Chapter 1. Introduction

1.1 Introduction

This book is intended as a textbook for the study of materials behavior in light water reactors. The target audience is upper-division undergraduate students and beginning graduate students in nuclear engineering. No specific knowledge of either nuclear engineering or materials science is assumed, since the basic concepts necessary to study materials behavior in the reactor environment are introduced in the book.

Covered are the basic principles and practical problems related to materials used in a light water reactor, the primary coolant circuit and the pressure vessel. The book is organized into two volumes: the first includes basic topics such as thermodynamics of solids, crystallography, dislocations, phase transformations, mechanical behavior, aqueous corrosion and radiation effects, and the second emphasizes applications such as stress-corrosion cracking, dimensional instability and materials degradation in the reactor environment.

1.2 What is a Light-Water Reactor?

Nuclear reactors have been built for numerous purposes: electric power generation; research and teaching; materials testing; production of medical and industrial radioisotopes; weapons production; propulsion; and industrial and residential heating. Of these many variants, the power reactor is by far the most significant, in terms of number of units, nuclear fuel consumption, radioactive waste production, personnel employed, and total monetary investment.

Many power reactor types have been designed and developed, but most have had only brief commercial lives[1]. The *light water reactor* (LWR) has survived as the major provider (85%) of the nuclear-generated electricity in the world[1]. Figure 1.1 illustrates the parts of what is called the *nuclear steam supply*, ranging from the *containment*, which is the iconic symbol of a nuclear power plant, to the *fuel pellet*, the system's smallest component.

[1] Most of the remainder is generated by the heavy-water moderated CANDU (Canadian Deuterium-Uranium) reactor, which is not covered in this book.

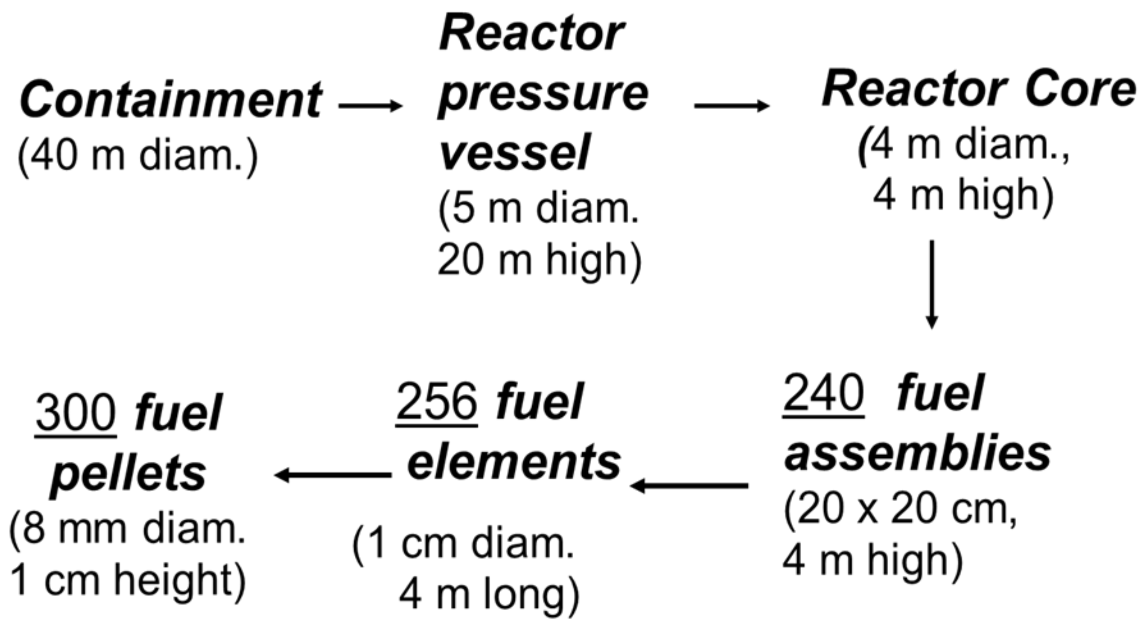


Fig. 1.1 Components of a PWR 1000 MWe nuclear steam supply

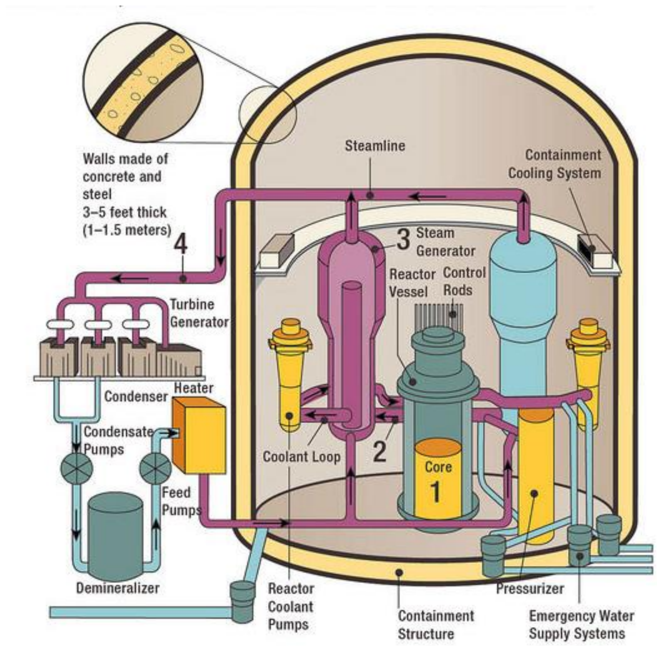
The nuclear steam supply includes all components needed to produce high-pressure steam. The *balance-of-plant* includes all components by which electric power is extracted from part of the steam's energy. Figure 1.2 shows schematic illustrations of the most common types of Light Water Reactors: the Pressurized Water Reactor, PWR (1.2 (a)) and a Boiling Water Reactor (1.2 (b)). In the PWR the *reactor steam* is created in the steam generator, which is the link between the primary and seconday circuits. The term *balance-of-plant* applies to the non-nuclear part of the reactor and covers the secondary circuit and turbine for the PWR and the turbine for the BWR.

The *reactor pressure vessel* (RPV) contains the *reactor core* and the water coolant that receives the heat it generates. Table 1.1 lists the characteristics of the RPVs of the two reactor types. The overall dimensions of the BWR vessel are significantly larger than those of the PWR, mainly because steam is generated directly in the core in the former but in the steam generator in the latter. Because the BWR vessel operates at less than half the pressure of the PWR, its pressure vessel wall is considerably less thick than that of a PWR.

