8

Types of Nuclear Reactors

8.1 Survey of Reactor Types

8.1.1 Uses of Reactors

The first nuclear reactors were built to produce ^{239}Pu for bombs. Subsequently, reactors have been used for many other purposes, of which electricity generation is now, by far, the most prominent. Further uses have been to propel ships (mostly naval vessels), to produce radioisotopes, and, to a limited extent, to supply heat. Many additional reactors have been built for teaching or research, much of the latter involving the study of the properties of materials under neutron bombardment and the intrinsic properties of neutrons and other subatomic particles.

In some cases, applications have been combined. For example, the N reactor at Hanford and the Chernobyl-type reactors were used for both ^{239}Pu production and electricity generation. There has also been limited use of waste heat from reactors to produce hot water or steam for industrial applications, for heavy water production, and for desalination. These reactors have been primarily in the USSR and Canada [1]. The total thermal capacity of these facilities is much less than that of electricity-generating nuclear reactors, although the use of heat from reactors could increase in the future, for example if used for hydrogen production (see Section 16.6).

The focus in this book is on the use of reactors for electricity generation, but the issue of ^{239}Pu production still arises. If the reactor fuel contains ^{238}U, then ^{239}Pu will inevitably be produced following neutron capture in ^{238}U. In breeder reactors, the production of ^{239}Pu is a central goal, with the ^{239}Pu intended as fuel for further reactors. In nonbreeders, which means almost all of the world's operating reactors, ^{239}Pu is a by-product, but its fission, nonetheless, often contributes significantly to the reactor's total energy output. (We

will return in Chapter 17 to the link between power reactors and the possible use of their ^{239}Pu for bombs.)

8.1.2 Classifications of Reactors

Thermal Reactors and Fast Reactors

In previous chapters, reference has been made to the thermalization of neutrons in reactors (i.e., to the slowing of the neutrons to thermal energies). Reactors designed to operate in this fashion are termed *thermal* reactors. However, it is also possible to operate a reactor with "fast" neutrons. There is no moderator in such a reactor and few neutrons reach thermal energies, but many lose energy by inelastic scattering and fission occurs at energies ranging from the MeV region down to the keV region and below. These reactors are called *fast-neutron* reactors or just *fast* reactors. The only prominent example of an operating fast power reactor is the liquid-metal reactor (see Section 8.3.3), although other types of fast reactors have been built for experimental purposes.

Homogeneous and Heterogeneous Reactors

We have been tacitly assuming that the reactors under consideration are what are sometimes known as *heterogeneous*, in which the fuel, coolant, and moderator (if any) are distinct physical entities. All reactors used today for power generation are of this form. However, in the early days of nuclear power, there was considerable exploration of an alternative configuration, the *homogeneous* reactor, defined as "a reactor whose small-scale composition is uniform and isotropic" [2, p. 378]. Homogeneity can be achieved if the fuel is in liquid form, where the liquid is circulated for heat transfer to a steam generator.

One variant of this reactor type was known as the *aqueous homogenous reactor* because the fuel was mixed with water (H_2O or D_2O). In the so-called homogeneous reactor experiment, two small reactors, called HRE-1 and HRE-2, were built at Oak Ridge National Laboratory (ORNL) in the 1950s.[1] For HRE-2, the fluid was uranyl sulfate (UO_2SO_4) in heavy water (D_2O), with the uranium highly enriched in ^{235}U. This program had the potential of developing a thermal breeder reactor, but although HRE-2 operated uninterruptedly for over 100 days at 5 MWe, some difficulties developed, and the program was dropped in favor of alternative liquid-fuel projects.

One alternative was the *molten-salt reactor*. The fluid was a mixture of fluoride compounds, including the fissile component $^{235}UF_4$ and the fertile component ThF_4. After initial operation, $^{233}UF_4$ was successfully tried as an

[1] For a description of the history of this program and the ORNL program on molten-salt breeders, see Chapter 6 of Ref. [3]. Technical aspects of fluid-fuel reactors are also discussed in Ref. [4, pp. 403–413].

alternative to $^{235}UF_4$. Like HRE-2, this reactor was designed to be a thermal breeder reactor. Again, there was initial success in the reactor operation, but a decision was made in the 1960s to abandon development of thermal breeders in favor of fast breeder reactors.[2] A further homogenous reactor approach, a *liquid metal thermal breeder* using uranium compounds in molten bismuth, was also investigated but was abandoned without construction even of a test reactor.

At present, there are no electricity-generating homogenous reactors, and we will not consider them further in the chapter. Instead, we will restrict consideration to heterogeneous reactors, which are so dominant that it is unusual to include the specification "heterogeneous." Nonetheless, some interest remains in homogenous reactors, particularly in molten-salt reactors (see, e.g., Ref. [5]).[3] In fact, one of the reactors selected for possible long-term development under the Generation IV program is a molten-salt reactor (see Chapter 16). Overall, although we focus in this chapter on reactors of the sort in actual use or in immediate prospect, it is well to remember that on a longer time scale, a wide array of variants are possible. Some of these are being tentatively explored in the thinking underway in the United States and elsewhere on nuclear options for the future, as discussed in Chapter 16.

8.1.3 Components of Conventional Reactors

Overall

Any generating plant consists of an array of structural components and a system of mechanical and electrical controls. In a nuclear plant, there are special demands on structural integrity and reliability. In addition, a nuclear reactor is characterized by the use of specialized materials, some aspects of which were already discussed in Chapter 7. In standard reactors, these are the fuel itself, the coolant, the moderator, and neutron-absorbing materials used to control the power level. A main distinction between different reactor types lies in the differences in the choices of fuel, coolant, and moderator.

Fuels

There are few nuclides that can be used as reactor fuels. The paucity of possible candidates can be seen by examining the properties of the naturally occurring heavy elements:

[2] The fast breeder reactor program was subsequently sharply reduced, with the centerpiece of the U.S. fast breeder reactor program, the Clinch River Breeder Reactor, abandoned in 1975.

[3] There has been speculation about a quite different sort of molten-salt reactor, driven by a proton accelerator. If pursued, this would represent a radical departure from the sorts of reactors that have been built to date.

- *Uranium (atomic number $Z = 92$).* This is the main fuel in actual use, especially ^{235}U which is fissile. In addition, ^{238}U is important in reactors, primarily as a fertile fuel for ^{239}Pu production, and ^{233}U could be used as a fissile fuel, formed by neutron capture in ^{232}Th.
- *Protactinium (atomic number $Z = 91$).* The longest-lived isotope of protactinium (^{231}Pa) has a half-life of 3.3×10^4 yr, and therefore there is essentially no Pa in nature.
- *Thorium (atomic number $Z = 90$).* Thorium is found entirely as ^{232}Th, which is not fissile (for thermal neutrons). It can be used as a fertile fuel for the production of fissile ^{233}U.

Between thorium ($Z = 90$) and bismuth ($Z = 83$), the isotope with the longest half-life is ^{226}Ra ($T = 1600$ years); therefore, there are no fuel candidates, quite apart from the issue of fissionability. By the time the atomic number is as low as 83, the threshold for fission is much too high for a chain reaction to be conceivable (see, e.g., Ref. [6, p. 574]). Thus, uranium and thorium are the only natural elements available for use as reactor fuels. In addition, ^{233}U and ^{239}Pu can be produced from capture on ^{232}Th and ^{238}U in reactors. This means that the nuclides listed in Table 6.1 exhaust the practical possibilities for reactor fuels. Of these, only ^{235}U is both fissile and found in nature in useful amounts.

Restricting consideration to uranium fuel, there are a number of options as to the form of the fuel. Reactors can operate over a considerable range of enrichments in ^{235}U. Enrichment to a concentration in the neighborhood of 4% is now somewhat typical in the light water reactors that account for most of today's nuclear generation (see Section 8.1.4), with a trend over time toward higher enrichments and greater burnup of the fuel. In today's heterogeneous reactors, the fuel is solid. For the most part, it is in an oxide form, as UO_2, but metallic fuel is a possibility and has been used in some reactors. The fuel usually is in cylindrical pellets with typical dimensions on the scale of 1 cm, but some designs for future reactors are based on fuel in submillimeter microspheres embedded in graphite, with the goal of enhanced ruggedness at high temperatures.[4]

Moderators

As discussed in Section 7.2, a moderator is required if the reactor is to operate at thermal neutron energies. This means that most operating reactors use moderators, with the fast breeder reactor the exception. The options for moderating materials are limited:

- *Hydrogen ($Z = 1$).* The isotopes ^{1}H and ^{2}H are widely used as moderators, in the form of light (ordinary) water and heavy water, respectively.

[4] In particular, this fuel is for high-temperature gas-cooled reactors (see Section 16.4.3).

- *Helium ($Z = 2$).* The isotope ^{4}He is not used, because helium is a gas and excessive pressures would be required to obtain adequate helium densities for a practical moderator; ^{3}He would be similarly excluded, but, in addition, it is a strong neutron absorber and would be unsuitable as a moderator.
- *Lithium ($Z = 3$).* The isotope ^{6}Li (7.5% abundant) has a large neutron-absorption cross section, making lithium impractical as a moderator.
- *Beryllium ($Z = 4$).* ^{9}Be has been used to a limited extent as a moderator, especially in some early reactors. It can be used in the form of beryllium oxide, BeO. However, beryllium is expensive and toxic.
- *Boron ($Z = 5$).* The very large neutron-absorption cross section in ^{10}B (20% abundant) makes boron impossible as a moderator.
- *Carbon ($Z = 6$).* Carbon in the form of graphite has been widely used as a moderator. It is important that the graphite be pure (i.e., be free of elements that have high absorption cross sections for neutrons).

There are no advantages in considering elements heavier than carbon. The effectiveness for moderation decreases with increasing mass, and there are no counterbalancing advantages in other properties. Again, therefore, there is a limited list of candidates: light water, heavy water, graphite, and beryllium. Any of these can be used with uranium enriched in ^{235}U. With natural uranium, it not possible to achieve a chain reaction with a light water moderator, but it is practical to use heavy water or graphite, both of which have high moderating ratios (see Table 7.2).

Coolants

The main function of the coolant in an electricity generating plant is to transfer energy from the hot fuel to the electrical turbine, either directly or through intermediate steps. During power plant operation, cooling is an intrinsic aspect of energy transfer. However, in a nuclear reactor, cooling has a special additional importance, because radioactive decay causes heat production to continue even after the reactor is shut down and electricity generation has stopped. It is still essential to maintain cooling to avoid melting the reactor core, and in some types of reactor accidents (e.g., the accident at Three Mile Island) cooling is the critical issue.[5]

The coolant can be either a liquid or a gas. For thermal reactors, the most common coolants are light water, heavy water, helium, and carbon dioxide. The type of coolant is commonly used to designate the type of reactor. Hence, the characterization of reactors as light water reactors (LWRs), heavy water reactors (HWRs), and gas-cooled reactors (GCRs).

[5] If the fuel is designed to operate at high enough temperatures, cooling can be provided by radiation from the fuel, and maintaining the flow of coolant would not be essential under accident conditions. However, at present no operating reactors are designed for such high temperatures.

The coolant may also serve as a moderator, as is the case for LWRs and HWRs. In gas-cooled reactors, the density of the coolant is too low to permit it to serve as the chief moderator, and graphite is used. Fast reactors, in which fission is to occur without moderation to thermal energies, usually use a coolant that has a relatively high atomic mass number (A).[6] Generically, these reactors are termed liquid-metal reactors, because the coolant is a liquid metal, most commonly sodium ($A = 23$).

Control Materials

As discussed in Section 7.5.2, control materials are needed to regulate reactor operation and provide a means for rapid shutdown. Boron and cadmium are particularly good control materials because of their high cross sections for the absorption of thermal neutrons. These control materials are usually used in the form of rods. Control rods for pressurized water reactors (PWRs) commonly use boron in the form of boron carbide (B_4C) or cadmium in a silver–indium alloy containing 5% cadmium. Control rods for boiling water reactors (BWRs) commonly use boron carbide [7, p. 715]. In addition, boron may be introduced into the circulating cooling water to regulate reactor operation.

8.1.4 World Inventory of Reactor Types

Reactor Sizes

The earliest reactors had generating capacities well below 100 MWe, but there was a rapid transition to 1000-MWe reactors and larger. The move to a larger size was motivated by the desire to capture economies of scale. Some analysts suggest that this escalation proceeded too rapidly, especially in the United States, and was responsible for some of the difficulties encountered in achieving short construction times and reliable operation.

The mean capacity of all reactors in operation worldwide in 2003 was about 820 MWe (see Table 2.1). At the extremes, a class of older British gas-cooled reactors have capacities of 50 MWe, whereas four PWRs that went into operation in France in 2000 each have a capacity close to 1450 MWe [8].[7] While most reactors built in recent years—including in France, Japan, South Korea, the United States, and the United Kingdom (for the one reactor in 1995)—are large, considerable attention is being given to smaller reactors. Although going to smaller reactors means sacrificing economies of

[6] It is also possible to use helium as a coolant because there the helium gas is not dense enough to be an effective moderator.

[7] Four 50-MWe units at the Calder Hall power plant in the United Kingdom were officially shut down on March 31, 2003 after operations that began in the late 1950. (Three of these units had actually suspended operations in 2001, but remained listed as "operating" in standard tables [9]). Four other 50-MWe reactors remain in operation at the Chapelcross plant.

scale, some advantages can be regained if a number of identical units are placed at the same site. (Questions of future reactor size are discussed further in Section 16.1.3.)

Types of Reactors

A variety of different reactors are in use in the world today, although there was greater diversity in the early days of reactor design. Table 8.1 lists the types of nuclear power plant in operation in late 2003 as well as those reported to be under construction or on order [10]. The dominant reactor is the light water reactor (LWR), which uses ordinary water as both the coolant and moderator and enriched uranium in UO_2 pellets as the fuel. There are two types of light water reactor: the pressurized water reactor (PWR) and the boiling water reactor (BWR). Together, they account for 88% of the world's present generating capacity and 85% of the capacity nominally being built or on order. The main types of reactors in past or present use for electricity generation are as follows:

PWR. The pressurized water reactor accounts for almost two-thirds of all capacity and is the only LWR used in some countries, for example France, the former Soviet Union, and South Korea.

BWR. The boiling water reactor is a major alternative to the PWR and both are used, for example, in the United States and Japan.

Table 8.1. World totals for nuclear reactors in commercial operation and under construction, November 2003, classified by reactor type: number of reactors and capacity (in GWe).

	Number		Capacity (GWe)					First
Type	Oper	Cons	Oper	Cons	Usual Fuel[a]	Moderator	Coolant	Developed
PWR	263	18	236.0	16.0	UO_2 enr	H_2O	H_2O	USA
BWR[b]	92	5	80.6	6.4	UO_2 enr	H_2O	H_2O	USA
PHWR[c]	39	8	19.3	3.1	UO_2 nat	D_2O	D_2O	Canada
GCR	26	0	10.9	0.0	U, UO_2[d]	C	CO_2	UK
LGR	17	1	12.6	0.9	UO_2 enr	C	H_2O	USA/USSR
LMFBR	3	0	1.0	0.0	$UO_2 + PuO_2$	None	Liq Na	Various
TOTAL	440	32	360	26				

[a]Fuel designations: enr = enriched in ^{235}U, nat = natural.

[b]The listing for BWRs includes two ABWRs in operation and four under construction.

[c]Includes one 148-MWe HWLWR in Japan.

[d]Natural U used for GCR; enriched UO_2 used for AGCR.

Source: Capacity data are from Ref. [10]. Fuel and country data are from Ref. [11, p. 67].

ABWR. The advanced boiling water reactor incorporates improvements over earlier BWRs. It is in use in Japan, with additional units under construction in Japan and Taiwan.

PHWR. The pressurized heavy water reactor uses heavy water for both the coolant and moderator and operates with natural uranium fuel. It was developed in Canada and is commonly referred to as the CANDU.[8] Other countries with CANDU units in operation include India, South Korea, and Argentina.

GCR. The gas-cooled, graphite-moderated reactor uses a CO_2 coolant and a graphite moderator. Its use is limited to the United Kingdom; it is sometimes known as the Magnox reactor. A larger second-generation version is the advanced gas-cooled, graphite-moderated reactor (AGCR).

LGR. The light-water-cooled graphite-moderated reactor uses water as a coolant and graphite (in addition to water) for moderation. The world's major currently operating LGRs are the RBMK reactors in the former Soviet Union (11 in Russia and 2 in Lithuania).[9] There were four such units at the Chernobyl plant at the time of the accident there, but they have all been shut down.

HTGR. The high-temperature gas-cooled reactor uses helium coolant and a graphite moderator. The only HTGR that had been operating in the United States (Fort St. Vrain) has been shut down, and there are no HTGRs being used elsewhere for electricity generation, although active studies of variants of the HTGR are underway.

LMFBR. The liquid-metal fast breeder reactor uses fast neutrons and needs no moderator. A liquid metal is used as coolant, now invariably liquid sodium. There are only two LMFBR reactors in operation (one each in France and Russia).[10]

HWLWR. The heavy-water-moderated, light-water-cooled reactor is an unconventional variant of the heavy water reactor, and only one has been in recent operation, a 148-MWe plant in Japan. A new 700-MWe version of the HWLWR is being designed in Canada, the ACR-700 (see Section 16.2.2).

The dominance of light water reactors, both for plants in operation and those under construction, is seen in Table 8.1. These reactors were first developed in the United States, in both the PWR and BWR configurations, and have become the reactors of choice in almost all other major nuclear

[8] This acronym stands for Canadian deuterium uranium and has an obvious double meaning.

[9] In addition, there are four 11-MWe LGRs in Russia.

[10] In addition, the 246-MWe Monju reactor in Japan is listed by the IAEA as connected to the grid, but it has been shut down since 1995 after operating for only a few months [8].

countries. The main exceptions are Canada, the United Kingdom, the former Soviet Union (FSU), and India. Even in the United Kingdom and the FSU, the most recently completed reactors are PWRs.

The number of reactors under construction or on order as of late 2003 (32) was small compared to the number in operation (440). The average capacity of these reactors is about 830 MWe—very close to the average for operating plants. They range in size from four 202-MWe PHWRs being built in India [12] to two ABWRs in Japan with capacities near 1300 MWe.

History of Commercial Reactor Development

After World War II, the leading countries in nuclear reactor development were the United States, the United Kingdom, Canada, and the Soviet Union. Each went in a different direction.

The first U.S. power reactors, beyond plutonium-producing or experimental reactors, were built for submarines, not for civilian electricity generation. The earliest were a PWR for the submarine *Nautilus*, commissioned in 1955, and a sodium-cooled reactor for the submarine *Seawolf*. The *Seawolf* reactor had difficulties, and sodium-cooled reactors were abandoned by the navy in 1956 in favor of light water reactors [13, p. 423]. The navy PWR program provided the foundation for the development of PWRs for electricity generation, starting with the 60-MWe reactor at Shippingport, Pennsylvania, in 1957.

As was noted in Section 2.3.2, during the 1950s a varied array of reactors were ordered in the United States. These even included a small fast breeder reactor (Fermi I) in Michigan, which went into operation for a few years starting in 1966. However, after 1967, the only commercial power reactors put into operation in the United States have been PWRs and BWRs, with the sole exception (in 1979) of the trouble-plagued Fort St. Vrain HTGR in Colorado, which has since been shut down. The commercial BWRs were an outgrowth of a program of experimental BWR development carried out in the mid-1950s at Argonne National Laboratory.

The United Kingdom and Canada followed routes that did not require enriched uranium. The United States had a monopoly on uranium enrichment at the time, and although it presumably would have provided enriched uranium to such close allies, there may have been a reluctance on their part to become dependent. The United Kingdom program began very early, with two 50-MWe reactors at Calder Hall in 1956. These were GCRs, with graphite moderation and CO_2 cooling. They differed from most later reactors in the world in that they used uranium metal for the fuel, not uranium dioxide (UO_2). They gained the name Magnox, because the fuel pin cladding material was a magnesium alloy called Magnox [11, p. 165]. The GCRs that were built later had increasing size, up to 420 MWe. From the mid-1970s, with one exception, the few new plants brought on line in the United Kingdom have been AGCRs in the

600–700 MWe range. Like the Magnox reactors, they use graphite moderation and CO_2 cooling, but their fuel is enriched UO_2.

On the whole, after a fast start, the British reactor program has moved fitfully, with indecision abetted by North Sea oil and natural gas. After 1989, only one new reactor was put into commercial operation in the United Kingdom, the 1188-MWe Sizewell B reactor in 1995 [8]. Interestingly, it is a PWR, selected after prolonged study, adding further to the dominance of LWRs in the world nuclear picture.

The Canadian nuclear program offers the main alternative to the LWR among reactors now in operation or under construction. This program has involved only one type of reactor, the pressurized heavy water reactor (PHWR) known as the CANDU (Canadian deuterium uranium). Use of a deuterium moderator enabled Canada to use natural, rather than enriched, uranium. This was an attractive option for a country that had sophisticated scientific and engineering capabilities, including experience with heavy water reactors gained during World War II, but no enrichment facilities. The larger CANDU reactors (greater than 600 MWe) had the best cumulative capacity factors of any reactor type, as of the end of 2001 [14].

The PHWR has made substantial inroads outside of Canada, in particular in India, South Korea, and Argentina, with smaller programs elsewhere. In Canada, these reactors vary relatively little in size, starting at 525 MWe and most recently built at 881 MWe. India has emulated the Canadian example of reliance on PHWRs, although initially at the smaller size of about 200 MWe. The first of these were constructed under Canadian supervision, but, subsequently, India has assumed independent responsibility. South Korea has fewer PHWRs than India (four compared to fourteen) but they are larger—each about 650 MWe.

The other major dissenter from LWRs had been the Soviet Union, but this is now changing. The Soviet Union began with six 100-MWe light-water-cooled, graphite-moderated reactors (LGR) put into operation from 1958 to 1963 [8]. These were followed by larger LGR reactors with capacities of about 950 MWe, the so-called RBMK reactors among which were the Chernobyl reactors. These LGRs were built to produce both plutonium and electricity, as was the now-closed Hanford-N reactor in Washington. The Soviet Union also developed its own PWRs, the WWER series, and these are now the most numerous reactors in Russia and are widely used in much of Eastern Europe [8].

Overall, what worldwide growth there is in nuclear power is now primarily in the form of LWRs, with the Canadian PHWR as the only other significant player. It is not clear whether this is because of intrinsic technical and economic advantages of water-cooled reactors or because of historical and commercial forces. For the future, there is considerable interest in new HTGRs as a relatively near-term option and in liquid-metal-cooled reactors for the longer term (see Chapter 16), but the dominance of water-cooled reactors, and particularly LWRs, has not yet been seriously challenged.

8.2 Light Water Reactors

8.2.1 PWRs and BWRs

The two types of LWR in use in the world are the pressurized water reactor (PWR) and the boiling water reactor (BWR). The difference between them, as embodied in the names, is in the condition of the water used as coolant and moderator. In the PWR, the water in the reactor vessel is maintained in liquid form by high pressure. Steam to drive the turbine is developed in a separate steam generator. In the BWR, steam is provided directly from the reactor. These differences are brought out in the schematic representation of the two reactor types in Figure 8.1.

Under typical conditions in a PWR, temperatures of the cooling water into and out of the reactor vessel are about 292°C and 325°C, respectively, and the pressure is about 155 bar [7, p. 713].[11] For the BWR, typical inlet and outlet temperatures are 278°C and 288°C, respectively, and the pressure is only about 72 bar. The high pressure in the PWR keeps the water in a condensed phase; the lower pressure in the BWR allows boiling and generation of steam within the reactor vessel.

Neither the PWR nor BWR has an overwhelming technical advantage over the other, as indicated by the continued widespread use of both. Among the major LWR users, the United States, Japan, and Germany use both types, while France, South Korea, and Russia use PWRs almost exclusively in the LWR part of their programs. Overall, the number of PWRs in operation is significantly greater than the number of BWRs, and PWRs also have a lead in reactors listed as under construction. The future is not clear-cut, however. For example, in Japan, all three reactors under construction in 2003 were BWRs, including two ABWRs. In the following discussion, we will emphasize the PWR in giving specific illustrations but will consider both to some extent.

8.2.2 Components of a Light Water Reactor

The containment structure and enclosed components for a typical PWR and a typical BWR are shown schematically in Figures 8.2 and 8.3.[12] The most conspicuous difference between them is the absence of a steam generator in the BWR. At the heart of the reactors, literally and figuratively, is the reactor core, contained within the reactor pressure vessel. The pressure vessel encloses three vital components:

- The fuel itself, contained in many small fuel rods comprising the reactor core.
- The surrounding water, acting as coolant, moderator, and heat-transfer agent.

[11] 1 bar $= 10^5$ newton/m^2 $= 0.987$ atm.

[12] These diagrams are copied from a draft version of Ref. [15].

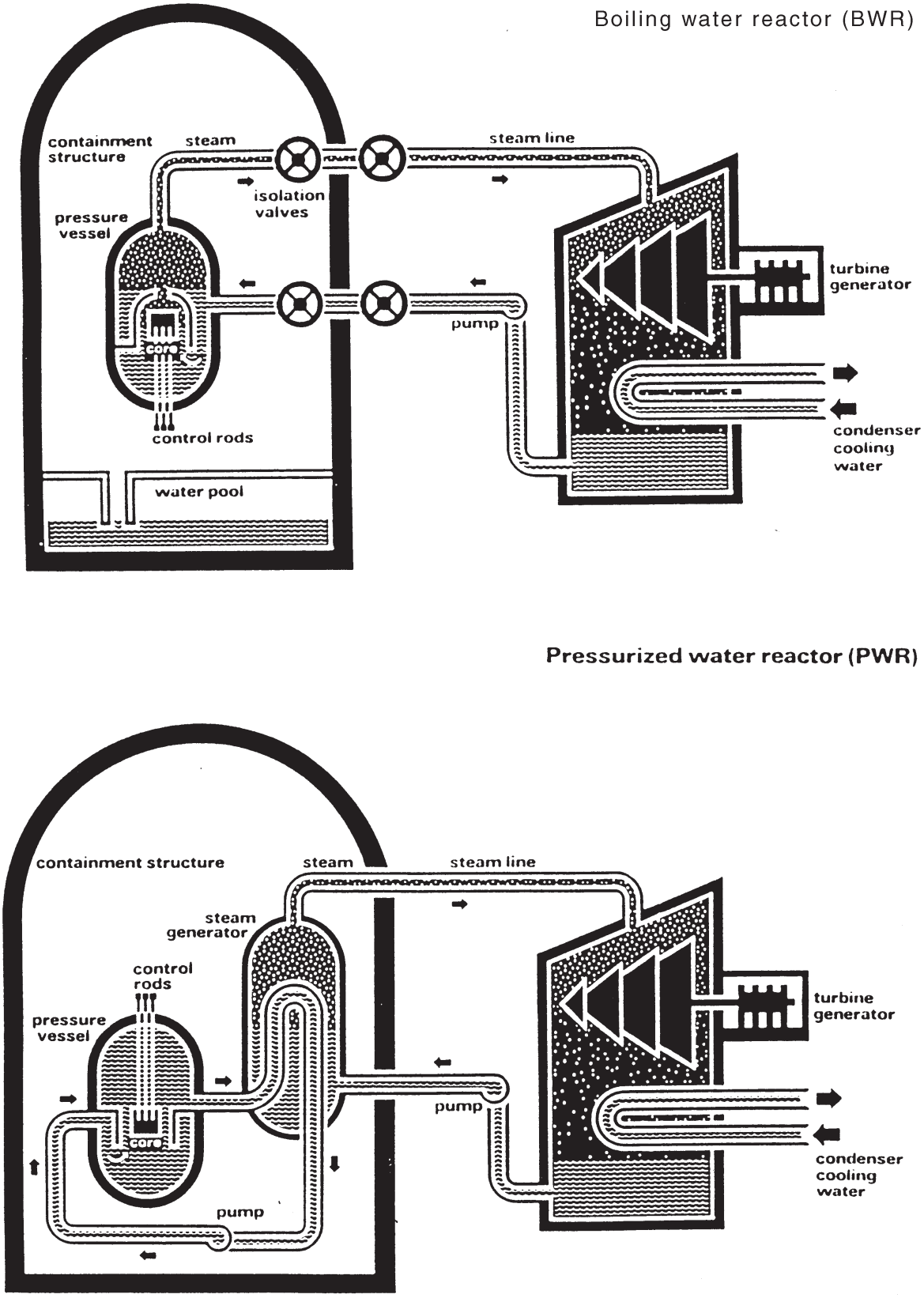

Fig. 8.1. Schematic representation of BWR and PWR systems, emphasizing the difference in the means for providing steam to the steam turbine. [Adapted from figures provided by the U.S. Council on Energy Awareness.]

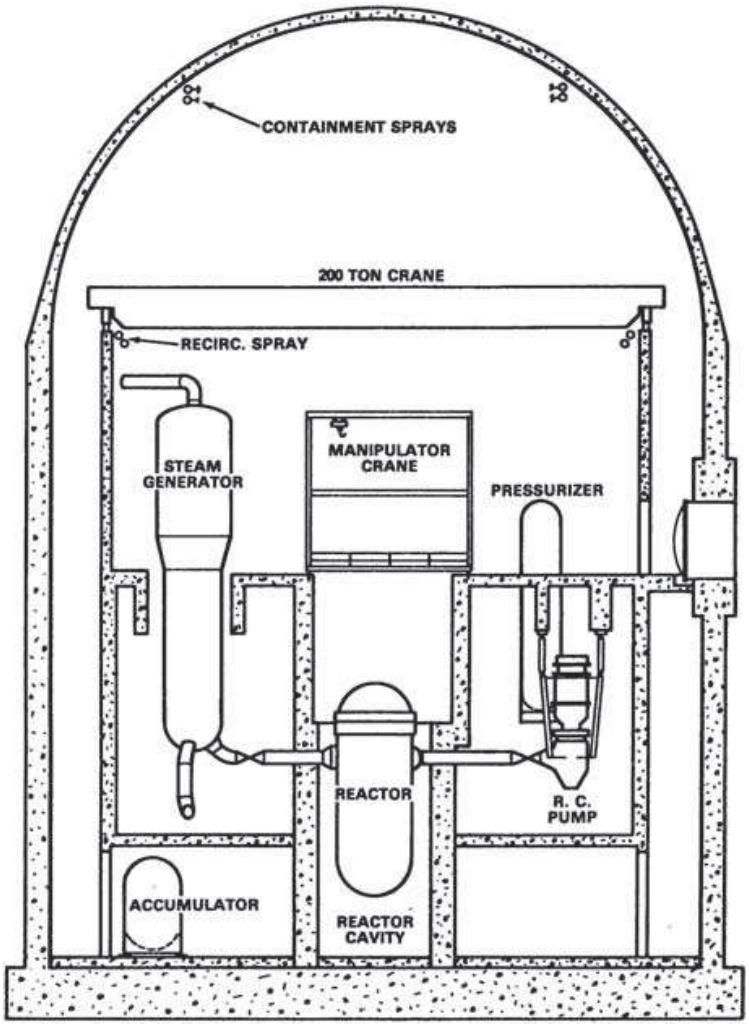

Fig. 8.2. Schematic diagram of containment building with enclosed reactor vessel and steam generator for an illustrative PWR: the 781-MWe Surry plant. The output of the steam generator drives the turbines, external to the containment building. (From [15, p. 4–4].)

- Control rods, used to establish the reactivity at the desired level and shut the reactor down in case of an emergency.

The reactor pressure vessel is a massive cylindrical steel tank. Typically for a PWR, it is about 12 m (40 ft) in height and 4.5 m (15 ft) in diameter [16, p. 304]. It has thick walls, about 20 cm (8 in.), and is designed to withstand pressures of up to 170 atm.

A second major component, or set of components, is the system for converting the reactor's heat into useful work. In the BWR, steam is used directly from the pressure vessel to drive a turbine. This is the step at which electricity is produced. In the PWR, primary water from the core is pumped at high pressure through pipes passing through a heat exchanger in the steam generators. Water fed into the secondary side of the steam generator is converted into steam, and this steam is used to drive a turbine. The secondary loop is also

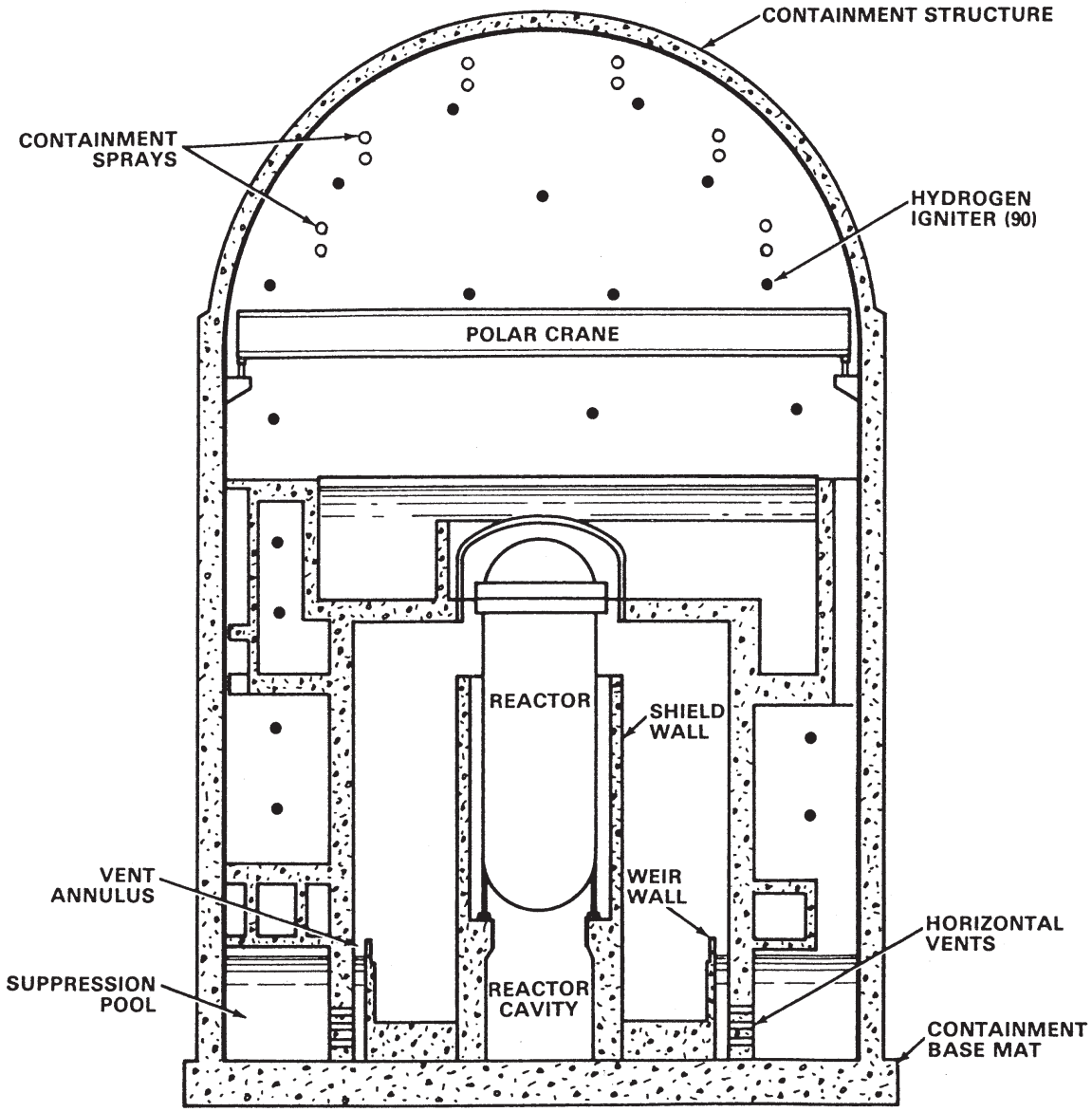

Fig. 8.3. Schematic diagram of containment building with enclosed reactor vessel for an illustrative BWR: the 1142-MWe Grand Gulf plant. Steam from the reactor vessel drives the turbines, external to the containment building. (From [15, p. 4–40].)

closed. The exhaust steam and water from the turbine enter a condenser and are cooled in a second heat exchanger before returning to the steam generator.

The cold side of the condenser heat exchanger represents the tertiary loop for the PWR. In principle, this loop need not be closed, and the condenser cooling water could circulate to and from a river or the ocean. More commonly, this condenser output is circulated through a cooling tower, where it is cooled by evaporation and the ultimate heat sink becomes the air. The part of the water that is lost as steam from the cooling tower is made up by water drawn, say, from a river.

Cooling towers became represented as ominous-looking symbols during the Three Mile Island accident, but they themselves are quite benign. The steam

that may be seen rising from a cooling tower is isolated from water passing through the reactor core by the heat exchangers and, thus, is not a source of radioactive emissions, even if the primary cooling water is slightly radioactive. Such cooling towers are not unique to nuclear power plants and are used in other facilities, including coal-fired plants, where it is necessary to dissipate large amounts of heat.

The pressure vessel and the steam generators are contained within a massive structure, the *containment building*, commonly made of strongly reinforced concrete. In some designs, the concrete containment is lined with steel; in others, there is a separate inner steel containment vessel. The containment is intended to retain activity released during accidents and is believed capable of protecting a reactor against external events including some airplane impacts (see Section 17.5.5). In the Three Mile Island accident, the containment very successfully retained the released radioactivity, although it may be noted that the physical structure was not put fully to the test because there was no explosion or buildup of high pressures. At Chernobyl, there was no containment, with disastrous results. In principle, were the possibility of an accident made negligibly small with improved reactor design and if terrorist attacks were not a concern, a containment would be unnecessary. Nonetheless, it is widely considered to be an important safety feature, providing an additional layer of protection.

8.2.3 PWR Reactor Cores

We consider here the specific characteristics of a reactor core based on a Westinghouse Corporation PWR design, but the gross features are similar for all large LWRs.[13] The reactor fuel is in the form of cylindrical uranium oxide (UO_2) pellets, about 0.8 cm in diameter and 1.35 cm in length. The pellets are placed in tubes—called fuel rods or fuel pins—made of zircaloy, a zirconium alloy (98% Zr, 1.5% Sn, and small amounts of other metals [16, p. 234]) selected on the basis of structural strength and low neutron absorption. The zircaloy cladding is thin, 0.06 cm. The fuel rod is typically 3.7 m (12 ft.) in length and 1.0 cm in diameter. There is some free space within the fuel rod to allow for the expansion of the fuel pellets and to accommodate gaseous fission products such as xenon and krypton. The fuel expansion is due to both increased temperature and the replacement in the fuel of one uranium atom by two fission-product atoms. Noble gases produced as fission products may be trapped as bubbles in the fuel or may escape from the fuel into the gap between the fuel and cladding.

A 17×17 array of fuel rods forms a "fuel bundle" or *assembly*. Although this would allow 289 fuel rods in an assembly, positions are left open in some

[13] Most of the detailed numbers in this paragraph are based on Westinghouse Corporation information, as reported in Ref. [16], especially Table 9.1. They are specific to this particular Westinghouse design; others differ in detail.

assemblies for the insertion of control rods or measuring instrument rods. The total core contains 193 assemblies and 50,952 fuel rods. Some 53 of the assemblies have spaces for clusters of 20 control rods, which can be moved in and out within the assembly. These control rods are made from a silver–indium–cadmium alloy.

Fuel assemblies are replaced periodically, but not all in the same period of reactor refueling. Thus, although a particular fuel assembly may remain in the reactor for 3 years, one-third of the core can be changed every year. (Recently, cores have been designed to have a longer time between fuel changes.) As the ^{235}U is consumed in the reactor, the reactivity of the fuel decreases. This is compensated for in several ways. Burnable poisons in the fuel are consumed, control rods that limit the reactivity are partly withdrawn, or the soluble poison concentration in the cooling water is reduced.

8.3 Burners, Converters, and Breeders

8.3.1 Characterization of Reactors

As discussed in Section 7.1.1, the condition for a chain reaction is that for every neutron initiating fission in one generation, one or more neutrons initiate fission in the next generation. If, in addition, another fissile nucleus is produced for every ^{235}U atom consumed, then there is no decrease in the amount of nuclear fuel available. This is the principle of the breeder reactor.

The *conversion ratio* C (or *breeding ratio* B) is defined as the ratio of the rate of production of fissile nuclei to the rate of consumption of fissile nuclei [see Eq. (7.18)].[14] For uranium fuel, this is the ratio of ^{239}Pu produced to ^{235}U consumed. If the conversion ratio is small, the reactor is sometimes called a *burner*; if the conversion ratio is between about 0.7 and 1.0, it is commonly called a *converter*; and if it exceeds unity, the reactor is called a *breeder* (see, e.g., Ref. [16, p. 458]).

8.3.2 Achievement of High Conversion Ratios in Thermal Reactors

Difficulty of Reaching a Conversion Ratio of Unity with ^{235}U

As discussed in Section 7.4, the limiting condition for a breeder reactor is that the conversion ratio, C, be at least 1. This means that the number η_o of neutrons produced for each neutron absorbed in ^{235}U, must be two or more.[15] For thermal neutrons absorbed in ^{235}U, $\eta_o = 2.075$ (see Table 7.1). Were there no losses, this would suffice for breeding: 1 neutron for continuing the

[14] Sometimes, a distinction is made in terminology, with conversion ratio used when $C < 1$ and breeding ratio used when $C > 1$.

[15] The relation among η_o, η, and the commonly cited parameter ν, is discussed in Section 7.1.2.

chain reaction, 1 neutron for production of ^{239}Pu, and 0.08 neutrons free to be "wasted." Such efficient utilization cannot be achieved, however, because there is absorption in the moderator and other nonfuel materials, as well as escape of neutrons from the core. Therefore, a ^{235}U-fueled thermal breeder reactor is not practical. Nonetheless, the production of ^{239}Pu is significant in uranium-fueled thermal reactors because its fission increases the overall energy output from the reactor fuel beyond that gained from the ^{235}U alone.

Potential of ^{233}U for a Thermal Breeder Reactor

The number of neutrons produced is significantly higher for ^{233}U ($\eta_o = 2.296$) than for ^{235}U, and there have been serious suggestions for developing ^{233}U thermal breeders. These date to as early as 1945, in work done by Eugene Wigner's group in Chicago [3, Chapter 6]. A cycle is envisaged in which ^{233}U is produced initially in a reactor with ^{235}U as the fissile fuel and ^{232}Th as the fertile fuel. Subsequently, a ^{233}U–^{232}Th cycle could, in principle, be self-sustaining. Not only is η_o higher for ^{233}U than for ^{235}U, but the capture cross section is significantly higher for ^{232}Th than for ^{238}U at thermal energies, making the conversion ratio higher than for a cycle based on ^{238}U (see Table 7.1).

However, although thermal breeders based on ^{233}U are, in principle, possible and preliminary exploratory work toward their development was done at Oak Ridge National Laboratory in the 1950s, thermal breeders were abandoned in favor of the fast breeder reactor. It is conceivable that interest in thermal breeders could revive, but, to date, the few breeder reactors that have gone into commercial operation have all been fast breeders.

High Conversion Ratios Without Breeding

Before turning to fast breeder reactors, it may be noted that even if breeding is not achieved with thermal reactors, a high conversion ratio can still be desirable. One motivation could be plutonium production. Another motivation is the extension of fuel resources. As the conversion ratio increases, the energy output increases for a given original ^{235}U content. A high conversion ratio means a high ratio of capture in ^{238}U to absorption in ^{235}U. This must be accomplished without losing criticality. Greater losses of neutrons to ^{238}U can be compensated for by smaller losses in the moderator and reactor structure.

The use of carbon instead of light water as a moderator is favorable on two counts if a high conversion ratio is desired (in addition to the advantage that with a carbon moderator it is possible to use natural uranium). Because carbon is a less effective moderator than water, more collisions are required to reach thermal energies; therefore, there is more possibility of neutron capture in ^{238}U at intermediate energies. Further, because of the low neutron-capture cross section in ^{12}C (see Section 7.2.2), the loss of thermal neutrons to ab-

sorption will be less for carbon than for light water.[16] Together, this means a higher conversion ratio.

The same general arguments apply to heavy water reactors. The conversion ratio is higher in a heavy water reactor than in a light water reactor due to less effective moderation in heavy water and a lower neutron-capture cross section. For both graphite-moderated and heavy water reactors, there have been suggestions that the ^{232}Th–^{233}U cycle be used, to further increase conversion and extend the life of the uranium fuel, even without breeding.

It may be noted that reactors designed with production of plutonium for weapons in mind, either as the main or as an auxiliary function, have been mostly graphite moderated. Examples include the Windscale plant in England, the plutonium production reactors at Hanford, and the RBMK reactors built in the USSR. The five heavy water reactors at the Savannah River (South Carolina) complex for plutonium production are the major exception.

8.3.3 Fast Breeder Reactors

Plutonium as Fuel for Fast Breeders

A thermal breeder reactor is not possible using ^{239}Pu due to the high ratio α of the capture cross section to the fission cross section for thermal neutrons.[17] However a fast breeder reactor is possible. It relies on a chain reaction in which the neutrons are not thermalized but instead produce fission at relatively high energies. If ^{239}Pu is the fissile fuel, the cycle uses ^{238}U as the fertile fuel and ^{239}Pu is both consumed and produced in the reactor. The cycle is started using ^{239}Pu produced in a uranium-fueled reactor.

Although most of the fission neutrons are emitted with energies above 1 MeV, they can lose energy through inelastic scattering in ^{239}Pu and ^{238}U. Fission is therefore produced at energies extending over a very broad energy region, from above 1 MeV to below 1 keV. In the high-energy part of this region the conditions are very favorable for breeding. For example, for 1.0-MeV neutrons on ^{239}Pu, $\sigma_f = 1.7$ b, and the ratio (α) of the capture cross section to the fission cross section is less than 0.03. The low value of this ratio means that almost all absorption in ^{239}Pu leads to fission. With about three neutrons per fission at 1 MeV, breeding with ^{239}Pu is readily achieved, with 1 neutron for continuing the chain reaction, 1^+ for breeding, and 1^- for losses. At lower

[16] In terms of the formalism introduced in Chapter 7, this means that carbon leads to a lower resonance escape probability p and a higher thermal utilization factor f than does light water [see the four-factor formula, Eq. (7.5)]. Criticality can still be maintained ($k = 1$), and C will be greater (see Eq. (7.23)).

[17] For ^{239}Pu, $\alpha = 0.360$ and $\eta_o = 2.115$ at thermal neutron energies. This may be compared to the values for ^{233}U: $\alpha = 0.086$ and $\eta_o = 2.296$. For breeding, the crucial condition is that $\eta_o > 2$. Therefore ^{233}U is significantly better than ^{239}Pu at thermal energies.

neutron energies, the cross section σ_f remains high, above 1.5 b for the most part, but the ratio α rises—reaching about 0.3 at 40 keV—greatly reducing the number of "surplus" neutrons [17, p. 753]. The actual conversion ratio depends on the neutron reactions over the full range of "incident" neutron energies, with the lower-energy neutrons contributing even if not as effectively as the high-energy neutrons.

To avoid thermalization of the neutrons, fast breeder reactors use a coolant with a relatively high mass number A. Liquid metals have the best combination of high A and favorable heat-transfer properties, and the fast breeder reactors in actual use have been *liquid-metal fast breeder reactors* (LMFBR). The standard choice for the coolant is liquid sodium (^{23}Na).

The fuel is made of pellets of mixed plutonium and uranium oxides, PuO_2 (about 20%) and UO_2 (about 80%). Uranium depleted in ^{235}U is commonly used, it being available as a residue from earlier enrichment. The fission cross section for ^{239}Pu is between 1.5 and 2.0 b over virtually the entire fast-neutron region (from 10 keV to 6 MeV), whereas for ^{238}U it is below 0.2 b for $E_n < 1.4$ MeV and falls rapidly at lower E_n (see Figure 6.1 and Ref. [17]). Therefore, fission in ^{239}Pu is much more probable than fission in ^{238}U. The most probable fast-neutron reactions in ^{238}U are inelastic scattering, which produces lower energy neutrons, and capture, which produces ^{239}Pu.

Status of Fast Reactor Programs

The main incentive for the development of fast breeder reactors is the extension of uranium supplies. A fast breeder economy would extract much more energy per tonne of uranium than is obtained from other reactors (e.g., the LWRs). Further, with more energy per unit mass, it becomes economically practical to use more expensive uranium ores, increasing the ultimate uranium resource. A secondary incentive is the easing of the waste disposal problem if plutonium and uranium (and possibly other actinides) are reused in a closed cycle rather than disposed of as waste.

However, during the 1980s and 1990s, growth of nuclear power fell far short of earlier expectations, there was little pressure on uranium supplies, and interest in fast breeder reactors declined. Further, the initial fast breeder reactors proved to be more expensive than alternatives, such as the LWR or HWR. Particularly in the United States, there was also the concern that the large-scale use and availability of ^{239}Pu might increase dangers from terrorism and nuclear weapon proliferation (see Section 9.4.2 and Chapter 17). These factors made breeder programs a vulnerable target, at a time when there was significant opposition to any projects to advance nuclear power. Nonetheless, some development of breeders has continued, in part to maintain the technology as insurance against future needs.

In a later turn of the argument, it has been pointed out that a liquid-metal fast reactor (LMR) can be used to *destroy* unwanted plutonium and other heavy elements which are in weapons stockpiles or nuclear wastes. In

this reversal of motivation, the LMR would be used to consume plutonium, rather than to produce plutonium as a fuel. There is flexibility in this, because as LMR technology and facilities are developed, they could be turned to either purpose. However, if the driving fear is concern about misuse of plutonium, it may appear more desirable to dispose of plutonium from weapons stockpiles in ways that do not involve expanding a technology that is closely related to potential plutonium production.

France had led in the development and deployment of breeder reactors, with two completed reactors, the 233-MWe Phenix, put into operation in 1973, and the 1200-MWe SuperPhenix at Creys-Malville, which first generated electricity in 1986 but was finally shut down in 1998 after a troubled history of recurring technical difficulties. Small breeder reactors in Great Britain and Kazakhstan were also shut down in the 1990s, and there now remain only two LMFBRs operating to produce electricity, Phenix in France and a 560-MWe reactor in Russia. However, interest in breeder reactors remains in a number of countries, with ongoing and new activity, and it would be premature to write breeder reactors off as an option for the future.[18]

The United States breeder reactor program has been marked by indecision and opposition, with successive projects started and abandoned. It had started auspiciously, with the operation of the Experimental Breeder Reactor (EBR-I) in Idaho. On December 20, 1951 EBR-I generated the first electricity from a nuclear reactor produced anywhere—enough for four light bulbs. Its output was shortly thereafter increased to 0.1 MWe [19]. Two larger, fast reactors made important research contributions in subsequent years—the Experimental Breeder Reactor II (EBR-II) in Idaho and the Fast Flux Test Reactor (FFTR) in Washington—but both of these projects have been terminated. The most ambitious breeder proposal in the United States was for the Clinch River Breeder Reactor (CRBR)—a project that was under active consideration in the 1970s but was terminated by Congress in 1983.

Following the end of the CRBR project, a major fast reactor development program was undertaken at Argonne National Laboratory as part of the *integral fast reactor* plan (see Section 16.5). Advocates of this program stressed its potential to offer a high degree of safety against reactor accidents and to consume nuclear wastes in an on-line process. The breeding potential was often secondary in these arguments, and the planned LMR need not have operated as a breeder, namely with a conversion ratio greater than unity. Nonetheless, the basic configuration of the system was similar to that of a breeder reactor. Culminating several years of debate, most funding for this project was terminated in the mid-1990s.

[18] Current breeder projects include the 246-MWe Monju breeder reactor in Japan, which is now shut down due to operating difficulties but which may be restarted, a 750-MWe LMFBR being planned in Russia (Beloyarsk-4) with completion scheduled for 2009, a prototype 500-MWe breeder being planned in India (Kalpakkam), and a 65-MWt fast neutron reactor being built in China [18].

Despite these difficulties, the long-term argument for breeder reactors remains and breeder development may intensify, especially if the proliferation problems can be satisfactorily addressed. Some of the reactors now being considered under the program of the Generation IV International Forum (see Chapter 16) are designed to operate with a fast-neutron spectrum and thus have the potential of being operated as breeder reactors.

8.4 The Natural Reactor at Oklo

A remarkable discovery was made in 1972 by French scientists analyzing uranium extracted from the Oklo uranium mine in Gabon. The uranium was depleted in ^{235}U, sometimes by large amounts, although, normally, the isotope ratios in uranium are nearly constant over the surface of the Earth. It was soon suspected and then demonstrated that this isotopic anomaly was due to a natural uranium chain reaction occurring more than a billion years ago. Conclusive evidence in support of this explanation was provided by the relatively high abundance of intermediate-mass nuclei, the rare earths, which are characteristic fission products but are not normally found in large abundance in nature (see, e.g. