NUCLEAR ENGINEERING AND TECHNOLOGY FOR THE 21ST CENTURY – MONOGRAPH SERIES

Nuclear Reactor Thermal-Hydraulics: Past, Present and Future

Pradip Saha
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Series Editors’ Preface

*Nuclear Engineering and Technology for the 21st Century—Monographs Series*

Nuclear engineering and technology play a vital role in achieving low carbon emission goals worldwide, while providing reliable, baseload power to the world economy. Presently over 12 percent of the world’s energy needs are satisfied by nuclear power—with 30 countries operating 436 nuclear power plants and 3 countries (France, Slovakia, and Belgium) using nuclear power to provide over half their power needs (source: Nuclear Energy Institute: http://www.nei.org).

The country with the largest number of operational nuclear power plants (the United States) has 102 plants and uses nuclear power to provide over 19 percent of its needs. Concurrently, the advanced nuclear power plant designs are the basis for extensive, ongoing research and development efforts in many countries with the promise of enhanced sustainability, safety, and proliferation-free power-sources with ever-higher operational efficiencies and capacity factors. Consequently, there are many fruitful topics of interest in the nuclear engineering field to be addressed in this exciting monograph series.

The *Nuclear Engineering and Technology for the 21st Century* monograph series provides current and future engineers, researchers, technicians and other professionals and practitioners with practical, concise but key information concerning the nuclear technologies from areas of medical applications, mining, processing and manufacturing, environmental monitoring to safe and energy-efficient plant operation and electricity generation. Each monograph should provide a well rounded and definitive state-of-the-art review of its subject, with a focus on applied research and development, and best industry practices, processes and related technological applications. The series is envisaged as a collection of 80 to 100 pages monograph publications which can stand as the most authoritative source of information on current state of a topic, application or discipline. Core topics include, but are not limited to:

- best practices in power plant operation
- nuclear science and technology in medicine,
- irradiation technologies and applications,
Nuclear Reactor

- fuel cycle processes, engineering and technologies,
- nuclear reactor thermal hydraulics and/or neutronics
- materials for current and advance power generation
- nuclear safety and environmental impact
- next generation of nuclear power plants
- radiation in our environment
- radioecology, radiobiology, radiation chemistry

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Contents

Series Editors’ Preface iii
Abstract vii
Acknowledgements ix

1. Introduction 1
   1.1 Nuclear fission and heat generation 3
   1.2 Reactor classification 7
   1.3 Role of thermal hydraulics 13
      1.3.1 Desired features of a reactor coolant 13
      1.3.2 Reactor thermal hydraulics during normal operation 15
      1.3.3 Reactor thermal hydraulics during abnormal
            or accident conditions 16
   1.4 Scope of this monograph 18

2. Steady state reactor thermal-hydraulics 19
   2.1 Single-phase reactor core flow 19
   2.2 Two-phase boiling core flow 20
      2.2.1 Void-quality relationship 23
      2.2.2 Pressure drop 24
   2.3 Departure from nucleate boiling and boiling transition 25
      2.3.1 Lower-quality CHF 26
      2.3.2 Higher-quality CHF or boiling transition 27
      2.3.3 Enhancement of CHF or critical power 29
      2.3.4 Margin to critical condition 31
   2.4 Subchannel analysis 34
      2.4.1 Traditional approach 34
      2.4.2 Two-fluid three-field approach 43

3. Nuclear reactor safety systems 55
   3.1 Active safety systems 55
      3.1.1 Generation II PWRs 56
      3.1.2 Generation II BWRs 58
      3.1.3 Generation II PHWRs 60
      3.1.4 Generation III PWRs and BWR 61
   3.2 Passive safety systems 68
      3.2.1 AP1000 69
      3.2.2 ESBWR 72
      3.2.3 Light water cooled and advanced SMRs 75

4. Nuclear reactor safety analysis 81
   4.1 Pre-1975 methodology (Conservative analysis) 82
   4.2 Development of best-estimate methodology (1975–1990) 85
   4.3 Consolidation of best-estimate methodology (1991–present) 97
      4.3.1 Application for performance enhancement and economics 97
      4.3.2 Application to new reactors 101
Nuclear Reactor

