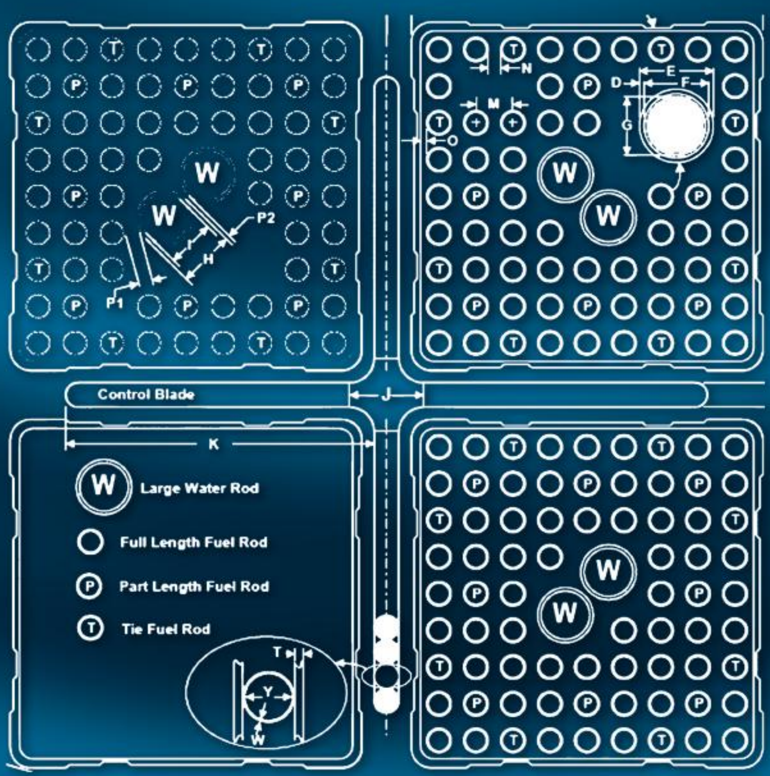
SECOND EDITION—REVISED PRINTING

NUCLEAR SYSTEMS

VOLUME 1

THERMAL HYDRAULIC FUNDAMENTALS



NEIL E. TODREAS • MUJID S. KAZIMI

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To our families for their support in this endeavor
Carol, Tim, and Ian
Nazik, Yasmeen, Marwan, and Omar

Contents

Preface

In the 22 years since the first edition appeared, nuclear power systems have regained favor worldwide as a scalable, dependable, and environmentally desirable energy supply technology. While water-cooled reactor types promise to dominate the deployment of nuclear power systems well into this century, several alternate cooled reactors among the six Generation IV proposed designs are now under intensive development for deployment in the next decades.

Consequently, while this new edition continues to emphasize pressurized and boiling light water reactor technologies, the principal characteristics and analysis approaches for the Generation IV designs of most international interest, the liquid-metal- and gas-cooled options, are introduced as well.

To accomplish this, the content of most chapters has been amplified by introducing both new analysis approaches and correlation methods which have gained the favor of experts over the past several decades. Nevertheless, these enhancements have been introduced into the sequence of chapters of the first edition, a sequence which has proven to be pedagogically effective. This organization of text chapters is shown in Table 0.1.

The content of these chapters emphasizing new material in particular is summarized below:

- Chapter 1 surveys the characteristics of all current principal reactor types: operating and advanced light water-cooled designs as well as the six Generation IV designs.
- Chapter 2 emphasizes the relations that integrate the performance of the core nuclear steam supply system and balance of plant. In doing so, the performance measures of the multiple disciplines—neutronics, thermal hydraulics, fuel behavior, structural mechanics, and reactor operations—are introduced and interrelated. In particular, power density and specific power measures are thoroughly developed for standard fuel pin arrays as well as inverted (fuel and coolant are spatially interchanged) assembly configurations.
- Chapter 3 presents a thorough description of the magnitude and spatial distribution of the generation and deposition of energy within the reactor vessel assembly. Emphasis is given to the evolution of the American Nuclear Society Decay Heat Generation Standard up to 2005. Multiple other sources of stored energy are present in reactor systems such as those associated with elevated temperature and pressure conditions of the primary and secondary reactor coolants and the chemical reactions of materials of construction under extreme temperatures characteristic of accident conditions. The stored energy and the energy liberated by the chemical reactions characteristic of the light water reactor materials are now presented in detail.

TABLE 0.1
Text Contents by Chapter and Subject

<table>
<tr><th colspan="5">1—Reactor-Type Overview</th></tr>
<tr><th colspan="5">2—Core and Plant Performance Measures</th></tr>
<tr><th colspan="5">3—Fission Energy Generation and Deposition</th></tr>
<tr><th></th><th>Conservation Equations</th><th>Thermodynamics</th><th>Fluid Flow</th><th>Heat Transfer</th></tr>
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</table>

- Chapters 4 and 5, which present the mass, momentum, energy, and entropy conservation equations for single- and two-phase coolants in differential and control volume formulations, are unchanged from the first edition.
- Chapter 6 presents the analysis of power generation cycles, both Rankine and Brayton types. The analysis of the supercritical carbon dioxide (SCO_2) cycle, which has recently been introduced as a performance and cost-efficient option for Generation IV systems, is presented. Of note, this presentation is designed to clearly illustrate how this recompression cycle must be designed and analyzed, a subtlety which otherwise easily confounds the inexperienced analyst.
- Chapter 7 applies thermodynamic principles for analysis of the fuel and coolant mixture in a severe accident, for analysis of containment design and accident performance, and for analysis of PWR pressurizer performance. The chapter remains as in the first edition with the fundamentals of pressurizer performance becoming the focus now by deletion of the more complex multiregion pressurizer model introduced in the first edition.
- Chapter 8 significantly amplifies the discussion of fuel pin materials and thermal analysis of various fuel geometries in the first edition. Extensive fuel and cladding material properties for thermal and fast neutron spectrum reactors are now included. Additionally, the case of annular fuel geometry, which is cooled inside and outside, is fully analyzed. Finally, the correlations for cladding-coolant surface oxidation are now included.
- Chapters 9 and 10 present the fluid flow and thermal analysis techniques for single-phase coolant in laminar and turbulent flow. Noteworthy new

material presents the analysis of geometries configured for enhanced heat transfer albeit with pressure loss penalty.