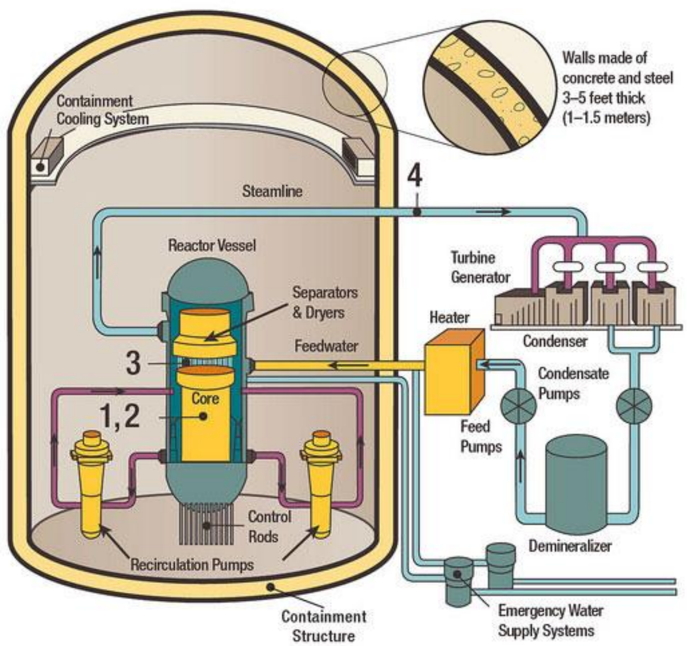
Table 1.1 Characteristics of the Reactor Pressure Vessel [2]

	PWR	**BWR**
Inside diameter, cm	450	600
Wall thickness, cm	25	15
Material	Low-alloy steel*	Low-alloy steel*
Height, m	13	24
Water pressure, MPa	15	7
Outlet Water Temperature, K	598-602	575

*SA508 is a typical alloy used for pressure vessel material. Its composition (%) is Mn 0.5-1.0, Ni 0.5-1.0, Cr 0.25-0.45, Mo 0.55-0.7, Si 0.15-0.4, C <0.27, V <0.05, P(max) 0.025, S(max)0.025



(a)



(b)

Fig.1.2. The principal components of two Light Water Reactors (a) a Pressurized Water Reactor (PWR) and (b) a Boiling Water Reactor (BWR).

The reactor core contains *fuel assemblies, support structures* and *control components*. Figure 1.3 shows a cutaway of the internal structure of a BWR, with one fuel assembly in place. The *core barrel* surrounds the array of fuel assemblies. The water-filled space between the core barrel and the RPV is about 50 cm wide. Its function is to reduce the energy of high-energy neutrons escaping the core in order to minimize radiation damage to the pressure-vessel wall.

The fuel assemblies of a PWR and a BWR shown in Fig. 1.4 are packed with fuel rods. The BWR fuel assembly is housed in a square can called a *duct*, whereas the fuel assemblies of PWRs are open. In both cases, coolant water flows upward in the fuel assembly, but the ductless PWR version permits horizontal flow, which mixes coolant between adjacent assemblies.

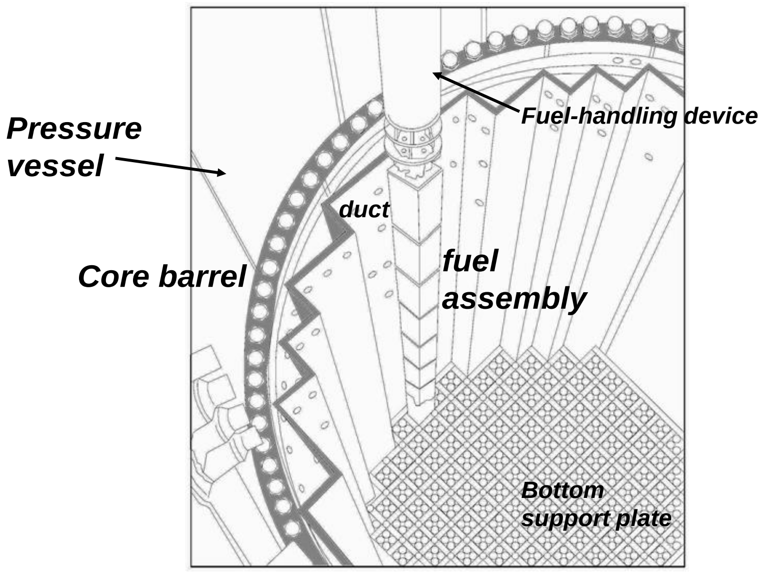


Fig. 1.3 A section of the inside of a BWR pressure vessel

The "spider" type assembly shown in the right hand side of Figure 1.4 (b) in the PWR is made of *control rods*, which are filled with neutron-absorbing materials such as silver-indium-cadmium alloys or gadolinium. As the U-235 in the fuel is gradually consumed (or "*burned*"), the control rods are withdrawn in order to maintain nuclear criticality. The alternative for controlling reactivity which is used in most recent core designs which have greater amount of reactivity in the core at the beginning of life, is to use dissolved boron in the reactor coolant as a *burnable poison*, since as the boron absorbs neutrons it turns into non absorbing Li, so that the amount of poison can be designed to slowly decrease with time so as to keep reactivity constant. In the BWR, instead of control rods, the neutron absorber (boron carbide) is contained in cruciform metal *control blades*. These blades fit between ducts and are moved through the *bottom head* of the pressure vessel (see Fig. 1.8) since the upper part of the BWR contains the steam separators. For reactivity control, the burnable poison is in the form of gadolinia, which is in solid form, mixed in with the uranium dioxide in the fuel pellet, and which serves the same function as the dissolved boron in PWRs.

In addition to the RPV, Fig. 1.2 shows other components inside the containment building, principally piping leading to and from the reactor, the large *feedwater pump*, and the *steam generator* (in PWRs). The *primary circuit* refers to the closed loop in which water circulates between the RPV and external components such as the steam generator (PWR) or the turbine (BWR). In the PWR, the primary circuit coincides with the nuclear steam supply; in the BWR, the turbine and condenser are parts of both the primary circuit and the balance-of-plant.