4.4 Methodology for the future (2010–)
  4.4.1 Interfacial area transport equation
  4.4.2 Computational fluid dynamics
  4.4.3 Consortium for advanced simulation of LWRs (CASL)

4.5 Analysis of generation IV reactors

5. Summary and conclusions

References

Author biography

Index
Abstract

This monograph summarizes the major developments on nuclear reactor thermal-hydraulics over the last fifty years, primarily for the water-cooled reactors, and provides a direction for the future thermal-hydraulic developments for water-cooled, including small modular reactors or SMR, and Generation IV reactors. This includes discussion on the steady-state reactor thermal hydraulics including subchannel analysis, evolution of emergency core cooling systems (ECCS) from active to fully passive systems to remove the decay heat, and development and consolidation of the best-estimate safety analysis methodology. With substantial increase in computing power, the computational fluid dynamics (CFD) tools for single-phase and multi-phase flows are being used more these days to address some of the important reactor thermal-hydraulics phenomena which could not be analyzed earlier using the traditional one-dimensional or coarse three-dimensional analysis tools. Development of multi-physics methodology encompassing neutronics, thermal-hydraulics, thermal-mechanical and coolant chemistry has also started.
Acknowledgements

The author thanks the management of GE Hitachi Nuclear Energy (GEH) for granting permission for preparation of this monograph and allowing its publication within the ASME “Nuclear Engineering and Technology for the 21st Century” Concise Monograph Series. Review of the manuscript and valuable comments provided by the author’s colleagues at GEH, Brett Doories and Glen Watford, are gratefully acknowledged. Comments and suggestions of the external reviewers are also appreciated.
1. Introduction

Nuclear power plants, which produce electricity with almost zero greenhouse gas or carbon emissions at stable and competitive costs, are operating in more than thirty countries throughout the world. According to International Atomic Energy Agency (IAEA) Reference Data Series No. 1 2015 Edition [1], in 2014, 438 nuclear power reactors with the total installed capacity of 376 GWe produced about 2410 TWh of electricity representing 11.1% of the total electricity produced in the world from all energy sources. Some countries in Europe (e.g., France, Slovakia, Hungary, Ukraine, Belgium, and Sweden) produced more than 40% of their electricity from nuclear. The United States of America (USA) produced about 800 TWh of electricity, representing 19.5% of total U.S. electricity production, from its 99 nuclear reactors. This represented over 60% of electricity produced from low carbon-emission sources including hydro, solar and wind [2]. Percentage of electricity from nuclear is also increasing in Asia, particularly in China, Korea and India. Thus nuclear power constitutes an element of the solution to global warming and a means of delivering electricity to both developed and emerging countries.

Figure 1-1 (reproduced from Ref. 1 with permission by the IAEA) shows the nuclear share of electricity in various countries in 2014. Generation of electricity from nuclear reactors started in 1950’s with rapid growth in 1960’s to 1980’s. After the Three Mile Island Unit 2 accident in 1979 [3–6] and Chernobyl Unit 4 accident in 1986 [7, 8], the growth of nuclear power in developed countries slowed down; however, the growth increased in large Asian countries, namely, China and India. It is estimated [1] that the world electricity generation by nuclear will slowly increase in the future and its share will be around 9–11% in 2030. Thus nuclear power will remain a significant contributor to the world’s electricity generation as more countries pledge to reduce the greenhouse gas emissions to combat the climate change. In the USA, nuclear’s share of electricity generation in 2040 is estimated to be around 16% [9] as shown in Figure 1-2.

There are many types of nuclear power reactors operating in the world today to generate electricity. However, the basic principles of these operating reactors are the same. In simple words, heat is generated in the reactor core by sustaining a fission chain reaction in nuclear fuel.
Figure 1-1  Nuclear share of total electricity generation in 2014 (reproduced from Ref. 1 with permission by the IAEA).
That heat is extracted by a coolant flowing through the reactor core to produce steam and drive a steam turbine to generate electricity.

1.1 Nuclear fission and heat generation

Nuclear fission reactions can occur when a neutron strikes the nucleus of a large fissile atom, typically the fissile isotopes uranium-235 or plutonium-239, causing that nucleus to split or fission. The result of a fission reaction is typically two fission fragments or smaller nuclei, two or more fast-moving high-energy neutrons, and significant heat. The new neutrons produced by a fission reaction initiate new fission reactions, resulting in a sustained fission chain reaction. This is depicted in Figure 1-3.