- Chapters 11 and 12 along with Chapter 13 present fluid flow and thermal analysis techniques for two-phase flow. Chapter 11 now has expanded coverage of critical flow and new coverage of flow instabilities and condensation. Boiling heat transfer is now separated into pool boiling and flow boiling in Chapters 12 and 13, respectively. The entire two-phase flow treatment in these three chapters has been expanded to include recent correlations of current usage, particularly for void fraction, pressure loss, and the critical condition.
- Chapter 14 has been thoroughly restructured to present a systematic analysis of heated flow channel performance for single- and two-phase flow utilizing the methods and correlations of Chapters 9 through 13. Specific focus is directed to conditions of equilibrium and nonequilibrium for both equal/unequal phasic temperatures (thermal equilibrium) and equal phasic flow velocities (mechanical equilibrium).

Finally, throughout the text we refer as appropriate to further elaboration of relevant technical information which appears in the companion *Volume II* of this text.*

* Todreas, N. E. and Kazimi, M. S., *Nuclear Systems II: Elements of Thermal Hydraulic Design*. New York: Taylor & Francis, 2001.

Preface to the First Edition

This book can serve as a textbook for two to three courses at the advanced undergraduate and graduate student level. It is also suitable as a basis for the continuing education of engineers in the nuclear power industry who wish to expand their knowledge of the principles of thermal analysis of nuclear systems. The book, in fact, was an outgrowth of the course notes used for teaching several classes at MIT over a period of nearly 15 years.

The book is meant to cover more than thermal hydraulic design and analysis of the core of a nuclear reactor. Thus, in several parts and examples, other components of the nuclear power plant, such as the pressurizer, the containment, and the entire primary coolant system, are addressed. In this respect the book reflects the importance of such considerations in thermal engineering of a modern nuclear power plant. The traditional concentration on the fuel element in earlier textbooks was appropriate when the fuel performance had a higher share of the cost of electricity than in modern plants. The cost and performance of nuclear power plants has proved to be more influenced by the steam supply system and the containment building than previously anticipated.

The desirability of providing in one book basic concepts as well as complex formulations for advanced applications has resulted in a more comprehensive textbook than those previously authored in the field. The basic ideas of both fluid flow and heat transfer as applicable to nuclear reactors are discussed in *Volume I*. No assumption is made about the degree to which the reader is already familiar with the subject. Therefore, various reactor types, energy source distribution, and fundamental laws of conservation of mass, momentum, and energy are presented in early chapters. Engineering methods for analysis of flow hydraulics and heat transfer in single-phase as well as two-phase coolants are presented in later chapters. In *Volume II*, applications of these fundamental ideas to multi-channel flow conditions in the reactor are described as well as specific design considerations such as natural convection and core thermal reliability. They are presented in a way that renders it possible to use the analytical development in simple exercises and as the bases for numerical computations similar to those commonly practiced in the industry.

A consistent nomenclature is used throughout the text and a table of the nomenclature is included in the Appendix. Each chapter includes problems identified as to their topic and the section from which they are drawn. While the SI unit system is principally used, British Engineering Units are given in brackets for those results commonly still reported in the United States in this system.

Acknowledgments

The authors are indebted to several fellow faculty, professional associates, staff, and students who provided significant assistance in preparing and reviewing this second edition of *Nuclear Systems Volume I*. Besides those named here, there were many who sent us notes and comments over the years, which we took into consideration in this second edition.

Our faculty colleague, Jacopo Buongiorno, provided essential insights on a number of technical topics introduced or expanded upon in this new edition. Also, he provided a collection of solved problems drawn from MIT courses taught using this text, which have been integrated into the exercises at the end of each chapter. Our faculty colleagues, Arthur Bergles, Michael Driscoll, and Eugene Shwageraus, as well as our professional associates, Mahmoud Massoud, Tom Newton, Aydin Karadin, and Charles Kling, carefully proofed major portions of the text and offered materials and comments for our consideration.

The following students greatly assisted in the preparation of the text by researching the technical literature as well as preparing figures and example solutions: Muhammed Ayanoglu, Tom Conboy, Jacob DeWitte, Paolo Ferroni, Giancarlo Lenci, and Wenfeng Liu. Bryan Herman prepared the manual of solutions to all homework problems in the text. In the proofreading of the final manuscript we were greatly assisted by the following students: Nathan Andrews, Tyrell Arment, Ramsey Arnold, Jacob DeWitte, Mihai Diaconeasa, You Ho Lee, Giancarlo Lenci, Alexander Mieloszyk, Stefano Passerini, Joshua Richard, Koroush Shirvan, John Stempien, and Francesco Vitillo.

The Problemsolver software was developed by Dr. Massoud, and several Excel spreadsheets were developed by Giancarlo Lenci for critical heat flux application. We convey our additional thanks to both for providing these useful problem-solving tools to enhance our text.

The preparation of the manuscript was expertly and conscientiously done by Richard St. Clair, who tirelessly worked on the iterations of insertions and deletions of material in the text. We also extend our gratitude to Paula Cornelio who prepared the majority of the new figures for this edition as she equally did for the original text.

We also acknowledge the review of our proposed plan for the technical content of this second edition by Professors Samim Anghaie, Fred Best, Larry Hockreiter, and Michel Giot. Their observations and suggestions were influential in our final selection of topics to be rewritten, added, and deleted, although the final selection was made by us.

PREPARATION OF THE SECOND EDITION-REVISED PRINTING

The students, Giancarlo Lenci and Pierre Guenoun, and faculty, Jacopo Buongiorno and Yuksel Parlatan, were of great assistance in identifying needed edits for this revised edition. Their efforts are greatly appreciated.

PROBLEMSOLVER SOFTWARE AND ERRATA

The CRC book website, found at https://www.crcpress.com/product/isbn/9781439808870 (Downloads/Updates tab), contains the Problemsolver software and book Errata. It also contains information on how to access the updated online listing of errata, on how to report newly identified errata to the authors, and on how to access extra technical content related to a future Third Edition.