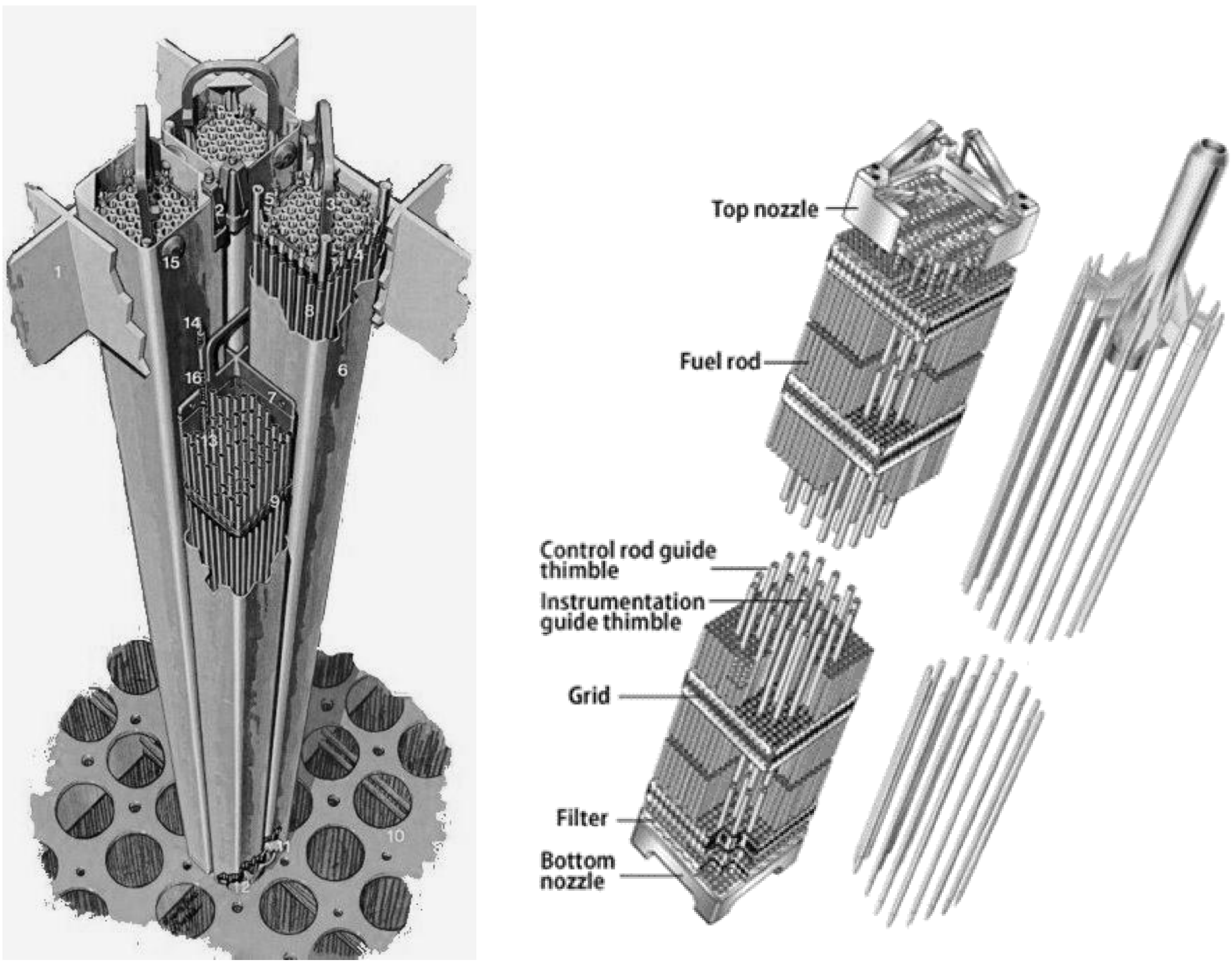


Fig. 1.4 fuel assembly: BWR (left) and PWR (right)

The low-alloy-steel RPV contains the high-pressure water that flows up through the core and removes fission heat. The top hemisphere of the pressure vessel (called the *upper head*) is bolted to the cylindrical side wall to permit opening of the vessel for refueling and maintenance. There are three types of penetrations of the reactor pressure vessel:

- inlet and outlet *nozzles* that connect the body of the pressure vessel to large pipes (~ 50 cm diameter). These pipes carry the hot coolant (liquid water for PWRs or saturated steam for BWRs) from the RPV to the other components in the primary circuit, and eventually return (relatively) cold water to the RPV.
- *control rods* and *safety rod drives*. Both types contain neutron absorbers; the former are slowly removed during reactor operation to make up for loss of U-235;the latter are used to shut down the reactor quickly in an accident situation
- instrumentation for measuring the in-core neutron flux and temperature.

The balance-of-plant includes the steam turbine, the electric generator, and a condenser to convert the exhaust steam to liquid water. The balance-of-plant could equally well be connected to a fossil unit (coal, natural gas, or oil) for its steam supply.

This book deals with the materials aspects of the components of the core and the primary circuit, but excludes the balance-of-plant and the concrete containment building that houses all components. The types of materials within the nuclear steam supply are limited by the need to withstand a very hostile environment. The core, where nuclear energy is converted to thermal energy, is characterized by high stresses, high temperatures, extreme radiation fields, and aggressive chemistry. The materials used must satisfy stringent criteria regarding their mechanical, physical, chemical, and nuclear properties.

This chapter describes the features common to both LWR versions, then outlines their specific characteristics and the materials used in their construction and operation. Also described are the current challenges and economic forces that drive nuclear materials research.

1.3 Fuel Rods and Fuel Assemblies

Energy from fission is converted into thermal energy in the *fuel element* or *fuel rod* (Fig. 1.5). The energy is then transferred as heat to the water coolant that flows through the core past the fuel rods. Fuel rods are constructed (in fuel-fabrication factories, separate from the reactor site) by inserting a stack of *fuel pellets* measuring about 1 cm diameter, 1 cm high into a 4-m length of metal tubing called *cladding*. A spring is placed on the pellet stack for mechanical stability, and caps are welded on the ends of the tube. The cladding tube performs the following functions:

- holds the fuel-pellet stack in a mechanically-stable, coolable geometry;
- retains fission products released from the fuel;
- prevents coolant water from contacting the UO_2 pellets

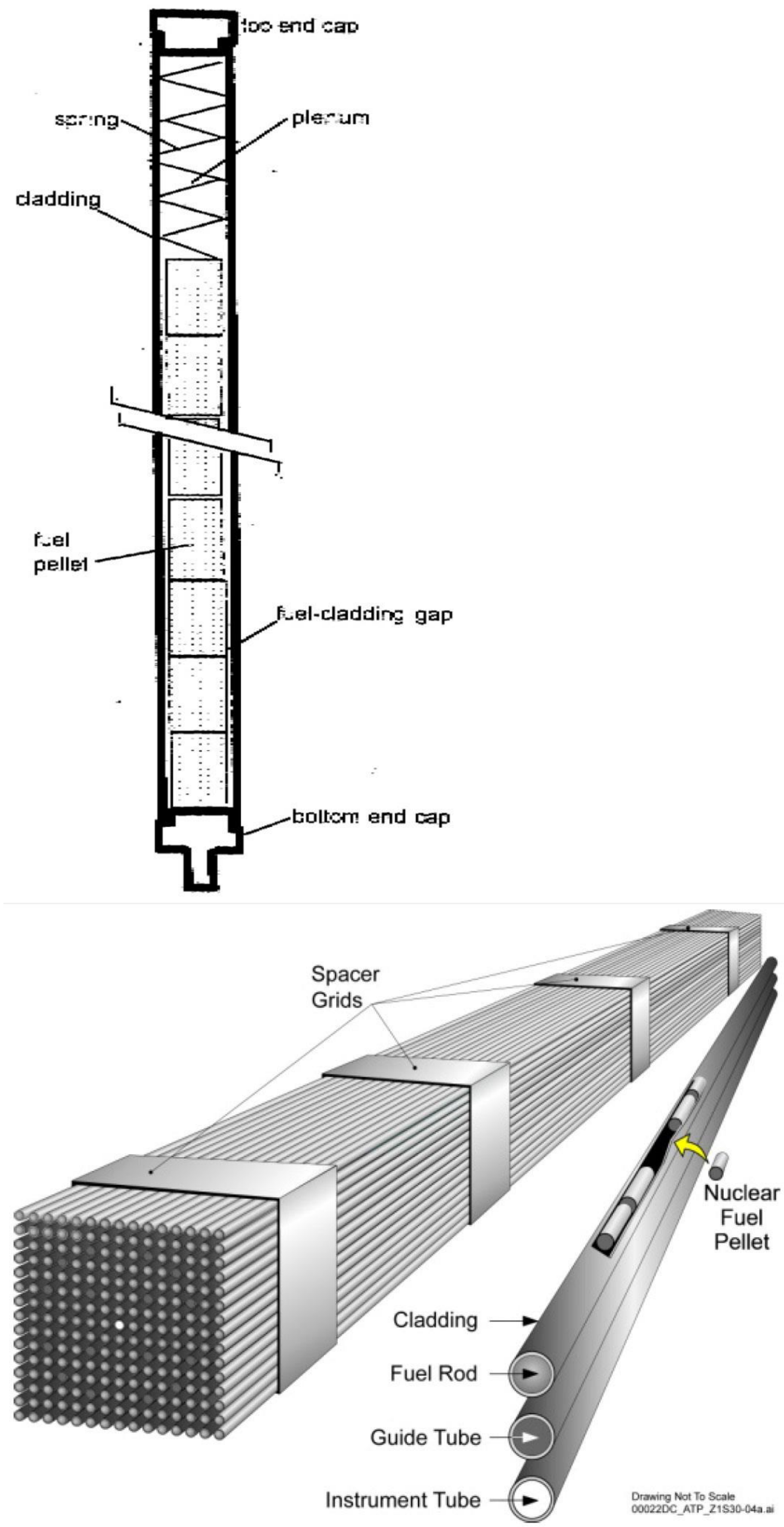


Fig. 1.5 (a) Cross section of a Nuclear Fuel Rod (b) a schematic of a fuel assembly

The fuel pellets are made of high-purity *uranium dioxide* (UO_2) enriched to 3-5% in U-235. A small fraction of the nuclear fuel used in the world is *mixed-oxide fuel (MOX)* being constituted of a mixture of PuO_2 in UO_2 [(U,Pu)O2] to utilize plutonium generated by irradiation of UO_2 or recovered from nuclear weapons for power production. Cladding is made of the zirconium alloy *Zircaloy* (Zr + Sn + some Fe,Cr,Ni), or more modern niobium-containing zirconium alloys, as discussed in Chapter 17. The choice of zirconium alloys is based mainly on its low neutron absorption cross section, although mechanical, chemical and physical properties are also considered.

For the purposes of handling during transportation, loading/unloading, storage, and for maintaining mechanical stability and proper spacing in the rapidly- flowing coolant, fuel elements are packaged into units known as *fuel assemblies.* These are shown in Fig. 1.4 for the two types of light-water reactors. The distance between the centers of adjacent fuel rods when loaded into the assembly is called the *pitch.* The rod separation is kept as small as possible in order to minimize the size of the assembly, and hence of the reactor core and the RPV. The minimum pitch is dictated by the maximum attainable coolant flow rate through the core.

The PWR fuel assembly contains as many as 17 fuel elements along each of its four sides, and is referred to as a 17x17 assembly. The overall assembly shape is square, as is the pattern of the rods forming the bundle. The square dimension of the assembly is 20 – 25 cm and its height is nearly 5 m.

At the bottom of the fuel assembly is a steel *lower end plate.* (see Fig. 1.4). This plate is penetrated by two sets of holes, each with a distinct function: the first set of small holes receives the lower ends of the fuel rods. A second set of larger holes serves as entries for cooling water into the core region. The tops of the fuel rods are set into indentations in an *upper end plate*. The upper and lower end plates are connected by long rods bolted on both ends in order to firmly fix the fuel rods.

Also shown in Fig. 1.4 is the square box (called the *duct*) into which the BWR fuel bundle fits. The walls of the duct are ~ 3 mm-thick Zircaloy. By varying the diameter of the coolant entry orifices in the bottom end plate, the flow to each channel can be adjusted to compensate for the core radial power distribution so as to maintain a uniform steam quality (i.e., vapor fraction) at the exit of all fuel assemblies. There is no duct housing the fuel assemblies in a PWR.

1.4 Fuel rod stabilization – grid spacers

Cladding tubes are quite flexible, especially in 4-m lengths. The stack of 1-cm-long fuel pellets inside the cladding does not provide rigidity to the fuel element. Bolting together the upper and lower end plates suffices only to hold the ends of the fuel rods. An additional design element is needed to prevent vibration of and possible damage to the rods during transportation to the reactor site or by the flow of high-velocity cooling water during reactor operation. To stabilize the fuel rod bundle, *grid spacers* are placed at regular intervals along the assembly (Fig. 1.4). A critical component of both the BWR and the PWR fuel assemblies is the *grid* spacer, also shown in Figure 1.4. The principal function of this piece is to hold the fuel rods in place. Fuel rods are very flexible; if held horizontally, a fuel rod would bend of its own weight. The vertical spacing of the grids along the fuel bundle is chosen to minimize vibration of the rods produced by the flowing coolant. Typically, seven grid spacers are placed along the assembly. They are made either of Zircaloy or of the high-nickel alloy Inconel. Details of a PWR grid spacer are shown in Fig. 1.6. BWR grid spacers are similar to PWR designs but without the control-rod guide tubes.

In both LWR types, the vertical sides of the square holes through which the fuel rods pass are formed into shapes resembling a leaf (spring clip), an "I" beam, or a dimple. These are the points of contact with the fuel rod, and the force that the contacts exert on the fuel rods must be set with great care. Too large a force causes cladding deformation, usually in the form of bowing of the rods. Too small a force allows the fuel rods to rattle about in their hole and could lead to failure by fretting.

1.5 Neutron-absorber devices

1.5.1 PWR control rods

In addition to fuel rods, LWR cores need *control elements* for regulation of the neutron population and *burnable poisons* to account for the decrease in nuclear fuel reactivity with time. In PWRs, movable *control rods* containing pellets of B_4C (scramming) or Ag-In-Cd (for fine tuning) fit into *guide tubes* that pass through and are welded to the grid spacer as shown in Fig. 1.6. The right-hand drawing in Fig. 1.4 shows the cluster of control rods passing through the grid spacers in a PWR. The guide tubes also hold gadolinia/urania burnable poison rods which are slowly withdrawn as fission proceeds. In addition, the guide tubes hold neutron sources for reactor startup.

1.5.2 BWR control blades

Grid spacers for BWRs do not have control rods inside the fuel assembly. Instead, as shown in Fig. 1.7, the burnable poison (boron nitride) is contained as pellets in small rods enclosed in a cruciform-shaped Zircaloy sheath. These *control blades* are inserted between the ducts (lower left-hand diagram of Fig. 1.9) and are moved up and down through penetrations in the bottom head of the RPV.

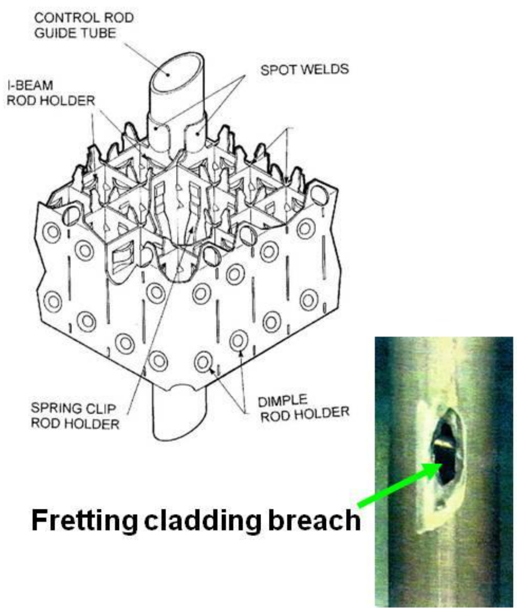


Fig. 1.6   PWR Grid spacer with control-rod guide tube [3]

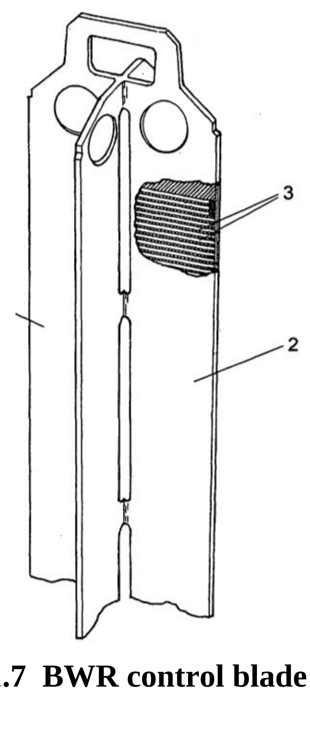


Fig. 1.7 BWR control blade

1.6 Coolant: H_2O

Besides the UO_2 fuel and the Zircaloy structures, the third material in the core and the primary circuit is ordinary water, the eponymous component of the LWR. Its high heat capacity and low viscosity make water a very efficient medium for collecting the heat emanating from the fuel rods and transporting it out of the RPV. The hydrogen in H_2O permits the cooling function to be combined with the nuclear function of moderating (slowing down) neutrons. However, the neutron absorption cross section of hydrogen, while not large, is high enough to require enrichment of the fuel to ^{235}U contents of 3% and up to 5% (as compared to the CANDU reactor which can use natural uranium (0.7% enriched) because it uses heavy water, in which deuterium is much less absorbent of neutrons than hydrogen).

Water has the advantage over other coolants in that, as steam, it can directly drive a turbine-generator, just as in coal- or gas-fired plants. The major disadvantages of water in a nuclear system are its high vapor pressure and its chemical reactivity to all metals in the primary circuit.

The efficiency of converting thermal energy to electrical energy increases with the temperature of the hot source, in this case the coolant leaving the core. However, a high exit temperature must be accompanied by a high system pressure (15 MPa for a PWR, 7 MPa for a BWR). The structures in the primary circuit must be sized to withstand the resulting stresses. In addition to purely mechanical constraints on material performance, these stresses can enhance certain corrosion processes (see Chap. 15).

Corrosion of metal structures is a persistent problem in LWRs: cladding corrosion (Chap 22) can limit the length of time that the fuel can remain in the reactor; stress-assisted cracking/corrosion of stainless steel in BWRs (Ch.25) has required expensive mitigation procedures or even complete replacement of primary circuit piping; other corrosion mechanisms afflict the components of the steam generators of PWRs and necessitate plugging of leaking tubes or even complete replacement of the entire unit

Coolant water also contains additives to help the metals in the primary circuit resist corrosion. In BWRs, hydrogen (as H_2) is added to the water to remove oxidizing free radicals that are produced by radiolysis in reactor. No solid additives are allowed because the water is vaporized to steam, and any solids in solution would concentrate in the liquid.

In PWRs, solids in solution are used for various purposes. Boric acid (H_3BO_4) is added for neutron reactivity control; LiOH is added to neutralize the acidity of the boric acid. Zinc is added to reduce radioactivity uptake in the primary-circuit piping (Chap. 21).

1.7 Water Reactors

1.7.1 PWR

As the name suggests, the pressurized water reactor contains liquid water throughout its primary circuit[2]. With reference to Fig. 1.8, the heated water exits the reactor pressure vessel at ~ 330^{o}C, safely below the 345^{o}C saturation temperature at the operating pressure of 15 MPa. The hot water passes through U-shaped tubes in the steam generator (Fig. 1.2) before being returned to the RPV at ~290^{o}C. The return water flows downward along the inner wall to the lower hemisphere of the RPV (called the *lower head*). From here, the flow is directed upward through the bottom end plate past the fuel rods and exits at the outlet plenum via a large (~50 cm diameter) pipe, where the primary circuit begins.

There are three levels of isotopic enrichment of the fuel, but the fuel rods in a particular assembly have the same enrichment. The enrichments correspond to fuel added in each of the three refueling outages. Every 18 - 24 months, one third of the core is removed and replaced with fresh fuel. Fuel in its 3rd cycle is the most highly burned and has the lowest neutron reactivity. Fresh fuel is the most reactive, so to flatten the radial power distribution in the core, it is placed along the core periphery. The downside of this fuel management strategy is the enhanced embrittlement of the RPV wall by fast neutrons leaking from the core (see Chap. 26).

Control rod drives (motors that move control rods up and down) are connected to pressure-tight penetrations in the upper head. The control rod drives are attached to the hub of the spoke-like control rod cluster mechanism shown in the lower-left fuel assembly in the bottom of Fig. 1.8. The partially-filled circles at the ends of the spokes represent control rods which fit into the guide tubes of the PWR assembly in Fig. 1.3. Not all fuel assemblies contain control rods; only two of the four-assembly sketch in Fig.1.8 are provided with them. The other two assemblies contain only fuel rods. The core cross section on the right at the bottom of Fig. 1.8 shows how the enrichments in the initial loading of fuel can be distributed in order to create a uniform power distribution.

1.7.2 BWR

The principal difference between the two types of LWRs is the conversion of liquid water to steam in the core of a BWR instead of in an external steam generator, as used with PWRs. The reactor in a BWR plant is an integral part of the power cycle. As shown in Fig. 1.2, the steam leaving the reactor pressure vessel flows directly to the turbine-generator. The low pressure exhaust steam from the turbine passes through a condenser to complete conversion to the liquid state. Following the condenser, the low pressure liquid coolant is pumped through units that remove impurities and add corrosion inhibitors. Finally, the chemically-prepared (or “polished”) coolant is repressurized by the main coolant pump to system pressure of 7 MPa and fed into the RPV.

As shown in Fig. 1.9, the coolant enters the core as a subcooled liquid at 280 ^{o}C. This feedwater is sucked into the lower head of the RPV by a *jet pump* (described below), then flows upward