Nuclear reactors typically fall into one of two types – Thermal or Fast reactors – based on the neutron spectrum or neutron energies at which the fission reactions occur:

- Thermal reactors optimize the fission reaction rate in their fuel by slowing down, or moderating, the high-energy fast neutrons that are produced as a result of fission reactions. This “moderation” of the neutrons is achieved by using a moderator material, such as graphite or heavy water, which slows down the neutrons and allows the fission reaction to occur at a controlled rate. Thermal reactors are the most common type of reactor and are used in most commercial power plants.

- Fast reactors, on the other hand, do not moderate the fission neutrons. Instead, they rely on the high-energy neutrons to directly initiate fission reactions. This allows for a more efficient conversion of the fission energy into electricity, but the reactor must be carefully designed to prevent the formation of supercritical conditions that could lead to a runaway reaction.

Figure 1-2  Forecast of U.S. electricity generation by fuel type (reproduced from Ref. 9 with permission by the USGAO).
the fast neutrons increases the likelihood that a “new” neutron will initiate fission to sustain the chain reaction. Most of the operating nuclear power reactors today are “thermal” reactors and the process of neutron moderation is depicted in Figure 1-4.

- Fast reactors do not moderate the fission neutrons, instead leaving them fast at high energy. Fast neutrons allow these reactors to be more effective than thermal reactors at creating or breeding new fuel through neutron absorption in uranium-238, creating more fissile material plutonium-239.

For both thermal and fast reactors, the fission reaction occurs in the central region of a reactor called the reactor core, which typically contains the following components:

- **Nuclear fuel**: Nuclear reactors need fissile isotopes, such as uranium-235 or plutonium-239 to sustain a chain reaction and generate heat. Most of the commercial power reactors are light water reactors (LWRs), which need slightly enriched (<5%) uranium-235 as their fissile fuel, the rest of the fuel being non-fissile uranium-238, some of which is converted to fissile
Introduction 5

plutonium-239 during reactor operation. Some reactors, such as the pressurized heavy water reactor (PHWR), can use natural uranium containing around 0.7% of fissile uranium-235, the rest being non-fissile uranium-238, as fuel and some reactors can utilize thorium-232 to produce fissile uranium-233. Nuclear fuel used in most of the operating reactors is in the uranium oxide form, although metallic or other forms are being considered for some reactor designs.

- **Fuel Cladding or Sheath:** In order to hold and contain the nuclear fuel, which is usually in pellet form, and the fission products or fragments that are created during reactor operation, most reactors use fuel cladding or sheath to encase the fuel pellets. The cladding may be a zirconium alloy, as in LWRs, or stainless steel or other materials designed to withstand the challenging conditions of a nuclear reactor.

- **Fuel Assembly:** Fuel pellets are usually stacked inside long fuel rods encapsulated by the cladding mentioned above. Several fuel rods are grouped together in square, hexagonal or circular arrays to form a fuel bundle or assembly. Spacers and other hardware hold the fuel bundle together. In some reactors, the fuel bundles are contained inside channel boxes to form the fuel assemblies.

Figure 1-4  Neutron slowdown by a moderator.
for proper coolant flow around the fuel rods. A typical reactor core contains hundreds of such fuel assemblies.

- **Moderator:** Only the thermal reactors use a moderator material to slow down the fission neutrons in order to sustain the fission reaction. In LWRs, the moderator is water, in PHWR, heavy water is the moderator, and in some advanced reactors such as high-temperature gas cooled reactor, graphite is used as the moderator. Fast reactors do not use a moderator.

- **Coolant:** A coolant, typically ordinary or light water in LWR, heavy water in PHWR, a gas (typically helium or carbon dioxide) in gas-cooled reactor, or a liquid metal (sodium) in sodium-cooled reactor, flows through the reactor core to remove heat generated by the fission reaction. In most of the reactor systems, this heat produces steam in a steam generator outside the reactor vessel containing the reactor core. However, in a Boiling Water Reactor (BWR), steam is produced as the water (coolant) passes through the reactor core inside the reactor vessel. In LWR, both Pressurized Water Reactor (PWR) and BWR, the same water is used as a moderator and a coolant. The coolant keeps the reactor core from overheating and avoids fuel cladding failure.

- **Reaction control:** Reactors can use different techniques to maintain the fission chain reaction at appropriate rates. For example, control rods, incorporating materials like boron that absorb neutrons to reduce or stop the nuclear chain reaction, may be inserted into reactor cores to provide control over the reaction rate. Neutron-absorbing liquid, such as boric acid, may be introduced into the coolant to achieve a similar effect.

Reactors also have components outside of the reactor core or vessel that help transfer heat from the core to generate steam and create electricity. Certain types of LWRs – pressurized water reactors (PWRs) – and PHWRs also contain a pressurizer to help maintain the desired coolant pressure. The reactor coolant is circulated through the core and is used to generate steam, which then powers a turbine to generate electricity. Schematic of a PWR system is shown in Figure 1-5 (reproduced from USNRC web site [10]). Typically, the primary coolant loop of a PWR operates at 15.5 MPa (~2250 psia) pressure with reactor core inlet and outlet temperatures of around 285°C and 325°C, respectively.
Note that the core outlet bulk temperature is less than the saturation temperature of 345°C, but some steam is produced at the upper section of the core because of “subcooled boiling” to be discussed later. The saturated steam in the steam generator is produced at around 7 MPa (~1015 psia) pressure and 285°C which flows to the turbine.

A peculiarity of a nuclear reactor is that it continues to generate some heat called “decay heat” even after the nuclear chain reaction is stopped by insertion of control rods or liquid boron or both. This is due to radioactive decay of fission products or fragments sealed inside the fuel cladding or sheath. Removal of this decay heat is of utmost importance even after the reactor is “shutdown”. For this reason, all nuclear reactors are equipped with Emergency Core Cooling System (ECCS) to remove this decay heat, which can be a significant percentage (1.5–0.5%) of the rated reactor power even after a few hours or days from reactor shutdown. Failure to remove this decay heat caused the core meltdown accidents in Three Mile Island Unit 2 reactor in 1979 [3–6] and more recently in Fukushima Daiichi reactors in Japan in 2011 [11–13]. More will be discussed on the topic of decay heat removal during transient and accident conditions in this monograph.

1.2 Reactor classification

The most common nuclear power reactors operating today are PWR and BWR – both are light water cooled and moderated reactors and fall under the broad category of light water reactor (LWR). This is followed
by the Pressurized Heavy Water Reactor (PHWR) or CANada Deuterium Uranium (CANDU) type reactor, which is moderated and cooled by different streams of heavy water, unlike PWR or BWR where the same light water stream is used for both moderation and cooling. Several gas-cooled, graphite-moderated reactors (GCR) in UK and light-water-cooled, graphite-moderated reactors (LWGR) in Russia are other types of operating reactors as shown in Table 1-1 (adapted from [14]). There were only two fast spectrum breeder reactors (FBR) operating at the end of 2014. However, a third FBR, BN-800, started operation in Russia in 2016.

Most of the power reactors under construction today are of PWR type, and a vast majority of them are being built in China. A few BWR, PHWR and FBR are also at various stages of construction [14].

Figure 1-6 (reproduced from [15]) shows various generations of nuclear power reactors with timeline.

Most of the reactors operating today are of Generation II, i.e., PWR, BWR or PHWR (or CANDU type). Schematic of a PWR system is shown in Figure 1-5; schematics of a BWR [16] and a PHWR [17] are shown in Figures 1-7 and 1-8, respectively. Note that in a BWR,

<table>
<thead>
<tr>
<th>Type</th>
<th>Number</th>
<th>Net Capacity (GWe)</th>
</tr>
</thead>
<tbody>
<tr>
<td>PWR (Pressurized Light-Water-Moderated and Cooled Reactor)</td>
<td>277</td>
<td>257.2</td>
</tr>
<tr>
<td>BWR (Boiling Light-Water-Cooled and Moderated Reactor)</td>
<td>80</td>
<td>75.5</td>
</tr>
<tr>
<td>PHWR (Pressurized Heavy-Water-Moderated and Cooled Reactor)</td>
<td>49</td>
<td>24.6</td>
</tr>
<tr>
<td>GCR (Gas-Cooled, Graphite-Moderated Reactor)</td>
<td>15</td>
<td>8.2</td>
</tr>
<tr>
<td>LWGR (Light-Water-Cooled, Graphite-Moderated Reactor)</td>
<td>15</td>
<td>10.2</td>
</tr>
<tr>
<td>FBR (Fast Breeder Reactor)</td>
<td>2</td>
<td>0.6</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td>438</td>
<td><strong>376.2</strong></td>
</tr>
</tbody>
</table>

*Total includes six reactors of 5.032 GWe in Taiwan.
saturated steam is generated at around 7.2 MPa (~1045 psia) pressure and 287°C temperature inside the reactor core and there is no separate steam generator like in a PWR (as seen in Figure 1-5) or PHWR (as seen in Figure 1-8). Thus the BWR is a direct power cycle. In PHWR (Figure 1-8), natural uranium containing about 0.7% uranium-235 is used as the nuclear fuel. Heavy water flowing through a large number of horizontal pressure tubes at around 11.2 MPa (~1625 psia) pressure is used as coolant whereas cooler, lower-pressure heavy water contained
in a large calandria vessel encompassing all the pressure tubes is used as moderator. Heavy water inlet and outlet temperatures in the pressure tubes are typically around 260°C and 305°C, respectively. Similar to a PWR, the outlet bulk coolant temperature in a PHWR is below saturation, but “subcooled boiling” causes some steam generation near the exit of the pressure tube. Like a PWR, PHWR also uses separate steam generators, although at a lower pressure of around 5 MPa, where steam is generated for the turbines to produce electricity. In most of the water reactors (BWR, PWR and PHWR), the steam generated is at saturation; however, in once-through steam generators employed in Babcock & Wilcox (B&W) PWR plants, namely, three units at Oconee, Three Mile Island Unit 1, Davis-Besse Unit 1, and others [18] superheated steam is produced which improves the overall thermodynamic efficiency of the plant. Steam generators in gas-cooled and sodium-cooled reactors also produce superheated steam for increased plant thermodynamic efficiency.

In Generation II reactor systems, the decay heat is removed by circulating or injecting coolant into the reactor core using active systems such as motor-driven pumps. This is the ECCS system mentioned earlier. After the TMI-2 accident in 1979, design of Generation III/III+ reactors started to improve safety and economics of nuclear power reactors. Some designs increased the reliability and redundancy of the active

**Figure 1-8** Schematic of a PHWR or CANDU system (reproduced from Ref. 17 with permission).
safety systems as in ABWR [19, 20] and in EPR [20], which may be categorized as Generation III reactors. The Generation III+ designs, in general, employ passive safety systems for enhanced safety, economics and simplicity, the examples are the AP1000 [20, 21], ESBWR [20, 22, 23], and ACR-1000 [24] designs. A few designs such as APR1400 [20] and AES-92 [25] use hybrid active and passive safety systems, and thus are sometimes referred to as Generation III+ designs. An excellent review of Generation-III/III+ reactors can be found in [25]. More will be discussed about the thermal hydraulics and safety of these reactor designs later in this monograph.

Efforts to develop Generation IV reactors started in 2001 with the formation of Generation IV International Forum (GIF) by nine countries. Today 13 countries (Argentina, Brazil, Canada, China, Euratom, France, Japan, the Republic of Korea, the Russian Federation, South Africa, Switzerland, the United Kingdom and the United States) are the members of GIF [15]. The objective of GIF is to develop the next or “fourth” generation of nuclear reactors with the following four goals:

- Sustainability
- Safety and reliability
- Economic competitiveness
- Proliferation resistance and physical protection.

In 2002, GIF selected the following six systems from nearly 100 concepts as Generation IV technologies:

- Gas-cooled fast reactor (GFR)
- Lead-cooled fast reactor (LFR)
- Molten salt reactor (MSR)
- Sodium-cooled fast reactor (SFR)
- Supercritical-water-cooled reactor (SCWR)
- Very-high-temperature reactor (VHTR).

Since the “starting point” and R&D funding of the different Generation IV systems were not equivalent, the degree of technical progress over the past decade has not been uniform for all systems. A number of participating countries devoted significant resources to the development of the SFR and VHTR, for example, in large part due to the considerable
historical effort associated with these technologies. Limited resources have so far been dedicated to the other systems.

Most of the advanced or Generation IV reactors can be classified as small modular reactors (SMRs) with the net electricity generation capacity of around 300 MWe or less. This is understandable since new technologies are often started at a smaller scale. However, in recent years, some reactor vendors are developing light water SMRs or LWSMR in contrast to the larger Generation III/III+ LWRs discussed above. This stems from the belief that LWSMRs can provide increased flexibility and options at lower costs. Light water SMRs, as their name implies, have two important design features, each of which leads to certain reactor characteristics. First, they are designed to be small compared to large LWRs – both in physical size and in power output. Second, they are designed modularly, i.e., components can be manufactured in a factory environment, where standardization and improvement of production techniques can help reduce costs, and the components can then be assembled on-site. Furthermore, light water SMRs are generally designed around the idea that there are trade-offs between the economy of scale that large LWRs provide (by lowering the cost per MWe with very large

![Diagram of a PWR type SMR](image)

**Figure 1-9** Schematic of a PWR type SMR (reproduced from Ref. 9 with permission by the USGAO).
facilities) and the potential economy of mass production that modular construction of larger numbers of smaller-sized reactors can provide. Figure 1-9 shows an example of a PWR type SMR [9] where the reactor core flow is achieved through natural, temperature driven flow and steam generation occurs within the reactor pressure vessel in contrast to a large PWR shown in Figure 1-5.

1.3 Role of thermal hydraulics

The reactor core must be cooled during normal operating condition as well as abnormal or even accident conditions. This is to transfer the heat generated by the fission reaction and decay heat to the coolant and to avoid overheating or failure of the fuel cladding, which is the first barrier for containing the highly radioactive fission products inside the nuclear power plant. Regulations may vary from country to country; however, the underlying requirement everywhere is the assurance of public health and safety at all times of reactor operation including abnormal or accident conditions. All reactors are therefore equipped with shutdown systems and ECCS to remove the decay heat.

1.3.1 Desired features of a reactor coolant

Hewitt and Collier [26] suggested the following desired features of a reactor coolant:

1. High specific heat ($c_p$) which helps reduce the mass flow rate of coolant ($W$) and the coolant temperature rise ($T_{out} - T_{in}$) in the reactor core to remove the generated heat ($Q$) as per the heat balance for a single-phase liquid or gas

$$Q = Wc_p(T_{out} - T_{in}) \quad (1-1)$$

Water with relatively high specific heat is an excellent coolant in that sense.

2. High rate of heat transfer which helps to keep the cladding temperature close to the coolant temperature. Fluid with high thermal conductivity such as liquid metal, e.g., sodium, is a great choice from that viewpoint, whereas gaseous coolant such as helium or carbon dioxide with low thermal conductivity is a poor choice.

3. Ease of pumping – fluid with low viscosity requires less pumping power for circulation of the coolant through the reactor and is
thus desirable. Hewitt and Collier [26] suggested a Figure of Merit $F$ proportional to the heat transfer rate from the reactor core to the required pumping power for coolant flow. Assuming the coolant flow to be 'turbulent' (which is true most of the time), one can express $F$ as

$$ F = \frac{c_p}{\mu^{0.2}} \rho^2 $$

(1-2)

where $\rho$ is the fluid density and $\mu$ is the dynamic viscosity. Table 1-2 shows typical values of the Figure of Merits for various reactor coolants relative to liquid sodium. Light and heavy water are superior to sodium, whereas gases are inferior to sodium from the viewpoint of Figure of Merit.

4. **Good nuclear properties** – coolants should have low neutron absorption so that a sufficient number of neutrons are available for the fission reaction. Heavy water and gases are good from that viewpoint.

5. **Compatibility** – the coolant should not corrode the reactor core and the circulating system, even under the condition of high temperature and high radiation flux expected in the reactor core.

6. **Well-defined phase state** – liquid coolants remain as liquid during both normal and accident conditions. Liquid metal such as sodium or molten salt with high boiling point is suitable from that viewpoint and can be operated at low or near atmospheric pressure. Light and heavy water cooled reactors, on the other hand, are more expensive.