Authors

Neil E. Todreas is Professor Emeritus in the Departments of Nuclear Science and Engineering and Mechanical Engineering at the Massachusetts Institute of Technology. He held the Korea Electric Power Corporation (KEPCO) chair in Nuclear Engineering from 1992 until his retirement to part-time activities in 2006. He served an 8-year period from 1981 to 1989 as the Nuclear Engineering Department Head. Since 1975 he has been a codirector of the MIT Nuclear Power Reactor Safety summer course, which presents current issues of reactor safety significant to an international group of nuclear engineering professionals. His area of technical expertise includes thermal and hydraulic aspects of nuclear reactor engineering and safety analysis. He started his career at Naval Reactors working on submarine and surface nuclear vessels after earning the B.Eng. and the M.S. in mechanical engineering from Cornell University. Following his Sc.D. in Nuclear Engineering at MIT, he worked for the Atomic Energy Commission (AEC) on organic cooled/heavy water-moderated and sodium-cooled reactors until he returned as a faculty member to MIT in 1970. He has an extensive record of service for government (Department of Energy (DOE), U.S. Nuclear Regulatory Commission (USNRC), and national laboratories) and utility industry review committees including INPO, and international scientific review groups. He has authored more than 200 publications and a reference book on safety features of light water reactors. He is a member of the U.S. National Academy of Engineering and a fellow of the American Nuclear Society (ANS) and the American Society of Mechanical Engineers (ASME). He has received the American Nuclear Society Thermal Hydraulics Technical Achievement Award, its Arthur Holly Compton Award in Education, and its jointly conferred (with the Nuclear Energy Institute) Henry DeWolf Smyth Nuclear Statesman Award.

Mujid S. Kazimi is Professor in the Departments of Nuclear Science and Engineering and Mechanical Engineering at the Massachusetts Institute of Technology. He is the director of the Center for Advanced Nuclear Energy Systems (CANES) and holds the Tokyo Electric Power Company (TEPCO) chair in Nuclear Engineering at MIT. Prior to joining the MIT faculty in 1976, Dr. Kazimi worked for a brief period at the Advanced Reactors Division of Westinghouse Electric Corporation and at Brookhaven National Laboratory. At MIT he was head of the Department of Nuclear Engineering from 1989 to 1997 and Chair of the Safety Committee of the MIT Research Reactor from 1998 to 2009. He has served since 1990 as the codirector of the MIT Nuclear Power Reactor Safety summer course. He is active in the development of innovative designs of fuel and other components of nuclear power plants and in analysis of the nuclear fuel cycle options for sustainable nuclear energy. He cochaired the 3-year MIT interdisciplinary study on the *Future of the Nuclear Fuel Cycle*, published in 2011. He has served on scientific advisory committees at the U.S. National Academy of Engineering and several other national agencies and laboratories in the United States, Japan, Spain, Switzerland, Kuwait, the United

Arab Emirates, and the International Atomic Energy Agency. He has authored more than 200 articles and papers that have been published in journals and presented at international conferences. Dr. Kazimi holds a B.Eng. degree from Alexandria University in Egypt, and M.S. and Ph.D. degrees from MIT, all in nuclear engineering. He is a fellow of the American Nuclear Society and the American Association for the Advancement of Science. Among his honors is the Technical Achievement Award in Thermal Hydraulics by the American Nuclear Society.

1 Principal Characteristics of Power Reactors

1.1 INTRODUCTION

This chapter presents the basic characteristics of power reactors. These characteristics, along with more detailed thermal hydraulic parameters presented in further chapters, enable the student to apply the specialized techniques presented in the remainder of the text to a range of reactor types. Water-, gas-, and sodium-cooled reactor types, identified in Table 1.1, encompass the principal nuclear power reactor designs that have been employed in the world. The thermal hydraulic characteristics of these reactors are presented in Sections 1.2 through 1.5 as part of the description of the power cycle, primary coolant system, core, and fuel assembly design of these reactor types. Three classes of advanced reactors are also presented in subsequent sections, the Generation III, III+, and IV designs. The Generation III designs are advanced water reactors that have already been brought into operation (ABWR) or are under construction (EPR). The Generation III+ designs are advanced water- and gas-cooled reactors, several of which are being licensed and brought into service in the 2010 decade [12]. These Generation III and III+ designs are discussed in Section 1.6. The Generation IV reactors described in Section 1.7 were selected by an international roadmapping process and are being pursued through an internationally coordinated research and development activity for deployment in the period 2020–2040 [13]. Figure 1.1 presents the evolution and categorization by the generation of the world's reactor types. Tables in Chapters 1 and 2 and Appendix K provide detailed information on reactor characteristics useful for application to specific illustrative examples and homework problems in the text.

1.2 POWER CYCLES

In these plants, a primary coolant is circulated through the reactor core to extract energy for ultimate conversion to electricity in a turbine connected to an electric generator. Depending on the reactor design, the turbine may be driven directly by the primary coolant or by a secondary coolant that has received energy from the primary coolant. The number of coolant systems in a plant equals the sum of the one primary and one or more secondary systems. For the boiling water reactor (BWR) and the high-temperature gas reactor (HTGR) systems, which produce steam and hot helium by passage of a primary coolant through the core, direct use of these primary coolants in the turbine is possible, leading to a single-coolant system. The BWR

TABLE 1.1
Basic Features of Major Power Reactor Types

Reactor Type	Neutron Spectrum	Moderator	Coolant	Fuel: Chemical Form	Fuel: Approximate Fissile Content (All ^{235}U Except the Sodium-Cooled Reactors)
Water-cooled	Thermal				
PWR		H_2O	H_2O	UO_2	3–5% enrichment
BWR		H_2O	H_2O	UO_2	3–5% enrichment
PHWR (CANDU)		D_2O	D_2O	UO_2	Natural
SGHWR[a]		D_2O	H_2O	UO_2	~3% enrichment
Gas-cooled	Thermal	Graphite			
Magnox			CO_2	U metal	Natural
AGR			CO_2	UO_2	~3% enrichment
HTGR			Helium	UO_2	~7–20% enrichment[b]
Sodium-cooled	Fast	None	Sodium		
SFBR[c]				UO_2/PuO_2	~15–20% of HM is Pu[e]
SFR[d]				NU–TRU–Zr[f] metal or oxide	~15% of HM is TRU

[a] Steam-generating heavy water reactor.
[b] Older operating plants have enrichments of more than 90% and used a variety of thorium and carbide fuel forms.
[c] Sodium-cooled fast-breeder reactor.
[d] Sodium fast reactor operating on a closed cycle.
[e] Heavy metal (HM).
[f] Natural uranium (NU), transuranic elements (TRU), and zirconium (Zr).