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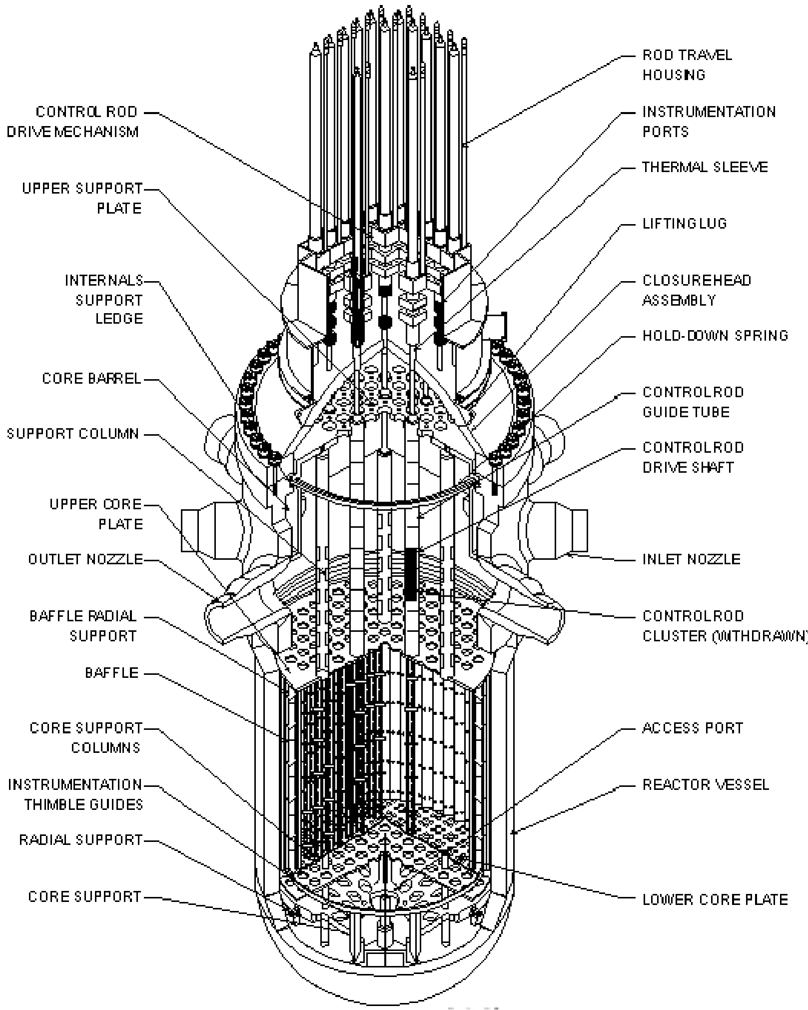
[2] In some PWRs, significant localized boiling takes place in the upper core regions, which reduces the local density of water. The result is a decrease in neutron moderation, a consequent reduction in the thermal neutron flux and fission rate. This phenomenon is labeled the a*xial offset anomaly.*

through the fuel assemblies stacked in the configuration in the lower right. In the lower third of the core, the coolant is heated to the saturation temperature (288^{o}C at 7.1 MPa). In the middle



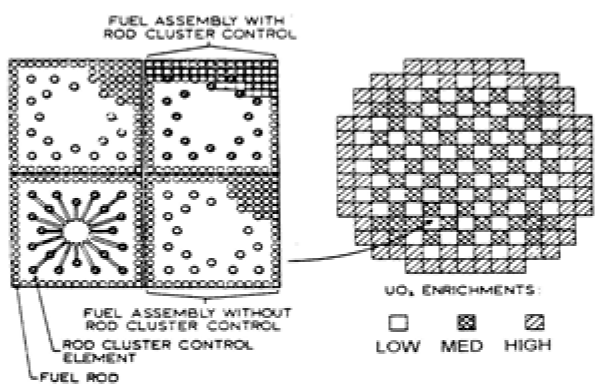


Fig. 1.8 Pressurized Water Reactor. Bottom left: cross section of 4 adjacent fuel assemblies; Bottom right: cross section of a typical loading of fuel assemblies containing various levels of enrichment in the core
(http://en.wikipedia.org/wiki/Pressurized_water_reactor)

third, the coolant rises as a foam which is converted to wet steam (i.e., vapor with entrained droplets) in the upper third of the core. Because high-velocity wet steam erodes metal surfaces, the separator unit above the core outlet permits gravity removal of the water droplets from the rising steam. Dryers just before the exit pipe convert any remaining liquid to steam. The outlet steam is not superheated (temperature greater than the saturation temperature) as it is in some fossil power plants.

The device resembling wings on the RPV in Fig. 1.9 is part of an auxiliary recirculation circuit which serves two purposes: first, via jet pumps, it delivers the main feed flow from the inlet pipe to the lower head; second, it captures the slowly-settling droplets from the steam separator and efficiently mixes them with the main coolant flow. As shown in the sketch, the recirculation pump is placed outside the RPV. The flow from this loop is injected into the funnel-like opening to the downcomer pipe situated between the core periphery and the inner wall of the RPV. The high flow rate of water in this device acts as a pump that efficiently returns all liquid above the core to the core entrance at the lower head.

The drives for the control blades enter the RPV through the bottom head. The crosses in the lower-right-hand diagram of Fig. 1.9 represent the distribution of cruciform-shaped control blades (Fig. 1.7) distributed among the fuel assemblies. Control material is contained in four cruciform blades that fit between four assemblies (Fig. 1.9, lower left). The control blade assembly is moved into and out of the core from the bottom of the reactor pressure vessel. The large number of control blades means that each blade controls a small amount of nuclear reactivity. This feature avoids large local power changes due to movement of a single blade, which could produce damaging thermal stresses in the nearby fuel rods. Reducing power around the failed rod decreases the loss of fission products and minimizes further degradation of the failed element.

The fuel assembly shown on the left in Fig. 1.4 is a 9x9 BWR design. Although BWR rods are larger than those of the PWR, the lateral dimension of the BWR assembly is smaller because there are fewer rods in each row.

For neutron moderation, a few of the rods at the center of the BWR assembly are filled with liquid water instead of fuel. In some designs, the center of the fuel assembly contains a square water-filled channel. The purpose of these “water rods” is to increase the hydrogen density and thereby provide more neutron moderation than is available in the steam filling the spaces between rods.

1.8 Reactor Pressure Vessel Internals and Ex-Core Materials

In addition to fuel and control materials, the RPV contains numerous other components. The structural alloys used in these components include:

- the reactor pressure vessel (ferritic steel ("low-alloy steel") clad internally with stainless steel),
- reactor internals (stainless steel and nickel-based alloys):
    - mechanical support for the reactor core,
    - structures to direct the flow of coolant
    - shielding for the RPV: core barrel; core shroud
    - instrumentation guide tubes.
    - steam separator (BWR); steam generator (PWR)

Table 1.2 summarizes the major reactor-related components of a 1000 MWe nuclear plant, the materials of which they are made, their approximate mass, and the principal materials-related issues encountered during operation. The last of these contains only the current issues; problems that appeared in the past but have been solved (e.g., irradiation densification of fuel) are not included in the list, although some of these are discussed in the book. Many of the materials problems in ex-core components listed in Table 1.2 have actually been encountered while others are potential problems, which nevertheless need to be considered in safety analyses. In the latter category are the consequences of irradiation hardening of the reactor pressure vessel, the chief one being brittle fracture. Although no RPV failure has occurred, preventative measures have resulted in power reduction in some PWR reactors believed to be prone to such an accident.

Only pressure-vessel failure and massive fracture of large-diameter pipes are reactor-safety issues. These and accidents that can cause fuel damage such as the reactivity-initiated accident are *design basis accidents*. For example, fracture of large diameter pipes can lead to a loss-of-coolant accident (LOCA). The other materials problems listed in Table 1.2 affect operation of the reactor, and include issues that:

- Exceed the technical specifications that govern the limits of operating the reactor (e.g., radioactivity in the primary circuit coolant due to fission products released from a failed fuel element)
- Require derating of the component (e.g., plugged tubes in the steam generator of a PWR)
- Force an unscheduled outage (e.g., a fuel element cladding failure large enough to release fuel to the coolant)
- impede regular maintenance because of high radiation levels in the primary circuit (e.g., excessive deposition of activated cobalt-60 in the primary circuit (Chap. 21))
- Make it necessary to show that degradation of plant materials is not excessive before plant life extension can be granted.

