusetts at Amherst and has since been a major organization in an effort to disseminate further data as they become available [9].

Three presidents of the American Nuclear Society (Alan Waltar, Larry Foulke, and Eric Loewen) have made special efforts to promote knowledge and understanding of radiation hormesis, the most recent being a 200-page publication of a special session held at the Chicago Annual Meeting of the ANS in June 2012, and is available online [10].

Figure 26.1, derived from the work of T.D. Luckey, represents the nominal hormetic effect of low levels of radiation on human beings. The ordinate axis indicates nominal negative



**FIGURE 26.1**  
Nominal hormetic effect of low levels of radiation on human beings.

effects of radiation, such as the induction of leukemia or solid cancers, the death rates from cancer, or the overall negative effect on health or longevity as a function of the amounts of radiation shown on the abscissa. The numerical values on the ordinate will likely vary around the 1.0 control group, depending on the type of negative biological effect being presented. However, the abscissa values appear to be nominally the same for all of these effects. The minimum negative effect occurs at approximately 100 mGy (10 rad) per year for chronic radiation exposure (effectively a continuous exposure), and at 3 cGy (3 rad) for acute radiation (received instantaneously or over a short period of time). The chronic radiation minimum value is approximately 50 times the nominal annual background radiation dose. The acute dose minimum (a desirable dose) is approximately 50% higher than that received from a full-body CT scan. Numerous health professionals and publications continue to caution against full-body CT scans, despite the fact that these probably are beneficial to overall health of the individual, irrespective of the outcome of the diagnosis.

However, despite the preponderance of data clearly showing radiation hormesis, few current textbooks used in our universities for nuclear engineering or health physics even mention the hormetic effects of radiation. The National Council on Radiation Protection and the International Council on Radiation Protection, and the Committee on the Biological Effects of Ionizing Radiation (BEIR) refuse to give more than casual comment about hormesis, taking the position that it is best to be "safe" in dealing with low levels of radiation, and keeping the doses as low as possible. The few significant deviations from the antihormesis approaches have been with the radiation councils in France, China, and Japan.

### **26.2.8 Can the Effect Be Tested Using the Methods Similar to Those Used in Certifying Beneficial Effects of a Medical Treatment or Pharmaceutical?**

In medical practice, the efficacy of a new pharmaceutical or new treatment modality is evaluated by what is usually designated as a "double-blind study." One group of patients will receive the treatment, and another similar group will receive a placebo. Neither the patients nor their physicians will know in which group a particular patient was placed. There is a general lack of understanding of the hormesis of radiation by the public, the media, and even many nuclear professionals. Because of this ignorance, it has always seemed most unlikely that a double-blind study of radiation hormesis would be allowed by authorities. However, a nominal double-blind study that was not planned occurred spontaneously in the 1980 decade in Taipei, Taiwan. Several large apartment buildings were built in 1982, using rebar contaminated with Co-60 that apparently came from a scrapped radiation therapy machine. The "contamination" of the concrete in the buildings was not discovered until 1992, which was two half-lives after the initial construction and occupancy of the buildings. In the 2004 and 2006 period, two studies reported on the health effects of radiation on the nearly 10,000 residents in those apartment buildings over a 9–20 year period. Average doses were only about 0.5 cGy though 11% of the cohort received average doses approaching 9 cGy. By analyzing the residents by age groups, Chen et al. [11] expected cancer death rates during that period would have been in the range of 200 among those 10,000 residents. However, less than 10 cancer deaths could be found from death records. This ratio is most significant, even if several standard deviations of uncertainty might be attributed to the accounting of actual cancer deaths.

In a second study, Hwang et al. [12] found virtually no difference in contracted cancers between the exposed population and the average citizens in Taiwan, except for a somewhat higher rate of leukemia among the exposed children. This difference between the two studies is not inconsistent with the hormetic effects observed with some

pharmaceuticals, such as those for treating the common cold, for which the OTC drugs are not recommended for children. Furthermore, the latter study tracked those who developed cancer, whereas the former study tracked those who died of cancer. Hormesis has the effect of curing those with an illness, and hence they do not die from that illness (including cancer).