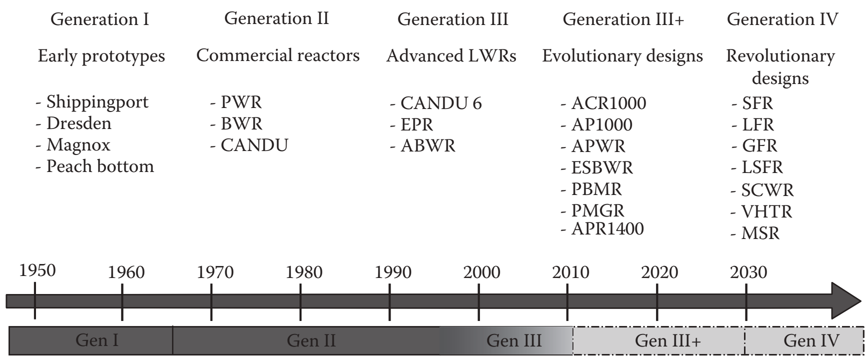

FIGURE 1.1 The evolution of nuclear power. (Adopted from U.S. Department of Energy, http://www.gen-4.org/Technology/evolution.htm.)

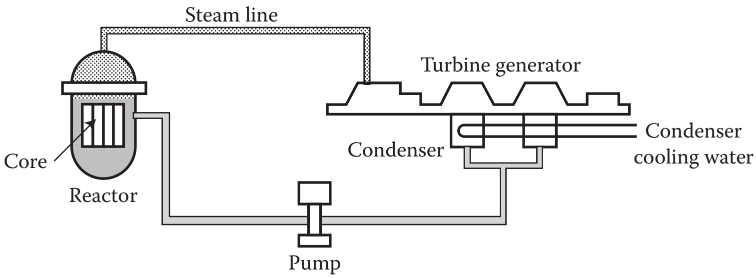

FIGURE 1.2 Direct, single-coolant Rankine cycle. (Adopted from U.S. Department of Energy.)

single-coolant system, based on the Rankine cycle (Figure 1.2), is in common use. The Fort St. Vrain HTGR plant used a secondary water system in a Rankine cycle because the technology did not exist to produce a large, high-temperature, helium-driven turbine. Although the HTGR direct turbine system has not yet been built for a commercial reactor, it would use the Brayton cycle, as illustrated in Figure 1.3. Thermodynamic analyses for typical Rankine and Brayton cycles are presented in Chapter 6.

The pressurized water reactor (PWR) and the pressurized heavy water reactor (PHWR) are two-coolant systems. This design is necessary to maintain the primary coolant conditions at a nominal subcooled liquid state while the turbine is driven by steam in the secondary system. Figure 1.4 illustrates the PWR two-coolant steam cycle.

The sodium-cooled fast reactors (both SFRs and SFBR) employ three-coolant systems: a primary sodium coolant system, an intermediate sodium coolant system, and a steam–water, turbine–condenser coolant system (Figure 1.5). The sodium-to-sodium heat exchange is accomplished in an intermediate heat exchanger (IHX), and the sodium-to-water/steam heat exchange in a steam generator. Three-coolant systems were specified to isolate the radioactive primary sodium coolant from the

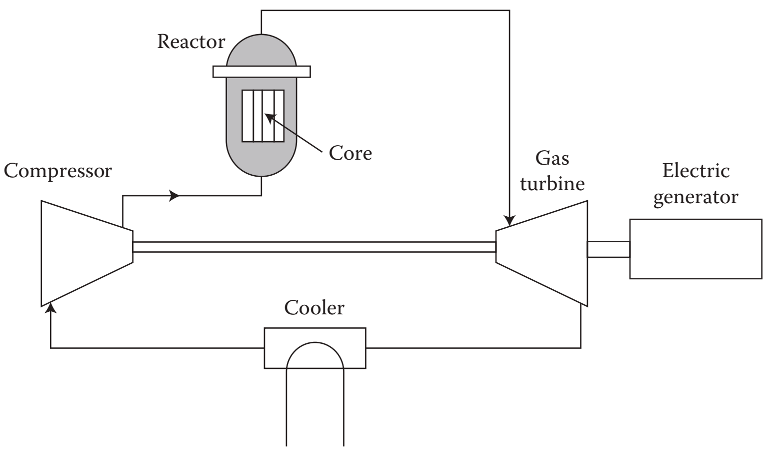

FIGURE 1.3 Direct, single-coolant Brayton cycle. (Adopted from U.S. Department of Energy.)

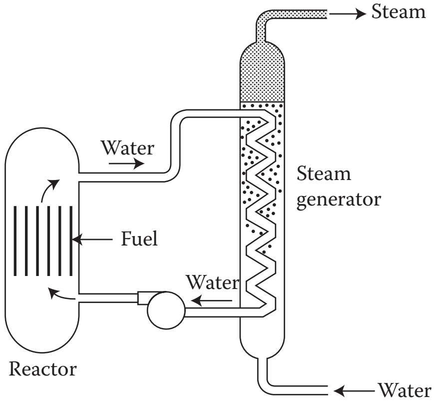

FIGURE 1.4 Two-coolant system steam cycle. (From Shultis, J. K. and Faw, R. E., *Fundamentals of Nuclear Science and Engineering*, 2nd Ed. CRC Press, Boca Raton, FL, 2008.)

steam–water circulating through the turbine, condenser, and associated conventional plant components. The SFR concept being developed in the United States draws on this worldwide SFR technology and the operational experience base, but it is not designed as a breeder. Sodium-cooled reactor characteristics and examples presented in this chapter are for both the SFBRs, which were built in the late 1900s, and the SFR, which is currently under development and design.

The significant characteristics of the thermodynamic cycles and coolant systems used in these reference reactor types are summarized in Table 1.2.