Table 1.2 Materials in a 1000 Mwe Nuclear Steam Supply

Component	**Material**$^{+}$	**Mass Mt**	**Materials Issues**
Fuel	UO_2, $(U,Pu)O_2$	100	Fission-gas release; (fission product swelling; thermal conductivity & burnup (Chap. 20)
Cladding, grid spacers	Zircaloy: 1.7 Sn; 0.5 (Fe, Ni, Cr); 0.1 O; bal Zr; Zr-Nb	25	Waterside corrosion and hydriding (Chaps.22 and 23); embrittlement (Chap. 26), growth (Chap 27); pellet-cladding interaction (Chap.23)

			; fretting
Neutron absorbers	Ag-Cd-In (PWR), B_4C (BWR); Gd_2O_3 (both)	~1	None except in severe accident
Reactor Pressure Vessel	Low-alloy steel 2 Cr ; 1 Mo; bal Fe	350	Radiation embrittlement (PWR only) (Chap 26)
Steam Generator (PWR)	Inconel 600: 25 Cr; 15 Fe; bal Ni and 690		Tube plugging, cracking and denting; leakage from the primary coolant to the secondary loop
Reactor Internals	Stainless Steel:18 Cr; 8 Ni; bal Fe; Inconel		Stress-Corrosion Cracking (Chap. 25), Fatigue.
Ex-core components, primary piping	Stainless steel	-	Stress-corrosion cracking (BWR) (Chap. 25)
Valves, pumps	Stainless steel; stellite: (high-cobalt steel)	-	Cobalt dissolution => activation in core => deposition in primary circuit (Chap. 21)
Special Components	alloy 718 Ni 52.5, Cr 19.0, Fe bal., Mo 3.0, Mn 0.35,		Creep (Chap 27), SCC (Chap 25)
+ **the number next to each element is the weight percent in the alloy**			

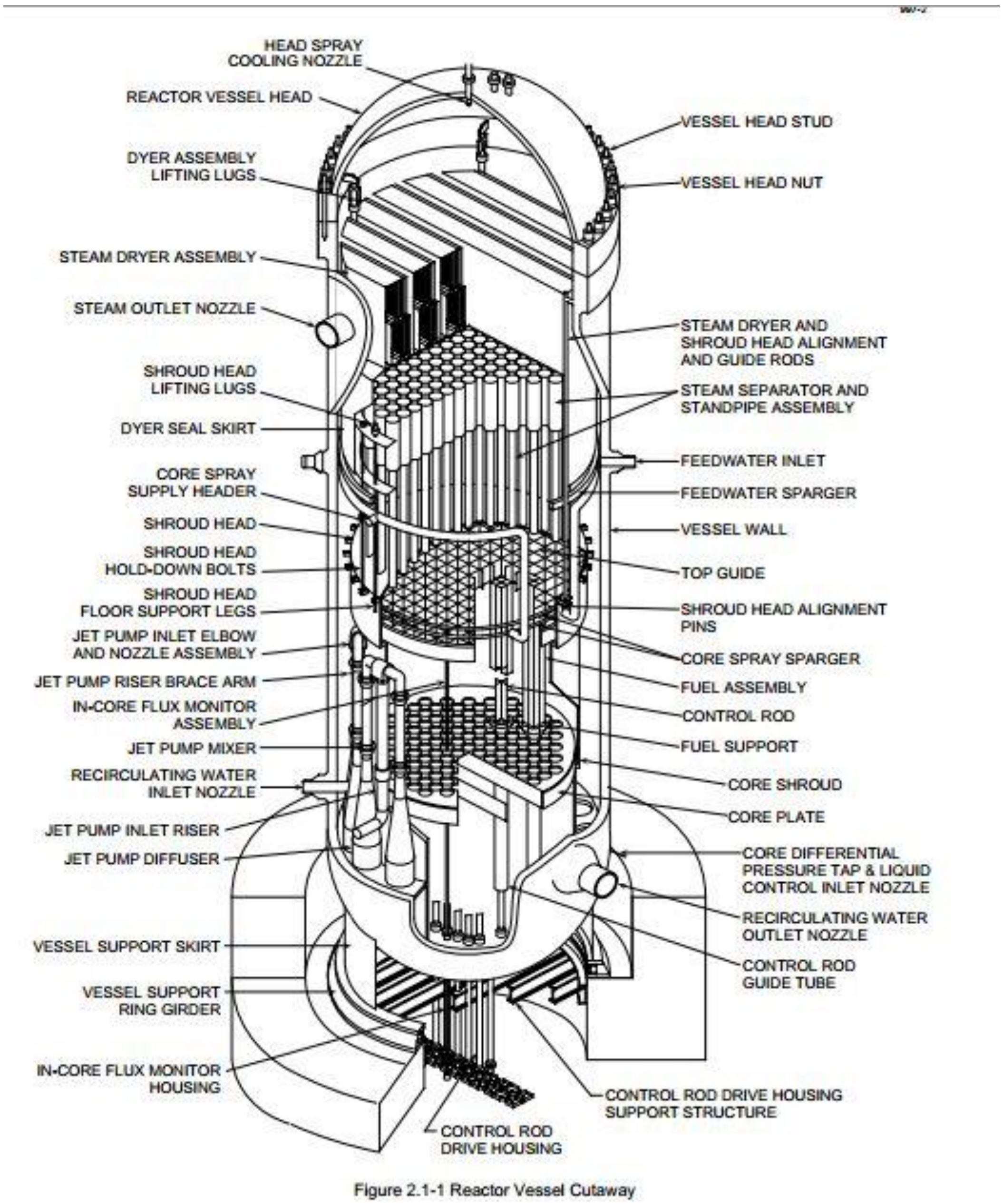


Fig. 1.9 Boiling Water Reactor.
http://www.fukuleaks.org/web/wp-content/uploads/2014/03/reactor_vessel_cutaway.jpg

1.9 Capacity Factor

The single most important measure of the efficiency of a power plant is its *capacity factor*. This quantity is defined as the net quantity of electricity generated in one year divided by the maximum possible amount had operation been continuous at the rated power. Because in the U.S. nuclear power plants are used as baseline power, this is normally equivalent to the *availability factor*, (fraction of time the plant is available for electricity production when called upon to produce.)

The capacity factor is generally less than 100% because of shutdowns due to the following:

- *scheduled outages* for refueling and regular maintenance (typically for 20 – 30 days every 18-24 months – a capacity factor loss of ~ 5%)
- *Forced outages* due to equipment or fuel-element failures
- Operation at reduced power due to:
  - *load-following* (changing power because of reduced or increased demand on the electrical grid, or because other sources, such as renewables, need to be used)
  - power maneuvering at a low rate (to avoid damage to fuel elements)
  - operation at less than maximum power: to avoid further degradation of damaged fuel; to accommodate a de-rated steam generator; to reduce the rate of accumulation of radiation damage by the reactor pressure vessel; also because of seasonal variations in the temperature of the cooling river water.

Plant downtime is mostly caused by refueling. Over a five-year period in the 1990s this represented the 29-day/year scheduled refueling outage, although this number is likely lower now. Numerous other problems in the nuclear steam supply such as shutdowns or de-rating due to steam generator problems together contribute a 6% (23 days/yr) capacity-factor loss. Although fuel-rod failures (Chap. 23) are not even in the top 50% of the reactor-related causes of forced outages, minimization of fuel-rod failures is of continuing concern because problems with this component can cause operational difficulties that are not immediately reflected in the capacity factor and also because the degradation mechanisms may become more severe at high burnup.

Although only one additional nuclear power plant (Browns Ferry, 2010) came on line in the last twenty years, the percentage of nuclear power in the total electricity mix in the U.S. steadily increased in the 1980s and 1990s, and stabilized in the 2000s, even as the total electric-energy generated has increased. Most of the growth in the nuclear contribution is attributable to the increase in capacity factors from ~ 57% in 1986 to over 90% in 2002, the equivalent of building 22 new nuclear power plants over that period. This was achieved by: reduction in scheduled outage times; increases from 12 to 18/24 month cycles (time between scheduled outages); and decreases in the number of forced outages. Additional capacity factor increases were provided by power uprates. The capacity factors have more or less held steady since then.

1.10 Current Challenges

As of the writing of this book, 101 commercial nuclear power plants operated in the United States, providing a total of ~100,000 MWe or slightly less than 20% of the nation's electricity. Over the past decade, operation of these power plants has continually improved, due in large part to the resolution of various materials problems. However, new problems, nearly all associated with materials performance, have arisen as nuclear plant operators seek to extract as much energy from the fuel as possible. The principal methods are:

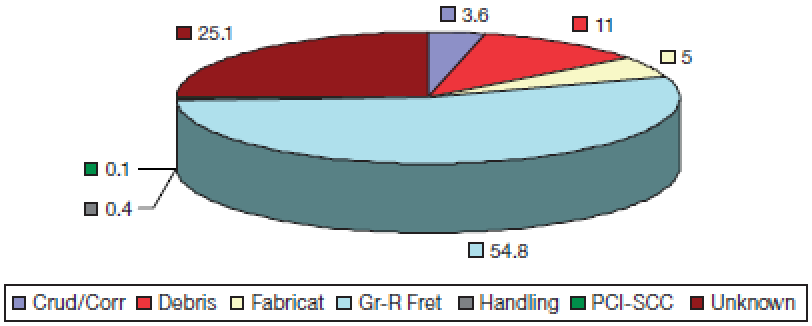
- Extended burnup increasing average fuel discharge burnup to 60 - 70 MWd/kgU delivers a proportional decrease in the volume of radioactive waste from spent fuel. By using more highly enriched fuel and consuming a larger fraction of the ^{235}U, less fuel is required to generate the same energy, thereby reducing fuel-fabrication costs. However, the fuel rods experience increased radiation damage, corrosion and more extensive hydriding of the cladding. Greater fission gas release leads to higher fuel-rod pressures and swelling of the fuel increases the stress in the cladding. Note that the extended burnup has so far been achieved with no loss in capacity factor which means fuel failures have decreased even as fuel duty increased.

- Power uprate: at constant coolant flow rate, increasing the power in a PWR results in a higher outlet coolant temperature. This increases the efficiency of the energy conversion and produces more electricity for a given thermal power. The higher fuel temperatures promotes undesirable coolant boiling on the cladding surface, with corresponding increases in thermally-activated processes that lead to fuel degradation, including fission-gas release, corrosion and hydriding.

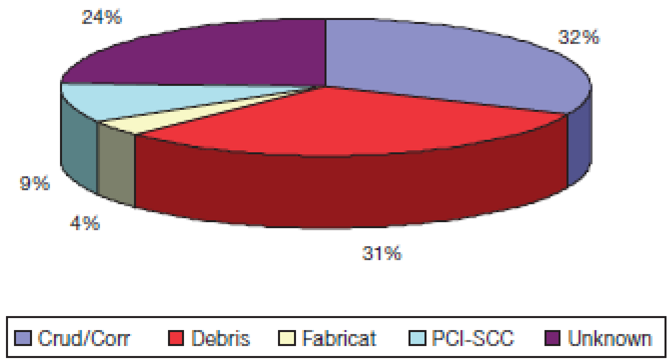
- Longer refueling cycle: increasing the time between refueling outages from 12 months to 18-24 months is directly reflected in an increased capacity factor

All of these increase the severity of the fuel duty, such that fuel reliability needs to be constantly improved. Fig. 1.10 shows the most common causes of fuel failure worldwide in the period 1994-2006 for both PWR and BWR. Grid to rod fretting dominates the failures in PWR which has led fuel vendors to pay close attention to the design of fuel assembly and fuel-coolant interactions. For BWR CRUD and corrosion represent a large fraction of the fuel failure causes (see greater discussion in Chapter 23).

Electricity deregulation has driven nuclear utilities to seek stringent cost-reduction measures in order to remain competitive. In spite of the increased demands on the fuel, cladding and structures, the fuel-performance requirements continue to increase.



(a)



(b)

Fig.1.10 Fuel leak causes in reactors worldwide from 1994 to 2006, in (a) PWRs and (b) BWRs. [4]

60% of the original current reactor licenses in the U.S. were scheduled to expire before 2020. Nearly all of the operators of currently-running reactors have applied for renewal of their operating licenses. Most plants in the United States have applied for life extensions of 20 years. The feasibility of continued efficient, economic and safe operation of light water reactors depends crucially on safely aging of nuclear materials. Two of the thorniest issues are:

- certifying that the RPV has not been excessively embrittled after exposure to fast neutrons during its normal life ,
- assuring that steam generator tube cracking (in PWRs) and corrosion and cracking of reactor internals (in BWRs) will not accelerate during the extended plant life.

1.11 Layout of the Book

To cover the materials issues that arise in light-water reactors, the book is split into two main sections: the first (chapters 2 to 15) deals with the fundamentals of materials, thermodynamics, atomic transport in solids, crystallography, defects, phase transformations, corrosion, mechanical behavior, and radiation damage; the second section (chapters 16 to 29) applies these fundamentals to the degradation processes that occur in service. The main focus is on the nuclear fuel and cladding, but the pressure vessel, reactor internals and the other parts of the nuclear supply system are also covered.

Understanding the behavior of materials in nuclear power plants is a complex endeavor. If in the early 1960s, reactor materials experts had been asked to predict the problems that would later occur with UO_2 fuel, Zircaloy cladding, pressure vessel steels, stainless steels, and other nuclear materials, they would have been hard put even to conceive of many of the problems that did arise, let alone predict their outcome. This is especially true for complicated failure mechanisms such as irradiation-assisted stress-corrosion cracking, which involve the synergistic interaction of mechanical, environmental and materials factors, or the complex long-term evolution of damage in pressure-vessel steels. In applying predictive mechanistic models, the complexity of real physical phenomena and the limitations of current understanding must not be forgotten.

1.12 References

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