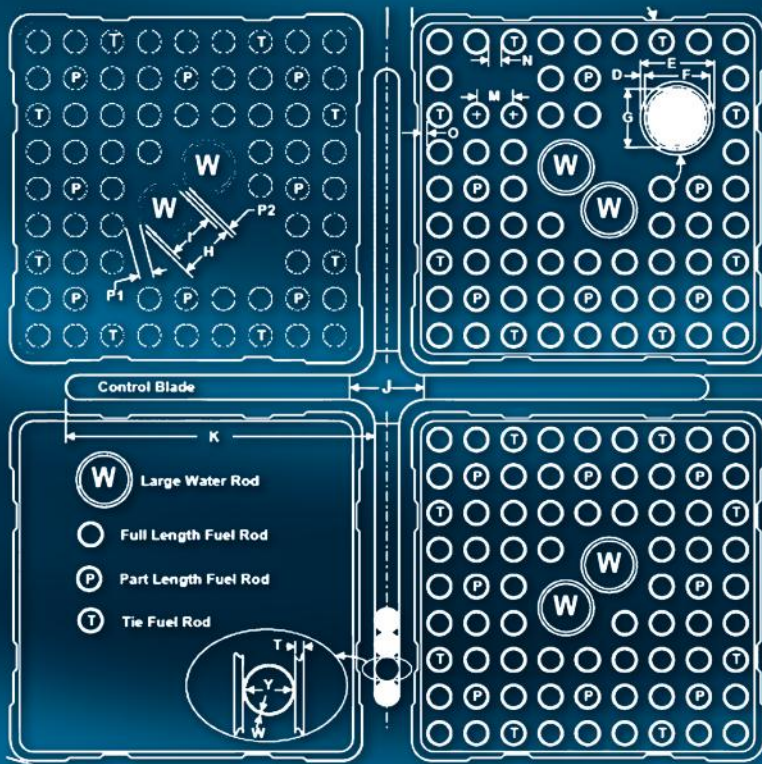


SECOND EDITION—REVISED PRINTING

NUCLEAR SYSTEMS

VOLUME 1

THERMAL HYDRAULIC FUNDAMENTALS



NEIL E. TODREAS • MUJID S. KAZIMI

SECOND EDITION

NUCLEAR SYSTEMS

VOLUME 1

THERMAL HYDRAULIC FUNDAMENTALS

SECOND EDITION

NUCLEAR SYSTEMS

VOLUME 1
THERMAL HYDRAULIC FUNDAMENTALS

NEIL E. TODREAS • MUJID S. KAZIMI



CRC Press

Taylor & Francis Group

Boca Raton London New York

CRC Press is an imprint of the
Taylor & Francis Group, an **informa** business

CRC Press
Taylor & Francis Group
6000 Broken Sound Parkway NW, Suite 300
Boca Raton, FL 33487-2742

© 2011 by Taylor & Francis Group, LLC
CRC Press is an imprint of Taylor & Francis Group, an Informa business

No claim to original U.S. Government works
Version Date: 20150603

International Standard Book Number-13: 978-1-4398-0888-7 (eBook - PDF)

This book contains information obtained from authentic and highly regarded sources. Reasonable efforts have been made to publish reliable data and information, but the author and publisher cannot assume responsibility for the validity of all materials or the consequences of their use. The authors and publishers have attempted to trace the copyright holders of all material reproduced in this publication and apologize to copyright holders if permission to publish in this form has not been obtained. If any copyright material has not been acknowledged please write and let us know so we may rectify in any future reprint.

Except as permitted under U.S. Copyright Law, no part of this book may be reprinted, reproduced, transmitted, or utilized in any form by any electronic, mechanical, or other means, now known or hereafter invented, including photocopying, microfilming, and recording, or in any information storage or retrieval system, without written permission from the publishers.

For permission to photocopy or use material electronically from this work, please access www.copyright.com (<http://www.copyright.com/>) or contact the Copyright Clearance Center, Inc. (CCC), 222 Rosewood Drive, Danvers, MA 01923, 978-750-8400. CCC is a not-for-profit organization that provides licenses and registration for a variety of users. For organizations that have been granted a photocopy license by the CCC, a separate system of payment has been arranged.

Trademark Notice: Product or corporate names may be trademarks or registered trademarks, and are used only for identification and explanation without intent to infringe.

Visit the Taylor & Francis Web site at
<http://www.taylorandfrancis.com>

and the CRC Press Web site at
<http://www.crcpress.com>

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

Contents

Preface.....	xxi
Preface to the First Edition	xxv
Acknowledgments.....	xxvii
Authors.....	xxix
Chapter 1 Principal Characteristics of Power Reactors	1
1.1 Introduction	1
1.2 Power Cycles	1
1.3 Primary Coolant Systems.....	4
1.4 Reactor Cores	11
1.5 Fuel Assemblies.....	12
1.5.1 LWR Fuel Bundles: Square Arrays.....	16
1.5.2 PHWR and AGR Fuel Bundles: Mixed Arrays	18
1.5.3 SFBR/SFR Fuel Bundles: Hexagonal Arrays	18
1.6 Advanced Water- and Gas-Cooled Reactors (Generations III and III+).....	20
1.7 Advanced Thermal and Fast Neutron Spectrum Reactors (Generation IV)	20
Problem	28
References	28
Chapter 2 Thermal Design Principles and Application	29
2.1 Introduction	29
2.2 Overall Plant Characteristics Influenced by Thermal Hydraulic Considerations	29
2.3 Energy Production and Transfer Parameters.....	35
2.4 Thermal Design Limits	37
2.4.1 Fuel Pins with Metallic Cladding	37
2.4.2 Graphite-Coated Fuel Particles	41
2.5 Thermal Design Margin	42
2.6 Figures of Merit for Core Thermal Performance	45
2.6.1 Power Density.....	45
2.6.2 Specific Power	46
2.6.3 Power Density and Specific Power Relationship.....	49
2.6.4 Specific Power in Terms of Fuel Cycle Operational Parameters.....	51
2.7 The Inverted Fuel Array	58
2.8 The Equivalent Annulus Approximation	62
Problems.....	67
References	69