1.3 PRIMARY COOLANT SYSTEMS

The Generation II BWR single-loop primary coolant system is illustrated in Figure 1.6, while Figure 1.7 highlights the flow paths within the reactor vessel. The

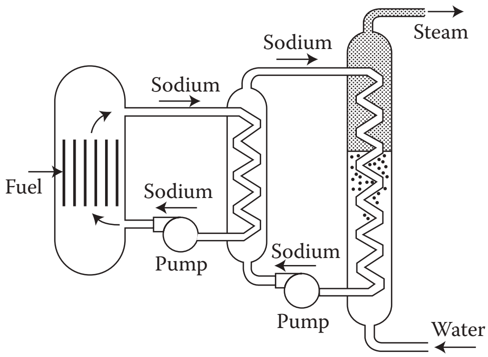

FIGURE 1.5 Three-coolant system steam cycle. (From Shultis, J. K. and Faw, R. E., *Fundamentals of Nuclear Science and Engineering*, 2nd Ed. CRC Press, Boca Raton, FL, 2008.)

TABLE 1.2
Typical Characteristics of the Thermodynamic Cycle for Six Reference Power Reactor Types

Characteristics	BWR	PWR	PHWR	HTGR	AGR	SFBR
			Reference Design			
Manufacturer	General Electric	Westinghouse	Atomic Energy of Canada, Ltd.	General Atomic	National Nuclear Corp.	Novatome
System (reactor station)	BWR/5 (NMP2)	(Seabrook)	CANDU-600	(Fulton)[a]	(Heysham 2)	(Superphenix-1)
Steam cycle						
No. coolant systems	1	2	2	2	2	3
Primary coolant	H_2O	H_2O	D_2O	He	CO_2	Liq. Na
Secondary coolant	[b]	H_2O	H_2O	H_2O	H_2O	Liq. Na/H_2O
			Energy Conversion			
Thermal power, MW_t	3323	3411	2180	3000	1551	3000
Electric power, MW_e	1062	1148	638	1160	660	1240
Efficiency (%)	32.0	33.7	29.3	38.6	42.5	41.3
			Heat Transport System			
No. of primary loops/pumps	2/2	4/4	2/2	6/6	8/8	Pool with 4 pumps
No. of IHXs	0	0	0	0	0	8
No. of steam generators	[b]	4	4	6	8	4
Steam generator type	[b]	U tube	U tube	Helical coil	Helical coil	Helical coil

continued

TABLE 1.2 (continued)
Typical Characteristics of the Thermodynamic Cycle for Six Reference Power Reactor Types

Characteristics	BWR	PWR	PHWR	HTGR	AGR	SFBR
Thermal Hydraulics						
Primary coolant						
Pressure (MPa)	7.14	15.51	10.0	5.0	4.27	~0.1
Inlet temp. (°C)	278.3	293.1	267	318	292	395
Ave. outlet temp. (°C)	286.1	326.8	310	741	638	545
Core flow rate (Mg/s)	13.671	17.476	7.6	1.41	3.92	15.7
Volume or mass	—	336 m^3	120 m^3	7850 kg	5300 m^3	3.2×10^6 kg Na/H_2O
Secondary coolant						
Pressure (MPa)	[b]	6.89	4.7	17.3	17.0	~0.1/17.7
Inlet temp. (°C)	[b]	227	187	188	157.0	345/237
Outlet temp. (°C)	[b]	285	260	513	543.0	525/490

Source: BWR and PWR: Adopted from *Seabrook Power Station Updated Safety Analysis Report*, Revision 8, Seabrook Station, Seabrook, NH, 2002; Appendix K. PHWR: Adopted from Knief, R. A. *Nuclear Engineering: Theory and Technology of Commercial Nuclear Power*, pp. 707–717. American Nuclear Society, La Grange Park, IL, 2008. HTGR: Adopted from Breher, W., Neyland, A., and Shenoy, A. *Modular High-Temperature Gas-Cooled Reactor (MHTGR) Status*. GA Technologies, GA-A18878, May 1987. AGR-Heysham 2: Adopted from AEAT/R/PSEG/0405 Issue 3. *Main Characteristics of Nuclear Power Plants in the European Union and Candidate Countries*. Report for the European Commission, September 2001; *Nuclear Engineering International*. Supplement. August 1982; Alderson, M. A. H. G. (UKAEA, pers. comm., October 6, 1983 and December 6, 1983). SuperPhenix-1: Adopted from IAEA-TECDOC-1531. *Fast Reactor Database 2006 Update.* International Atomic Energy Agency. December 2006. The Russian VVER is similar to the US PWR while their RBMK, which is no longer being built, is a low-enriched uranium oxide-fueled, light water-cooled, graphite-moderated pressure tube design.

[a] Designed but not built.
[b] Not applicable.

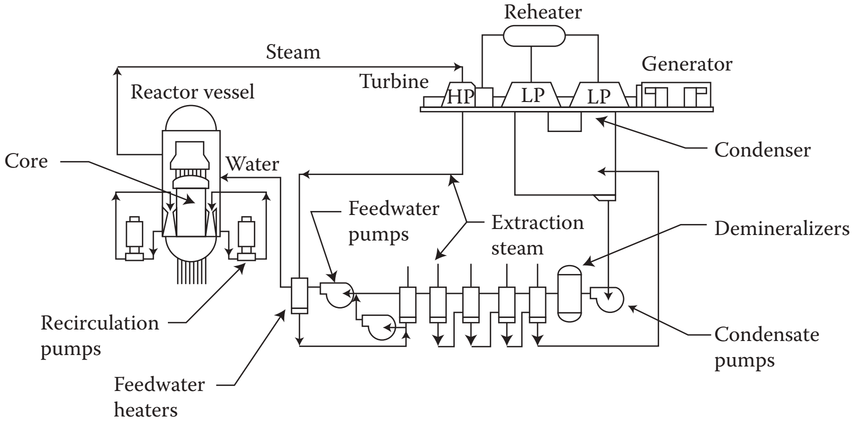

FIGURE 1.6 BWR single-loop primary coolant system. (From Shultis, J. K. and Faw, R. E., *Fundamentals of Nuclear Science and Engineering*, 2nd Ed. CRC Press, Boca Raton, FL, 2008.)