### Table 1-2 Relative figure of merit of various reactor coolants.

<table>
<thead>
<tr>
<th>Fluid</th>
<th>Pressure (MPa)</th>
<th>Temp. (deg-C)</th>
<th>Density (kg/m³)</th>
<th>Sp. Heat (kJ/kg-C)</th>
<th>Viscosity (kg/m-s)</th>
<th>Relative Figure of Merit</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sodium</td>
<td>0.1</td>
<td>550</td>
<td>817</td>
<td>1.26</td>
<td>2.30E-04</td>
<td>1.0</td>
</tr>
<tr>
<td>Light Water</td>
<td>15.5</td>
<td>270</td>
<td>782</td>
<td>4.93</td>
<td>1.00E-04</td>
<td>49.3</td>
</tr>
<tr>
<td>Heavy Water</td>
<td>7.0</td>
<td>270</td>
<td>770</td>
<td>5.09</td>
<td>0.98E-04</td>
<td>52.5</td>
</tr>
<tr>
<td>Helium</td>
<td>6.0</td>
<td>270</td>
<td>817</td>
<td>5.27</td>
<td>1.13E-04</td>
<td>63.4</td>
</tr>
<tr>
<td>Carbon dioxide</td>
<td>4.0</td>
<td>450</td>
<td>3.08</td>
<td>5.2</td>
<td>0.36E-04</td>
<td>1.1E-03</td>
</tr>
<tr>
<td>Carbon dioxide</td>
<td>4.0</td>
<td>450</td>
<td>29.5</td>
<td>1.2</td>
<td>0.30E-04</td>
<td>1.7E-03</td>
</tr>
</tbody>
</table>
hand, must be operated at high pressure to achieve higher core exit temperature for achieving reasonable plant thermodynamic efficiency (>30%).

7. Cost and availability – a reactor coolant should be readily available and not be very expensive since the inventory of coolant in a typical reactor system is quite high (hundreds of tons). Light water is a very good coolant from that viewpoint.

No single fluid meets all of the above desirable features. However, water (light as well as heavy) meets many of the above features and is used as coolant in most of the operating reactors in the world today.

1.3.2 Reactor thermal hydraulics during normal operation

Maximum fuel and cladding temperatures are the limiting thermal hydraulic criteria during normal reactor operation. For uranium oxide fuel, most prevalent in the operating reactors today, the maximum fuel temperature is around 2800°C, the melting point of the oxide [26]. For metallic fuel, being considered in some sodium-cooled fast reactors [27], the maximum fuel temperature is lower (~1075°C) because of lower melting temperature, but still higher than the sodium boiling temperature of ~883°C at atmospheric pressure.

The cladding temperature ($T_{clad}$) is determined from the fuel rod surface heat flux ($q''$), the coolant bulk temperature ($T_{bulk}$) and heat transfer coefficient ($h$)

$$T_{clad} = T_{bulk} + \frac{q''}{h}$$  \hspace{1cm} (1-3)

Physical properties of the fluid, particularly thermal conductivity and fluid velocity determines the heat transfer coefficient ($h$) for single-phase gas or liquid flow. For boiling fluid, the heat transfer coefficient is significantly higher than that for the single-phase flow. Typical values of heat transfer coefficients, surface heat fluxes in the hotter channels, clad-to-coolant temperature differences in the hotter channels and maximum cladding temperatures for various reactor systems under normal steady-state operation are shown in Table 1-3.

Note that for GCR such as advanced gas-cooled reactor (AGR) [26], the heat transfer coefficient is enhanced using fins or ribs so that a clad-to-coolant temperature difference of a few hundreds of Celsius is achieved. The maximum cladding temperatures shown in Table 1-3 are
16 Nuclear Reactor