Chapter 3	Reactor Energy Distribution.....	71
3.1	Introduction	71
3.2	Energy Generation and Deposition	71
3.2.1	Forms of Energy Generation	71
3.2.2	Energy Deposition.....	74
3.3	Fission Power and Calorimetric (Core Thermal) Power	75
3.4	Energy Generation Parameters.....	76
3.4.1	Energy Generation and Neutron Flux in Thermal Reactors	76
3.4.2	Relation between Heat Flux, Volumetric Energy Generation, and Core Power	82
3.4.2.1	Single Pin Parameters.....	82
3.4.2.2	Core Power and Fuel Pin Parameters	84
3.5	Power Profiles in Reactor Cores	87
3.5.1	Homogeneous Unreflected Core	87
3.5.2	Homogeneous Core with Reflector.....	88
3.5.3	Heterogeneous Core	89
3.5.4	Effect of Control Rods	89
3.6	Energy Generation Rate within a Fuel Pin.....	92
3.6.1	Fuel Pins of Thermal Reactors.....	92
3.6.2	Fuel Pins of Fast Reactors.....	92
3.7	Energy Deposition Rate within the Moderator	93
3.8	Energy Deposition in the Structure	94
3.8.1	γ -Ray Absorption	94
3.8.2	Neutron Slowing Down	97
3.9	Decay Energy during Operation and Post Shutdown.....	100
3.9.1	Fission Power after Reactivity Insertion	101
3.9.2	Power from Fission Product Decay	102
3.9.3	ANS Standard Decay Power	105
3.9.3.1	UO ₂ in Light Water Reactors	105
3.9.3.2	Alternative Fuels in Light Water and Fast Reactors	112
3.10	Stored Energy Sources	113
3.10.1	The Zircaloy–Water Reaction	118
3.10.2	The Sodium–Water Reaction	119
3.10.3	The Sodium–Carbon Dioxide Reaction	119
3.10.4	The Corium–Concrete Interaction	120
	Problems.....	124
	References	127
Chapter 4	Transport Equations for Single-Phase Flow.....	129
4.1	Introduction	129
4.1.1	Equation Forms	129
4.1.2	Intensive and Extensive Properties.....	131

4.2	Mathematical Relations	131
4.2.1	Time and Spatial Derivative	131
4.2.2	Gauss’s Divergence Theorem	134
4.2.3	Leibnitz’s Rules	134
4.3	Lumped Parameter Integral Approach	137
4.3.1	Control Mass Formulation	137
4.3.1.1	Mass	137
4.3.1.2	Momentum	138
4.3.1.3	Energy	138
4.3.1.4	Entropy	140
4.3.2	Control Volume Formulation	142
4.3.2.1	Mass	142
4.3.2.2	Momentum	142
4.3.2.3	Energy	143
4.3.2.4	Entropy	145
4.4	Distributed Parameter Integral Approach	148
4.5	Differential Conservation Equations	151
4.5.1	Conservation of Mass	153
4.5.2	Conservation of Momentum	155
4.5.3	Conservation of Energy	161
4.5.3.1	Stagnation Internal Energy Equation	161
4.5.3.2	Stagnation Enthalpy Equation	164
4.5.3.3	Kinetic Energy Equation	165
4.5.3.4	Thermodynamic Energy Equations	166
4.5.3.5	Special Forms	167
4.5.4	Summary of Equations	169
4.6	Turbulent Flow	169
	Problems	177
	References	181

Chapter 5 Transport Equations for Two-Phase Flow 183

5.1	Introduction	183
5.1.1	Macroscopic versus Microscopic Information	183
5.1.2	Multicomponent versus Multiphase Systems	184
5.1.3	Mixture versus Multifluid Models	184
5.2	Averaging Operators for Two-Phase Flow	185
5.2.1	Phase Density Function	187
5.2.2	Volume-Averaging Operators	187
5.2.3	Area-Averaging Operators	187
5.2.4	Local Time-Averaging Operators	187
5.2.5	Commutativity of Space- and Time-Averaging Operations	188
5.3	Volume-Averaged Properties	188
5.3.1	Void Fraction	188

	5.3.1.1	Instantaneous Space-Averaged Void Fraction	188
	5.3.1.2	Local Time-Averaged Void Fraction	189
	5.3.1.3	Space- and Time-Averaged Void Fraction	189
	5.3.2	Volumetric Phase Averaging	190
	5.3.2.1	Instantaneous Volumetric Phase Averaging.....	190
	5.3.2.2	Time Averaging of Volume-Averaged Quantities.....	190
	5.3.3	Static Quality.....	191
	5.3.4	Mixture Density	192
5.4		Area-Averaged Properties	192
	5.4.1	Area-Averaged Phase Fraction.....	192
	5.4.2	Flow Quality.....	197
	5.4.3	Mass Fluxes.....	198
	5.4.4	Volumetric Fluxes and Flow Rates	199
	5.4.5	Velocity (Slip) Ratio	200
	5.4.6	Mixture Density Over an Area.....	200
	5.4.7	Volumetric Flow Ratio	201
	5.4.8	Flow Thermodynamic Quality.....	201
	5.4.9	Summary of Useful Relations for One-Dimensional Flow	201
5.5		Mixture Equations for One-Dimensional Flow.....	203
	5.5.1	Mass Continuity Equation.....	204
	5.5.2	Momentum Equation.....	205
	5.5.3	Energy Equation.....	206
5.6		Control-Volume Integral Transport Equations	207
	5.6.1	Mass Balance.....	208
	5.6.1.1	Mass Balance for Volume V_k	208
	5.6.1.2	Mass Balance in the Entire Volume V	209
	5.6.1.3	Interfacial Jump Condition	210
	5.6.1.4	Simplified Form of the Mixture Equation.....	211
	5.6.2	Momentum Balance	211
	5.6.2.1	Momentum Balance for Volume V_k	211
	5.6.2.2	Momentum Balance in the Entire Volume V	213
	5.6.2.3	Interfacial Jump Condition	214
	5.6.2.4	Common Assumptions	215
	5.6.2.5	Simplified Forms of the Mixture Equation.....	215
	5.6.3	Energy Balance	218
	5.6.3.1	Energy Balance for the Volume V_k	218
	5.6.3.2	Energy Equations for the Total Volume V	221
	5.6.3.3	Jump Condition.....	222

5.7	One-Dimensional Space-Averaged Transport Equations	223
5.7.1	Mass Equations	223
5.7.2	Momentum Equations	224
5.7.3	Energy Equations	225
	Problems	228
	References	232

Chapter 6 Thermodynamics of Nuclear Energy Conversion

	Systems: Nonflow and Steady Flow: First- and Second-Law Applications	233
6.1	Introduction	233
6.2	Nonflow Process	235
6.2.1	A Fuel–Coolant Thermal Interaction	236
6.2.1.1	Step I: Coolant and Fuel Equilibration at Constant Volume	236
6.2.1.2	Step II: Coolant and Fuel Expanded as Two Independent Systems, Isentropically and Adiabatically	239
6.2.1.3	Step III: Coolant and Fuel Expanded as One System in Thermal Equilibrium, Adiabatically and Isentropically	243
6.3	Thermodynamic Analysis of Nuclear Power Plants	248
6.4	Thermodynamic Analysis of a Simplified PWR System	254
6.4.1	First Law Analysis of a Simplified PWR System	254
6.4.2	Combined First and Second Law or Availability Analysis of a Simplified PWR System	263
6.4.2.1	Turbine and Pump	264
6.4.2.2	Steam Generator and Condenser	264
6.4.2.3	Reactor Irreversibility	266
6.4.2.4	Plant Irreversibility	268
6.5	More Complex Rankine Cycles: Superheat, Reheat, Regeneration, and Moisture Separation	272
6.6	Simple Brayton Cycle	281
6.7	More Complex Brayton Cycles	285
6.8	Supercritical Carbon Dioxide Brayton Cycles	296
6.8.1	Simple S-CO ₂ Brayton Cycle	297
6.8.2	S-CO ₂ Brayton Cycle with Ideal Components and Regeneration	298
6.8.3	S-CO ₂ Recompression Brayton Cycle with Ideal Components	300
6.8.4	S-CO ₂ Recompression Brayton Cycle with Real Components and Pressure Losses	305
	Problems	308
	References	317

Chapter 7	Thermodynamics of Nuclear Energy Conversion Systems: Nonsteady Flow First Law Analysis	319
7.1	Introduction	319
7.2	Containment Pressurization Process	319
7.2.1	Analysis of Transient Conditions	321
7.2.1.1	Control Mass Approach	321
7.2.1.2	Control Volume Approach	323
7.2.2	Analysis of Final Equilibrium Pressure Conditions.....	325
7.2.2.1	Control Mass Approach.....	325
7.2.2.2	Control Volume Approach	326
7.2.2.3	Governing Equations for Determination of Final Conditions.....	327
7.2.2.4	Individual Cases	329
7.3	Response of a PWR Pressurizer to Load Changes.....	338
7.3.1	Equilibrium Single-Region Formulation.....	338
7.3.2	Analysis of Final Equilibrium Pressure Conditions.....	340
	Problems.....	347
Chapter 8	Thermal Analysis of Fuel Elements	359
8.1	Introduction	359
8.2	Heat Conduction in Fuel Elements	360
8.2.1	General Equation of Heat Conduction	360
8.2.2	Thermal Conductivity Approximations	360
8.3	Thermal Properties of UO_2 and MOX.....	366
8.3.1	Thermal Conductivity	366
8.3.1.1	Temperature Effects.....	366
8.3.1.2	Porosity (Density) Effects.....	367
8.3.1.3	Oxygen-to-Metal Atomic Ratio	371
8.3.1.4	Plutonium Content	371
8.3.1.5	Effects of Pellet Cracking.....	372
8.3.1.6	Burnup	373
8.3.2	Fission Gas Release.....	374
8.3.3	Melting Point	375
8.3.4	Specific Heat	375
8.3.5	The Rim Effect.....	377
8.4	Temperature Distribution in Plate Fuel Elements	380
8.4.1	Heat Conduction in Fuel.....	381
8.4.2	Heat Conduction in Cladding.....	382
8.4.3	Thermal Resistances	384
8.4.4	Conditions for Symmetric Temperature Distributions.....	384

- 8.5 Temperature Distribution in Cylindrical Fuel Pins 390
 - 8.5.1 General Conduction Equation for Cylindrical Geometry 390
 - 8.5.2 Solid Fuel Pellet..... 391
 - 8.5.3 Annular Fuel Pellet (Cooled Only on the Outside Surface R_{fo}) 392
 - 8.5.4 Annular Fuel Pellet (Cooled on Both Surfaces) 393
 - 8.5.5 Solid versus Annular Pellet Performance 397
 - 8.5.6 Annular Fuel Pellet (Cooled Only on the Inside Surface R_v) 399
- 8.6 Temperature Distribution in Restructured Fuel Elements 403
 - 8.6.1 Mass Balance..... 404
 - 8.6.2 Power Density Relations..... 405
 - 8.6.3 Heat Conduction in Zone 3 406
 - 8.6.4 Heat Conduction in Zone 2 407
 - 8.6.5 Heat Conduction in Zone 1 408
 - 8.6.6 Solution of the Pellet Problem..... 409
 - 8.6.7 Two-Zone Sintering..... 409
 - 8.6.8 Design Implications of Restructured Fuel..... 416
- 8.7 Thermal Resistance Between the Fuel and Coolant..... 416
 - 8.7.1 Gap Conductance Models 418
 - 8.7.1.1 As-Fabricated Gap 418
 - 8.7.1.2 Gap Closure Effects..... 420
 - 8.7.2 Cladding Corrosion: Oxide Film Buildup and Hydrogen Consequences 422
 - 8.7.3 Overall Thermal Resistance 425
- Problems..... 427
- References 435

Chapter 9 Single-Phase Fluid Mechanics 439

- 9.1 Approach to Simplified Flow Analysis 439
 - 9.1.1 Solution of the Flow Field Problem..... 439
 - 9.1.2 Possible Simplifications..... 440
- 9.2 Inviscid Flow 442
 - 9.2.1 Dynamics of Inviscid Flow 442
 - 9.2.2 Bernoulli’s Integral..... 443
 - 9.2.2.1 Time-Dependent Flow 443
 - 9.2.2.2 Steady-State Flow 450
 - 9.2.3 Compressible Inviscid Flow 454
 - 9.2.3.1 Flow in a Constant-Area Duct 454
 - 9.2.3.2 Flow through a Sudden Expansion or Contraction 456
- 9.3 Viscous Flow 456
 - 9.3.1 Viscosity Fundamentals 456

9.3.2	Viscosity Changes with Temperature and Pressure	458
9.3.3	Boundary Layer.....	459
9.3.4	Turbulence	463
9.3.5	Dimensionless Analysis	463
9.3.6	Pressure Drop in Channels.....	464
9.3.7	Summary of Pressure Changes in Inviscid/Viscid and in Compressible/Incompressible Flows	467
9.4	Laminar Flow Inside a Channel	467
9.4.1	Fully Developed Laminar Flow in a Circular Tube.....	468
9.4.2	Fully Developed Laminar Flow in Noncircular Geometries	472
9.4.3	Laminar Developing Flow Length.....	473
9.4.4	Form Losses in Laminar Flow	476
9.5	Turbulent Flow Inside a Channel.....	476
9.5.1	Turbulent Diffusivity	476
9.5.2	Turbulent Velocity Distribution.....	478
9.5.3	Turbulent Friction Factors in Adiabatic and Diabatic Flows	480
9.5.3.1	Turbulent Friction Factor: Adiabatic Flow.....	480
9.5.3.2	Turbulent Friction Factor: Diabatic Flow....	481
9.5.4	Fully Developed Turbulent Flow with Noncircular Geometries	483
9.5.5	Turbulent Developing Flow Length	484
9.5.6	Turbulent Friction Factors—Geometries for Enhanced Heat Transfer	487
9.5.6.1	Extended Surfaces	487
9.5.6.2	Twisted Tape Inserts	491
9.5.7	Turbulent Form Losses.....	495
9.6	Pressure Drop in Rod Bundles	498
9.6.1	Friction Loss along Bare Rod Bundles.....	498
9.6.1.1	Laminar Flow	498
9.6.1.2	Turbulent Flow	501
9.6.2	Pressure Loss at Fuel Pin Spacer and Support Structures.....	503
9.6.2.1	Grid Spacers	503
9.6.2.2	Wire Wrap Spacers	513
9.6.2.3	Grid versus Wire Wrap Pressure Loss.....	514
9.6.3	Pressure Loss for Cross Flow	515
9.6.3.1	Across Bare Rod Arrays	515
9.6.3.2	Across Wire-Wrapped Rod Bundles.....	518
9.6.4	Form Losses for Abrupt Area Changes.....	521
9.6.4.1	Method of Calculation	522
9.6.4.2	Loss Coefficient Values	526

Problems	528
References	532
Chapter 10 Single-Phase Heat Transfer	535
10.1 Fundamentals of Heat Transfer Analysis	535
10.1.1 Objectives of the Analysis	535
10.1.2 Approximations to the Energy Equation	535
10.1.3 Dimensional Analysis	537
10.1.4 Thermal Conductivity	538
10.1.5 Engineering Approach to Heat Transfer Analysis	539
10.2 Laminar Heat Transfer in a Pipe	543
10.2.1 Fully Developed Flow in a Circular Tube	544
10.2.2 Developed Flow in Other Geometries	548
10.2.3 Developing Laminar Flow Region	549
10.3 Turbulent Heat Transfer: Mixing Length Approach	551
10.3.1 Equations for Turbulent Flow in Circular Coordinates	551
10.3.2 Relation between ϵ_M , ϵ_H , and Mixing Lengths	555
10.3.3 Turbulent Temperature Profile	556
10.4 Turbulent Heat Transfer: Differential Approach	562
10.4.1 Basic Models	562
10.4.2 Transport Equations for the $k_t - \epsilon_t$ Model	563
10.4.3 One-Equation Model	565
10.4.4 Effect of Turbulence on the Energy Equation	565
10.4.5 Summary	565
10.5 Heat Transfer Correlations in Turbulent Flow	566
10.5.1 Nonmetallic Fluids—Smooth Heat Transfer Surfaces	566
10.5.1.1 Fully Developed Turbulent Flow	566
10.5.1.2 Entrance Region Effect	573
10.5.2 Nonmetallic Fluids: Geometries for Enhanced Heat Transfer	578
10.5.2.1 Ribbed Surfaces	578
10.5.2.2 Twisted Tape Inserts	582
10.5.3 Metallic Fluids—Smooth Heat Transfer Surfaces: Fully Developed Flow	587
10.5.3.1 Circular Tube	587
10.5.3.2 Parallel Plates	588
10.5.3.3 Concentric Annuli	588
10.5.3.4 Rod Bundles	588
Problems	593
References	600

Chapter 11	Two-Phase Flow Dynamics	603
11.1	Introduction	603
11.2	Flow Regimes	604
11.2.1	Regime Identification	604
11.2.2	Flow Regime Maps	605
11.2.2.1	Vertical Flow	607
11.2.2.2	Horizontal Flow	613
11.2.3	Flooding and Flow Reversal	615
11.3	Flow Models	619
11.4	Overview of Void Fraction and Pressure Loss Correlations.....	620
11.5	Void Fraction Correlations	620
11.5.1	The Fundamental Void Fraction- Quality-Slip Relation.....	622
11.5.2	Homogeneous Equilibrium Model	622
11.5.3	Drift Flux Model	625
11.5.4	Chexal and Lellouche Correlation.....	628
11.5.5	Premoli Correlation	632
11.5.6	Bestion Correlation.....	633
11.6	Pressure–Drop Relations	638
11.6.1	The Acceleration, Friction, and Gravity Components.....	638
11.6.2	Homogeneous Equilibrium Models	641
11.6.3	Separate Flow Models	645
11.6.3.1	Lockhart–Martinelli Correlation.....	646
11.6.3.2	Thom Correlation	649
11.6.3.3	Baroczy Correlation.....	652
11.6.3.4	Friedel Correlation.....	652
11.6.4	Two-Phase Pressure Drop	657
11.6.4.1	Pressure Drop for Inlet Quality $x = 0$	657
11.6.4.2	Pressure Drop for Nonzero Inlet Quality.....	661
11.6.5	Relative Accuracy of Various Friction Pressure Loss Models.....	661
11.6.6	Pressure Losses across Singularities.....	663
11.7	Critical Flow	665
11.7.1	Background	665
11.7.2	Single-Phase Critical Flow	666
11.7.3	Two-Phase Critical Flow	668
11.7.3.1	Thermal Equilibrium Models	669
11.7.3.2	Thermal Nonequilibrium Models	672
11.7.3.3	Practical Guidelines for Calculations	675
11.8	Two-Phase Flow Instabilities in Nuclear Systems.....	679
11.8.1	Thermal-Hydraulic Instabilities	679
11.8.1.1	Ledinegg Instabilities	680

11.8.1.2	Density Wave Oscillations.....	684
11.8.2	Thermal-Hydraulic Instabilities with Neutronic Feedback.....	686
Problems	687
References	693
Chapter 12	Pool Boiling.....	697
12.1	Introduction.....	697
12.2	Nucleation.....	697
12.2.1	Equilibrium Bubble Radius.....	697
12.2.2	Homogeneous and Heterogeneous Nucleation.....	700
12.2.3	Vapor Trapping and Retention.....	701
12.2.4	Vapor Growth from Microcavities.....	703
12.2.5	Bubble Dynamics—Growth and Detachment.....	705
12.2.6	Nucleation Summary.....	706
12.3	The Pool Boiling Curve.....	706
12.4	Heat Transfer Regimes.....	707
12.4.1	Nucleate Boiling Heat Transfer (between Points B–C of the Boiling Curve of Figure 12.8)	707
12.4.2	Transition Boiling (between Points C–D of the Boiling Curve of Figure 12.8).....	709
12.4.3	Film Boiling (between Points D–F of the Boiling Curve of Figure 12.8).....	709
12.5	Limiting Conditions on the Boiling Curve.....	717
12.5.1	Critical Heat Flux (Point C of the Boiling Curve of Figure 12.8).....	717
12.5.2	Minimum Stable Film Boiling Temperature (Point D of the Boiling Curve of Figure 12.8).....	719
12.6	Surface Effects in Pool Boiling.....	722
12.7	Condensation Heat Transfer.....	724
12.7.1	Filmwise Condensation.....	725
12.7.1.1	Condensation on a Vertical Wall.....	725
12.7.1.2	Condensation on or in a Tube.....	726
12.7.2	Dropwise Condensation.....	729
12.7.3	The Effect of Noncondensable Gases.....	729
Problems	732
References	737
Chapter 13	Flow Boiling.....	741
13.1	Introduction.....	741
13.2	Heat Transfer Regions and Void Fraction/Quality Development.....	741
13.2.1	Heat Transfer Regions.....	741
13.2.1.1	Onset of Nucleate Boiling, Z_{ONB}	746

	13.2.1.2	Net Vapor Generation, Z_{NVG}	748
	13.2.1.3	Onset of Saturated Boiling, Z_{OSB}	751
	13.2.1.4	Location of Thermal Equilibrium, Z_E	751
	13.2.1.5	Void Fraction Profile, $\alpha(z)$	751
13.3		Heat Transfer Coefficient Correlations	752
	13.3.1	Subcooled Boiling Heat Transfer	752
	13.3.1.1	Multiple Author Correlations	754
	13.3.1.2	Chen Correlation	755
	13.3.2	Bjorge, Hall, and Rohsenow Correlation	759
	13.3.3	Post-CHF Heat Transfer	763
	13.3.3.1	Both Film Boiling Regimes (Inverted and Dispersed Annular Flow)	764
	13.3.3.2	Inverted Annular Flow Film Boiling (Only)	765
	13.3.3.3	Dispersed Annular or Liquid Deficient Flow Film Boiling (Only)	766
	13.3.3.4	Transition Boiling	770
	13.3.4	Reflooding of a Core Which Has Been Uncovered	771
13.4		Critical Condition or Boiling Crisis	772
	13.4.1	Critical Condition Mechanisms and Limiting Values	773
	13.4.2	The Critical Condition Mechanisms	775
	13.4.2.1	Models for DNB	775
	13.4.2.2	Model for Dryout	776
	13.4.2.3	Variation of the Critical Condition with Key Parameters	776
	13.4.3	Correlations for the Critical Condition	777
	13.4.3.1	Correlations for Tube Geometry	779
	13.4.3.2	Correlations for Rod Bundle Geometry	787
	13.4.4	Design Margin in Critical Condition Correlation	810
	13.4.4.1	Characterization of the Critical Condition	810
	13.4.4.2	Margin to the Critical Condition	811
	13.4.4.3	Comparison of Various Correlations	811
	13.4.4.4	Design Considerations	816
		Problems	816
		References	819
Chapter 14		Single Heated Channel: Steady-State Analysis	823
	14.1	Introduction	823
	14.2	Formulation of One-Dimensional Flow Equations	823
	14.2.1	Nonuniform Velocities	823
	14.2.2	Uniform and Equal Phase Velocities	826
	14.3	Delineation of Behavior Modes	827

14.4	The LWR Cases Analyzed in Subsequent Sections	828
14.5	Steady-State Single-Phase Flow in a Heated Channel	829
14.5.1	Solution of the Energy Equation for a Single-Phase Coolant and Fuel Rod (PWR Case).....	829
14.5.1.1	Coolant Temperature	831
14.5.1.2	Cladding Temperature	832
14.5.1.3	Fuel Centerline Temperature	833
14.5.2	Solution of the Energy Equation for a Single- Phase Coolant with Roughened Cladding Surface (Gas Fast Reactor).....	835
14.5.3	Solution of the Momentum Equation to Obtain Single-Phase Pressure Drop	836
14.6	Steady-State Two-Phase Flow in a Heated Channel Under Fully Equilibrium (Thermal and Mechanical) Conditions.....	838
14.6.1	Solution of the Energy Equation for Two-Phase Flow (BWR Case with Single-Phase Entry Region)	838
14.6.2	Solution of the Momentum Equation for Fully Equilibrium Two-Phase Flow Conditions to Obtain Channel Pressure Drop (BWR Case with Single-Phase Entry Region)	844
14.6.2.1	Δp_{acc}	845
14.6.2.2	Δp_{grav}	846
14.6.2.3	Δp_{fric}	847
14.6.2.4	Δp_{form}	848
14.7	Steady-State Two-Phase Flow in a Heated Channel Under Nonequilibrium Conditions	851
14.7.1	Solution of the Energy Equation for Nonequilibrium Conditions (BWR and PWR Cases)	852
14.7.1.1	Prescribed Wall Heat Flux.....	852
14.7.1.2	Prescribed Coolant Temperature	859
14.7.2	Solution of the Momentum Equation for Channel Nonequilibrium Conditions to Obtain Pressure Drop (BWR Case)	868
	Problems	876
	References	884
	Appendix A: Selected Nomenclature	887
	Appendix B: Physical and Mathematical Constants	905
	Appendix C: Unit Systems	907
	Appendix D: Mathematical Tables	923

Appendix E: Thermodynamic Properties 931

Appendix F: Thermophysical Properties of Some Substances 953

**Appendix G: Dimensionless Groups of Fluid Mechanics
and Heat Transfer 959**

Appendix H: Multiplying Prefixes 961

Appendix I: List of Elements 963

Appendix J: Square and Hexagonal Rod Array Dimensions 965

Appendix K: Parameters for Typical BWR-5 and PWR Reactors 971

Index 977

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

	1—Reactor-Type Overview			
	2—Core and Plant Performance Measures			
	3—Fission Energy Generation and Deposition			
	Conservation Equations	Thermodynamics	Fluid Flow	Heat Transfer
Single-Phase Coolant	4—Differential and Integral Formulations		8—Fuel Pins and Assemblies 9—Single Channels	10—Single Channels
Two-Phase Coolant	5—Differential and Integral Formulations; 2 ϕ Parameter Definitions		11—Single Channels	12—Pool Boiling 13—Flow Boiling in Single Channel
Single-Phase and Two-Phase Coolants		6—Power Cycles 7—Components and Containment	14—Single-Channel Examples	

- 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 (SCO₂) 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	Approximate Fissile Content (All ²³⁵ U Except the Sodium-Cooled Reactors)
Water-cooled	Thermal				
PWR		H ₂ O	H ₂ O	UO ₂	3–5% enrichment
BWR		H ₂ O	H ₂ O	UO ₂	3–5% enrichment
PHWR (CANDU)		D ₂ O	D ₂ O	UO ₂	Natural
SGHWR ^a		D ₂ O	H ₂ O	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).

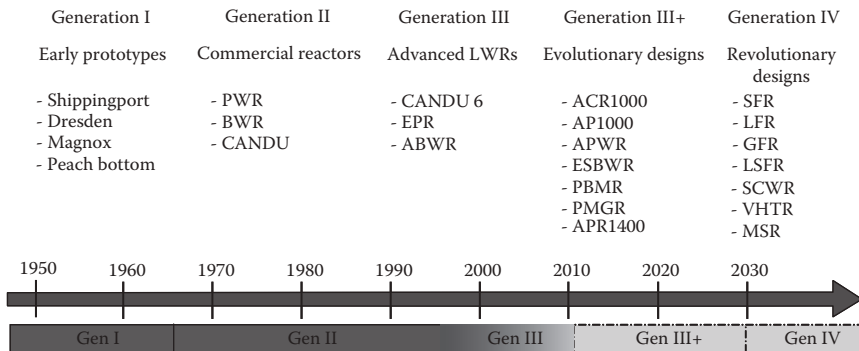


FIGURE 1.1 The evolution of nuclear power. (Adopted from U.S. Department of Energy, <http://www.gen-4.org/Technology/evolution.htm>.)

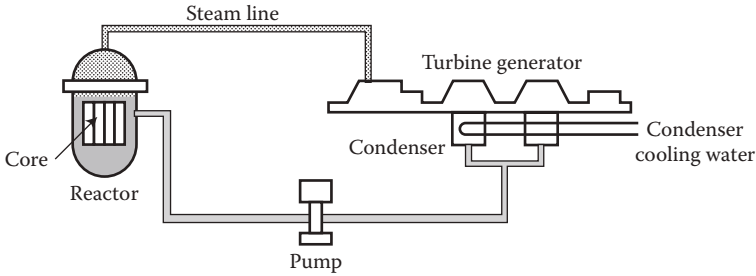


FIGURE 1.2 Direct, single-coolant Rankine cycle. (Adopted from U.S. Department of Energy.)

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

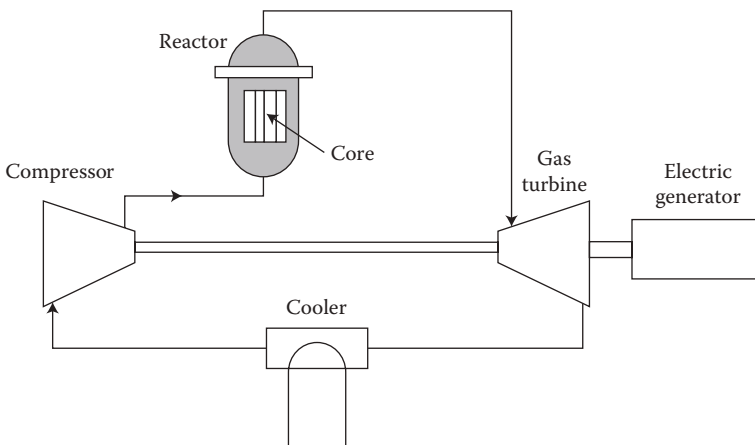


FIGURE 1.3 Direct, single-coolant Brayton cycle. (Adopted from U.S. Department of Energy.)

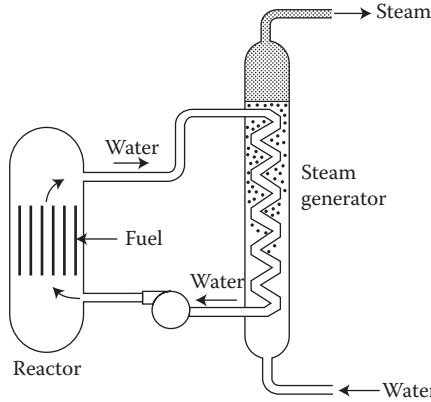


FIGURE 1.4 Two-coolant system steam cycle. (From Shultis, J. K. and Faw, R. E., *Fundamentals of Nuclear Science and Engineering*, 2nd Ed. CRC Press, Boca Raton, FL, 2008.)

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

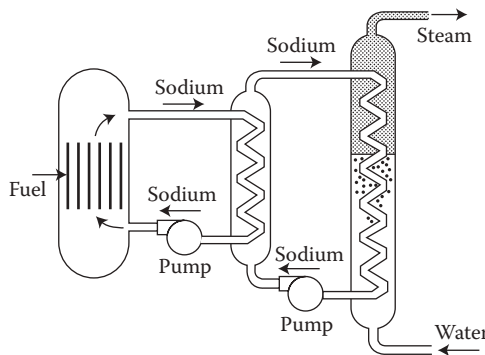


FIGURE 1.5 Three-coolant system steam cycle. (From Shultis, J. K. and Faw, R. E., *Fundamentals of Nuclear Science and Engineering*, 2nd Ed. CRC Press, Boca Raton, FL, 2008.)

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

Characteristics	BWR	PWR	PWR	PHWR	HTGR	AGR	SFBR
Manufacturer	General Electric	Westinghouse	Reference Design 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	2	3
Primary coolant	H ₂ O	H ₂ O	D ₂ O		He	CO ₂	Liq. Na
Secondary coolant	^b	H ₂ O	H ₂ O		H ₂ O	H ₂ O	Liq. Na/H ₂ O
			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 NaH ₂ O
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 Kinief, 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/RPSEG/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.

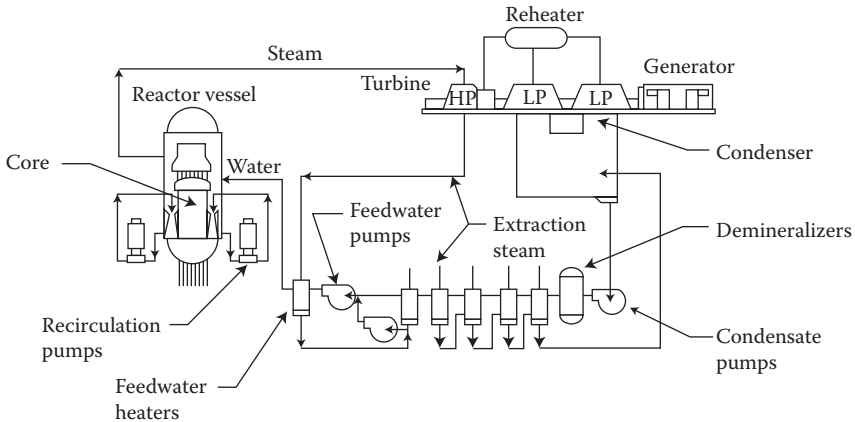


FIGURE 1.6 BWR single-loop primary coolant system. (From Shultis, J. K. and Faw, R. E., *Fundamentals of Nuclear Science and Engineering*, 2nd Ed. CRC Press, Boca Raton, FL, 2008.)

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

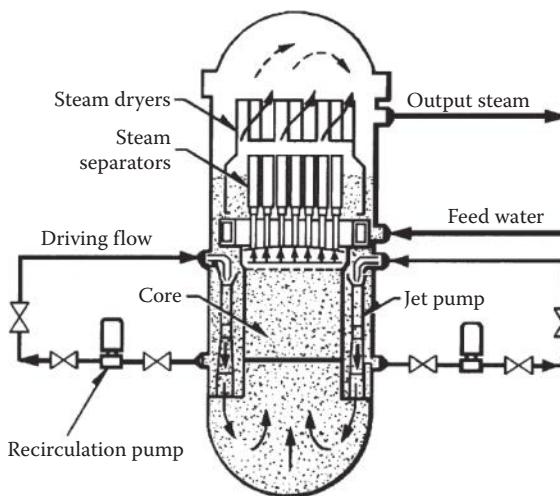


FIGURE 1.7 Steam and recirculation water flow paths in the Generation II BWR. (From Shultis, J. K. and Faw, R. E., *Fundamentals of Nuclear Science and Engineering*, 2nd Ed. CRC Press, Boca Raton, FL, 2008.)

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

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

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

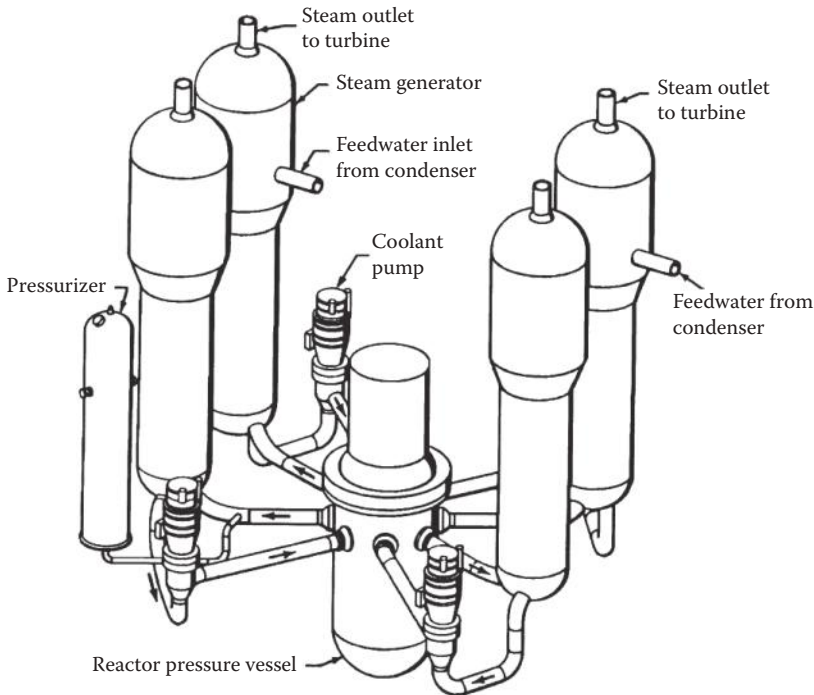


FIGURE 1.8 Arrangement of the primary system for a Generation II PWR. (From Shultis, J. K. and Faw, R. E., *Fundamentals of Nuclear Science and Engineering*, 2nd Ed. CRC Press, Boca Raton, FL, 2008.)

* 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

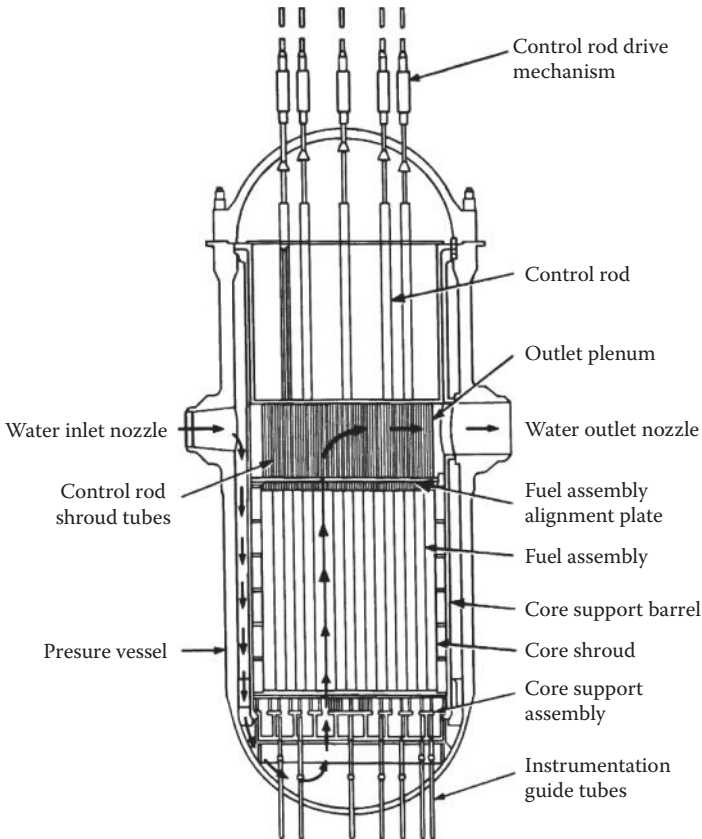


FIGURE 1.9 Flow path through a PWR reactor vessel. (From Shultis, J. K. and Faw, R. E., *Fundamentals of Nuclear Science and Engineering*, 2nd Ed. CRC Press, Boca Raton, FL, 2008.)

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

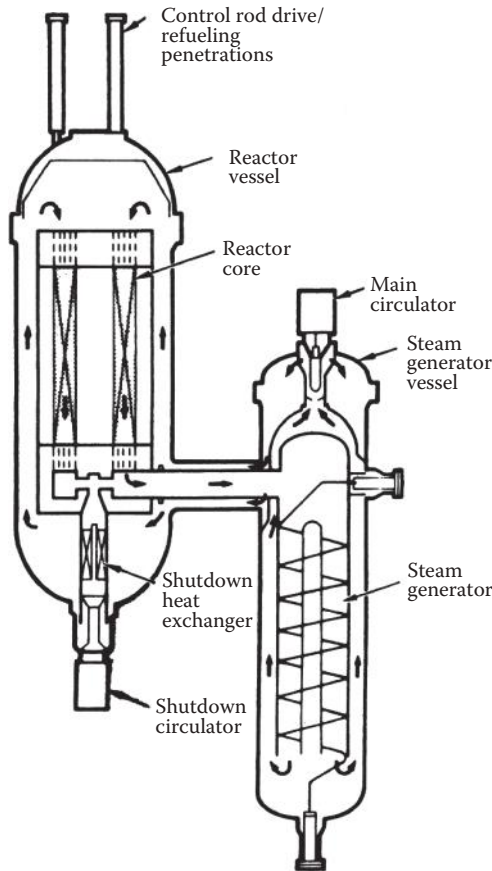


FIGURE 1.10 Modular HTGR primary coolant flow path. (Courtesy of U.S. Department of Energy.)

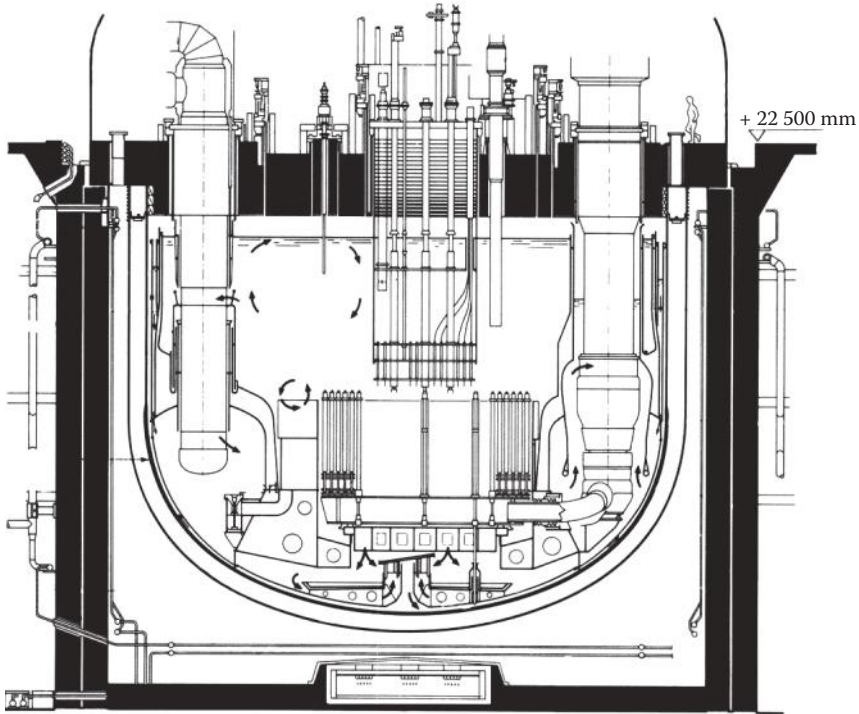


FIGURE 1.11 Primary system sodium flow path in the Superphenix reactor. (Courtesy of Électricité de France.)

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
Manufacturer	General Electric	Westinghouse	Reference Design			
System (reactor station)	BWR/5 (NMP2)	(Seabrook)	Atomic Energy of Canada, Ltd.	General Atomic	National Nuclear Corp.	Novatome
Moderator	H ₂ O	H ₂ O	CANDU-600	(Fulton)	(Heysham 2)	(Superphenix 1)
Neutron energy	Thermal	Thermal	D ₂ O	Graphite	Graphite	None
Fuel production	Converter	Converter	Thermal Converter	Thermal Converter	Thermal Converter	Fast Breeder
Geometry	Cylindrical pellet	Cylindrical pellet	Fuel^b	Microspheres ^c	Cylindrical pellet	Cylindrical pellet
Dimensions (mm)	9.60D × 10.0L	8.192D × 9.8L	Cylindrical pellet	400–800 μm D	14.51D × 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
Geometry	Pellet stack in clad tube	Pellet stack in clad tube	Fuel Rods Pellet stack in clad tube	Cylindrical fuel compacts	Pellet stack in clad tube	Pellet stack in clad tube

continued