steam–water mixture first enters the steam separators after exiting the core. After subsequent passage through a steam separator and dryer assembly located in the upper portion of the reactor vessel, dry saturated steam flows directly to the turbine. Saturated water, which is separated from the steam, flows downward in the periphery of the reactor vessel and mixes with the incoming main feed flow from the condenser. This combined flow stream is pumped into the lower plenum through jet pumps mounted around the inside periphery of the reactor vessel. The jet pumps are driven

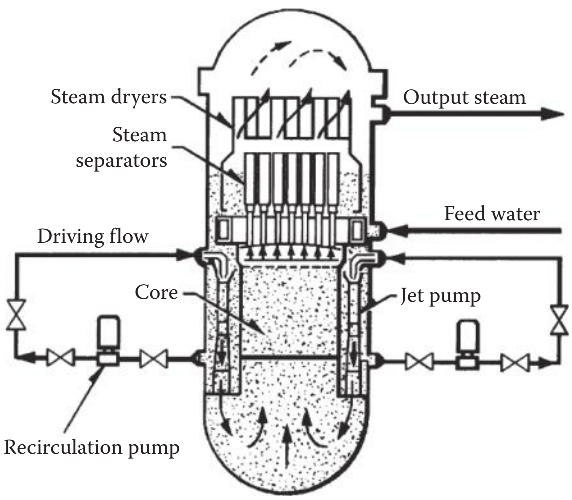

FIGURE 1.7 Steam and recirculation water flow paths in the Generation II BWR. (From Shultis, J. K. and Faw, R. E., *Fundamentals of Nuclear Science and Engineering*, 2nd Ed. CRC Press, Boca Raton, FL, 2008.)

by flow from recirculation pumps located in relatively small-diameter (~50 cm) external recirculation loops, which draw flow from the plenum just above the jet pump discharge location. In the ABWR, all external recirculation loops are eliminated and replaced with recirculation pumps placed internal to the reactor vessel. In the economic simplified boiling water reactor (ESBWR), all jet pumps as well as external recirculation pumps were eliminated by the natural circulation flow design.

In all BWRs the core flowrate is much greater than the feed water flowrate, reflecting the fact that the average core exit quality $\bar{x}_e$ is about 15%. Hence, the recirculation ratio (RR) is obtained as

$$\text{RR} \equiv \frac{\text{Mass flowrate of recirculated liquid}}{\text{Mass flowrate of vapor produced}} = \frac{1-\bar{x}_{\text{exit}}}{\bar{x}_{\text{exit}}} = \frac{0.85}{0.15} = 5.7 \tag{1.1}$$

The primary coolant system of a PWR consists of a multiloop arrangement arrayed around the reactor vessel. Higher power reactor ratings are achieved by adding loops of identical design. Designs of two, three, and four loops have been built with three- and four-loop reactors being the most common. In a typical four-loop configuration (Figure 1.8), each loop has a vertically oriented steam generator* and

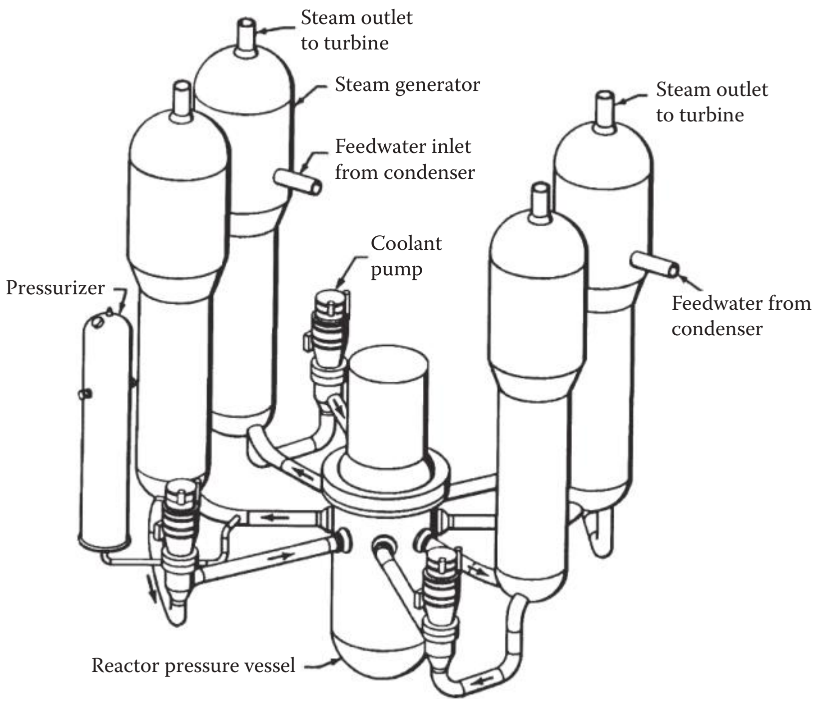

FIGURE 1.8 Arrangement of the primary system for a Generation II PWR. (From Shultis, J. K. and Faw, R. E., *Fundamentals of Nuclear Science and Engineering*, 2nd Ed. CRC Press, Boca Raton, FL, 2008.)

* Russian VVERs employ horizontal steam generators.

coolant pump. The coolant flows through the steam generator within an array of U tubes that connect the inlet and outlet plena located at the bottom of the steam generator. The system's single pressurizer is connected to the hot leg of one of the loops. The hot (reactor vessel to steam generator inlet) and cold (steam generator outlet to reactor vessel) leg pipes are typically 31–42 and 29–30 in. (78.7–106.7 and 73.7–76.2 cm) in diameter, respectively.

The flow path through the PWR reactor vessel is illustrated in Figure 1.9. The inlet nozzles communicate with an annulus formed between the inside of the reactor vessel and the outside of the core support barrel. The coolant entering this annulus flows downward into the inlet plenum formed by the lower head of the reactor vessel. Here it turns upward and flows through the core into the upper plenum that communicates with the reactor vessel's outlet nozzles.