### 26.2.9 Cancer, Overall Health, and Longevity

Because the study of health effects of ionizing radiation usually focuses on cancer, it is surprising that the data clearly show that the hormetic effects of low-level radiation doses improve the quality of life and longevity. It is uncertain if these improvements produced by low-level radiation are because of the lessening of cancer incidence or the curing of cancer once it develops. The work of Otto Raabe [13], especially experiments with cancer induced by high doses of radiation in dogs, may help clarify the above question. Raabe has essentially shown that cancer from radiation is not a statistical effect on the human body, but that cancer itself is essentially a deterministic effect. He has demonstrated that it takes repeated attacks, perhaps over years, to bring a cancer to fruition. This may explain the apparent delay of the order of 5 years for leukemia and up to 20 years for solid cancers from the time the individual was initially exposed to a cancer-causing agent (for instance, radiation, or chemical). These apparent latent periods for cancer are well documented.

### 26.2.10 Estimating Risks Using the Collective Dose Hypothesis

A corollary to the LNT is the Collective Dose Hypothesis, which, because of the presumed linearity of negative effect versus dose even to very low levels, says that the product of risk versus dose is a constant, regardless of the magnitude of either factor. The BEIR V report (1990) adopted a factor of 800 deaths per million person REM (cSv) of collective dose received. Thus, if 1,000,000 persons each received 1.0 REM (0.01 Sv), 800 of those would die. If 1 billion persons each received 0.001 REM (0.01 mSv, equivalent to about two days of natural background radiation), 800 of those would die as a result of the radiation exposure. Extending these risk estimates to low doses of radiation is as ridiculous as the example originally introduced by Alan Waltar, which is: If 100 aspirins were all taken by one person at one sitting, that person would die. Hence, 100 person-aspirins produce one death, so if 100 persons each took one aspirin (100 person-aspirins), one of those 100 would die. This kind of false reasoning in making risk analyses continues to be applied to nuclear power plant safety analysis reports, despite the fact that the LNT theory has been proven to be false when applied to low doses of radiation and is in fact 180° out of phase (what was assumed to be a negative effect is actually a positive effect up to a level that is orders of magnitude above average natural background).

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### 26.2.11 Conclusions

Radiation hormesis has been demonstrated to be real phenomenon in experiments on plants, bacteria, insects, animals, and with epidemiological and demographic data from humans. Not only does hormesis apply to radiation but also to low doses of physical,

chemical, and biological agents. This is especially true for most pharmaceuticals. So why is it considered so unusual for radiation? According to T.D. Luckey, there are over 3000 references which support the hormetic effect of radiation.

Failure for the industry and regulatory bodies to recognize the hormetic effects of radiation and to revise radiation protection standards accordingly is a travesty. To quote the late Theodore Rockwell, "...radiation protection policy and procedures do more harm than good, generally by focusing on reducing harmless radiation doses still further, fostering fear and uncertainty with no compensating benefit."

The harm in not applying the hormesis information to radiation exposures is considerable, not only in cost but in preventing appropriate development and utilization of nuclear power and of nuclear applications (such as CT scans). Quoting Nobel Laureate in Medicine, Rosalyn Yalow, "The unjustified excessive concern with radiation at any level, however, precludes beneficial uses of radiation and radioactivity in medicine, science, and industry" [14]. Doses below 10 cSv (10 REM or 0.1 Sv) per year ("chronic" dose) or 3 cSv (3 REM or 0.03 Sv) in a single dose are not worth avoiding, need not be a subject for regulation, and evidence shows that such doses are actually beneficial to health. Not until chronic doses are greater than about 500 times natural background levels, or acute doses 5 times that received in a full-body CT, is the physiological effect likely to be worse than it is for unexposed persons receiving only normal natural background radiation. In summary, radiation doses up to these limits are not merely harmless but beneficial to health. Radiation regulations need to be revised accordingly.

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