**Table 1-3** Typical thermal parameters for various reactors under normal steady-state operation.

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Heat Transfer Coefficient (W/m²·C)</th>
<th>Heat Flux in Hotter Channels (W/m²)</th>
<th>Clad-to-Coolant Temperature Difference (°C)</th>
<th>Maximum Cladding Temperature (°C)</th>
</tr>
</thead>
<tbody>
<tr>
<td>PWR/PHWR (Single-phase water)</td>
<td>30,000</td>
<td>1.5 × 10⁶</td>
<td>50</td>
<td>350–380</td>
</tr>
<tr>
<td>BWR (Boiling water)</td>
<td>60,000</td>
<td>1.0 × 10⁶</td>
<td>16</td>
<td>300–320</td>
</tr>
<tr>
<td>SFR (Liquid sodium)</td>
<td>55,000</td>
<td>2.0 × 10⁶</td>
<td>36</td>
<td>700–750</td>
</tr>
<tr>
<td>GCR/AGR (High-pressure carbon dioxide)</td>
<td>~1,000</td>
<td>1.0 × 10⁵</td>
<td>~100</td>
<td>750–800</td>
</tr>
</tbody>
</table>

Based on the specific cladding materials (Zircaloy for PWR, PHWR and BWR, and stainless steel for AGR and SFR) used in the reactors today.

An additional thermal criterion for water cooled reactors is the avoidance of critical heat flux (CHF) or departure from nucleate boiling (DNB) in PWR, CHF in PHWR, and avoidance of film dryout or boiling transition (BT) in BWR. All of these cause significant deterioration of heat transfer coefficients and increase the cladding temperature beyond the typical values mentioned above. These phenomena will be discussed in further details later in this monograph.

1.3.3 Reactor thermal hydraulics during abnormal or accident conditions

As mentioned earlier, a nuclear reactor must be cooled even under the abnormal and accident conditions in order to protect public health and safety. In most of the reactors, there are inherent safety mechanisms through negative moderator temperature, fuel temperature and void reactivity feedback. For example, in a BWR, a core power increase results in an increase of steam voids which has a detrimental effect on neutron moderation thus core power spontaneously decreases. In other reactors
which may be cooled by single-phase liquid or gas, an increase in moderator or fuel temperature (called “Doppler” effect) decreases nuclear fission and thus core power. In spite of these inherent safety mechanisms, all nuclear reactors are equipped with control rods and liquids with neutron absorbing material like boron to “shutdown” or stop the fission reaction to avoid any violation of safety or thermal limits. In addition, ECCS is employed in all nuclear reactors to remove the “decay heat” mentioned in Section 1.1 in order to meet the safety criteria for transients and accidents discussed below.

Events at a nuclear plant are typically categorized based on their frequency of occurrence. Those events that are expected to occur during the lifetime of a particular plant are often referred to as “transients.” Because transients are expected to occur, the success criteria are established to ensure that only a very small percentage of fuel rods (less than 0.1%) may experience CHF, DNB or BT. A more limited set of events are defined as design basis accidents or DBAs and are generally not expected to occur but are postulated because their consequences could include the potential for the release of significant amounts of radioactive material.

For the DBAs, the nuclear regulators of a country usually specify the acceptance criteria for emergency core cooling systems. In the United States, the U.S. Nuclear Regulatory Commission (USNRC) in 10 CFR (Code for Federal Regulation) Section 50.46 [28] specifies:

“Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated.”

The major criteria set forth in paragraph (b) are:

1. **Peak cladding temperature.** The calculated maximum fuel element cladding temperature shall not exceed 2200°F (1204°C).
2. **Maximum cladding oxidation.** The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. **Maximum hydrogen generation.** The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. **Coolable geometry.** Calculated changes in core geometry shall be such that the core remains amenable to cooling.

Some revisions to the above criteria are under consideration at this time. Nevertheless, it is clear from the above discussion that performance evaluation of the reactor safety systems including ECCS requires transient thermal hydraulic calculations to check if the safety systems will satisfy the criteria mentioned above for transients and DBAs. This is the topic of Chapter 4. However, we will first discuss the important thermal hydraulic aspects for steady state operation of a reactor in Chapter 2, followed by a general discussion on the active and passive safety systems employed in various reactors in Chapter 3.

### 1.4 Scope of this monograph

The purpose of this monograph is to provide a concise yet comprehensive account of thermal hydraulics in a nuclear power reactor during steady state and transients including accident conditions. Primary focus is on the water cooled reactors including PWR, BWR and PHWR. Brief comments are also made on the thermal hydraulics of more promising small modular reactors (SMRs) and Generation IV reactors.

This monograph does not cover the reactor kinetics or the physics of heat generation due to nuclear fission. Neither does it cover the thermal hydraulics of severe accidents leading to a core melt.