The HTGR primary system is composed of several loops, each housed within a large cylinder of prestressed concrete. A compact HTGR arrangement as embodied in the modular high-temperature gas-cooled reactor (MHTGR) is illustrated in

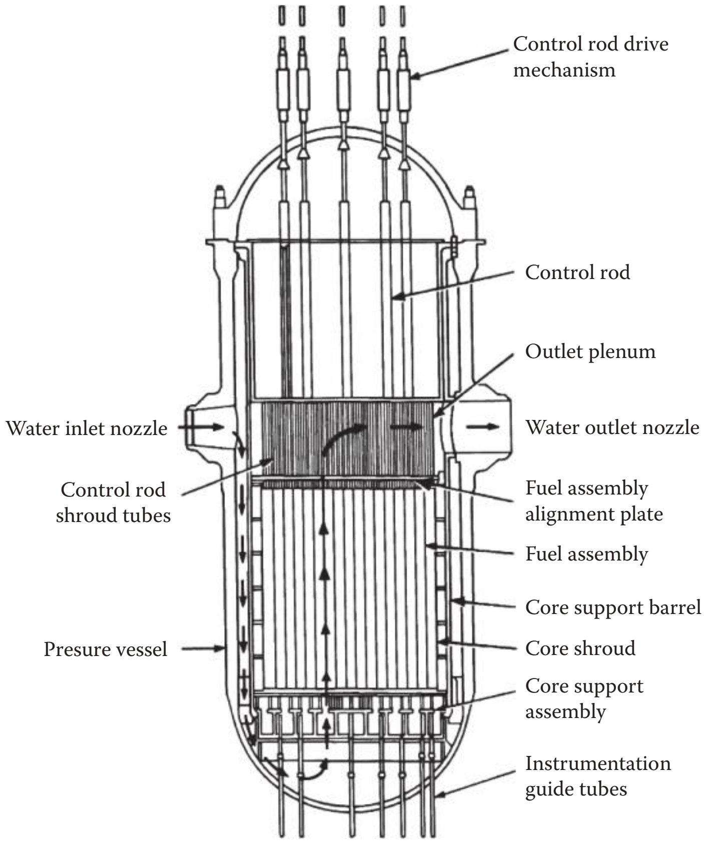

FIGURE 1.9 Flow path through a PWR reactor vessel. (From Shultis, J. K. and Faw, R. E., *Fundamentals of Nuclear Science and Engineering*, 2nd Ed. CRC Press, Boca Raton, FL, 2008.)

Figure 1.10. In this 588 MWe MHTGR arrangement [2], the flow is directed downward through the core by a circulator mounted above the steam generator in the cold leg. The reactor vessel and steam generator are connected by a short, horizontal cross duct, which channels two oppositely directed coolant streams. The coolant from the core exit plenum is directed laterally through the 47 in. (119.4 cm) interior diameter region of the cross duct into the inlet of the steam generator. The coolant from the steam generator and circulator is directed laterally through the outer annulus (equivalent pipe diameter of approximately 46 in. [116.8 cm]) of the cross duct into the core inlet plenum and then upward through the reactor vessel's outer annulus into the inlet core plenum at the top of the reactor vessel.

SFBR and SFR primary systems have been of the loop and pool types. The pool-type configuration of the Superphenix reactor [14] is shown in Figure 1.11. Its characteristics are detailed in Table 1.2. The coolant flow path is upward through the reactor core into the upper sodium pool of the main vessel. The coolant from this pool flows downward by gravity through the IHX and discharges into a low-pressure

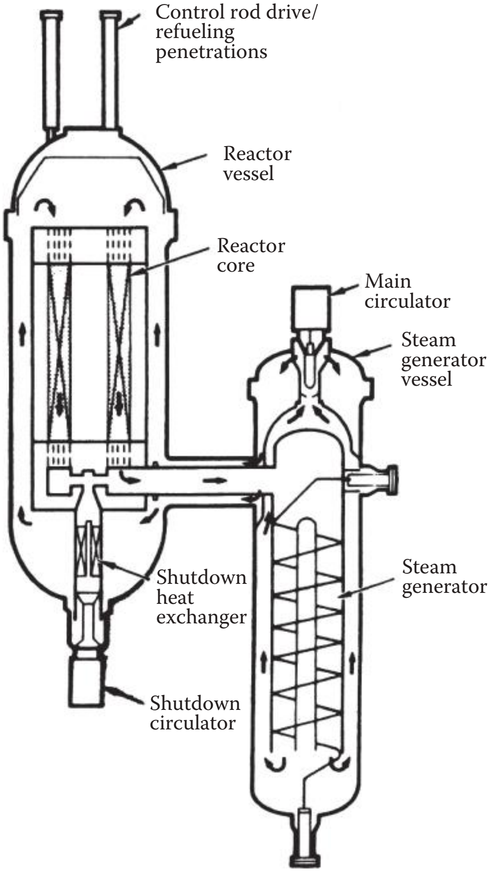

FIGURE 1.10 Modular HTGR primary coolant flow path. (Courtesy of U.S. Department of Energy.)

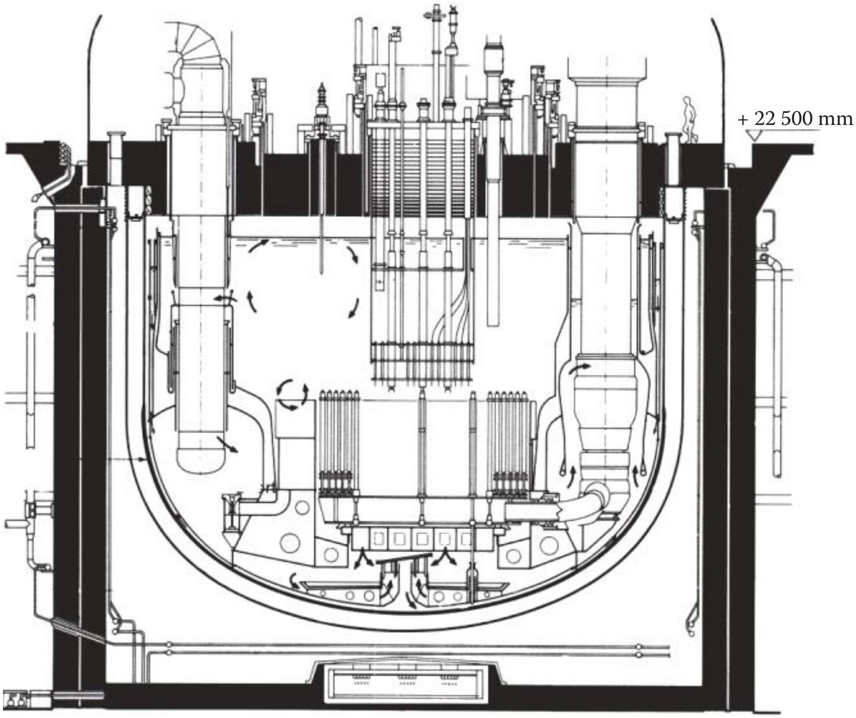

FIGURE 1.11 Primary system sodium flow path in the Superphenix reactor. (Courtesy of Électricité de France.)

toroidal plenum located in the periphery of the lower portion of the main vessel. Vertically oriented primary pumps draw the coolant from this low-pressure plenum and discharge it into the core inlet plenum.

1.4 REACTOR CORES

The reactor cores of all these reactors, except for the HTGR, are composed of assemblies of cylindrical fuel rods surrounded by the coolant that flows along the rod length. The prismatic HTGR core consists of graphite moderator hexagonal blocks that function as fuel assemblies. The blocks or assemblies are described in detail in Section 1.5.

There are two design features that establish the principal thermal hydraulic characteristics of reactor cores: the orientation and the degree of hydraulic isolation of an assembly from its neighbors. It is simple to adopt a reference case and describe the exceptions. Let us take as the reference case a vertical array of assemblies that communicate only at inlet and exit plena. This reference case describes the BWR, SFBR, and the advanced gas reactor (AGR) systems. The HTGR is nominally configured in this manner also, although leakage between the graphite blocks that are stacked to create the proper core length creates a substantial degree of communication between coolant passages within the core. The PHWR core consists of horizontal pressure tubes penetrating a low-pressure calandria tank filled with a heavy water moderator. The fuel assemblies housed within the pressure tubes are cooled by high-pressure heavy water, which is directed to and from the tubes by an array of inlet and outlet headers. The more advanced Canadian reactors use light water for cooling within the pressure tubes but retain heavy water in the calandria tank. Both the PHWR and the AGR are designed for online refueling.

The PWR and BWR assemblies are vertical, but unlike the BWR design, the PWR assemblies are not isolated hydraulically by enclosing the fuel rod array within ducts (called fuel channels in the BWR) over the core length. Hence, PWR fuel rods are grouped into assemblies only for handling and other structural purposes.

1.5 FUEL ASSEMBLIES

The principal characteristics of power reactor fuel bundles are the array (geometric layout and rod spacing) and the method of fuel pin separation and support along their span. The light water reactors (BWR and PWR), PHWR, AGR, and SFBR/SFR all use fuel rods. The HTGR has graphite moderator blocks in which adjacent penetrating holes for fuel and flowing helium coolant exist.

Light water reactors (LWRs), where the coolant also serves as the moderator, have small fuel-to-water volume ratios (commonly called the *metal-to-water ratio*) and consequently rather large fuel rod centerline-to-centerline spacing (commonly called the *rod pitch, P*). This moderate packing fraction permits the use of a simple square array and requires a rod support scheme of moderately small frontal area to yield low-pressure drops. The one LWR exception is the VVER, which uses a hexagonal array. A variety of grid support schemes have evolved for these applications.

Heavy water reactors and advanced gas reactors are designed for online refueling and consequently consist of fuel assemblies stacked within circular pressure tubes. This circular boundary leads to an assembly design with an irregular geometric array of rods. The online refueling approach has led to short fuel bundles in which the rods are supported at the assembly ends and by a center brace rather than by LWR-type grid spacers.

SFRs require no moderator and achieve high-power densities by compact hexagonal fuel rod array packing. With this tight rod-to-rod spacing, a lower pressure drop is obtained using spiral wire wrapping around each rod than could be obtained with a grid-type spacer. This wire wrap serves a dual function: as a spacer and as a promoter of coolant mixing within the fuel bundle. However, some SFR assemblies do use grid spacers.

The principal characteristics of the fuel for the six reference power reactor types are summarized in Table 1.3. The HTGR does not consist of an array of fuel rods within a coolant continuum. Rather, the HTGR blocks that contain fuel compacts, a coolant, and a moderator are designated as inverted fuel assemblies. In these blocks, the fuel–moderator combination is the continuum that is penetrated by isolated, cylindrically shaped coolant channels.

The LWRs (PWR and BWR), PHWR, AGR, and SFBR utilize an array of fuel rods surrounded by the coolant. For each of these arrays, the useful geometric characteristics are given in Table 1.3 and typical subchannels identified in Figure 1.12.

TABLE 1.3
Typical Characteristics of the Fuel for Six Reference Power Reactor Types

Characteristics	BWR	PWR	PHWR	HTGR	AGR	SFBR
			Reference Design			
Manufacturer	General Electric	Westinghouse	Atomic Energy of Canada, Ltd.	General Atomic	National Nuclear Corp.	Novatome
System (reactor station)	BWR/5 (NMP2)	(Seabrook)	CANDU-600	(Fulton)	(Heysham 2)	(Superphenix 1)
Moderator	H_2O	H_2O	D_2O	Graphite	Graphite	None
Neutron energy	Thermal	Thermal	Thermal	Thermal	Thermal	Fast
Fuel production	Converter	Converter	Converter	Converter	Converter	Breeder
			Fuel[b]			
Geometry	Cylindrical pellet	Cylindrical pellet	Cylindrical pellet	Microspheres[c]	Cylindrical pellet	Cylindrical pellet
Dimensions (mm)	$9.60D \times 10.0L$	$8.192D \times 9.8L$	$12.2D \times 16.4L$	400–800 μm D	$14.51D \times 14.51L$	7.14 D
Chemical form	UO_2	UO_2	UO_2	UC/ ThO_2	UO_2	PuO_2 /UO_2
Fissile (first core avg. wt% unless designated as equilibrium core)	^{235}U (3.5 eq. core)	^{235}U (3.57 avg. eq. core)	^{235}U (0.711)	^{235}U (93)	^{235}U (2 zones at 2.1 and 2.7)	^{239}Pu (2 zones at 16 and 19.7)
Fertile	^{238}U	^{238}U	^{238}U	Th	^{238}U	Depleted U
			Fuel Rods			
Geometry	Pellet stack in clad tube	Pellet stack in clad tube	Pellet stack in clad tube	Cylindrical fuel compacts	Pellet stack in clad tube	Pellet stack in clad tube

continued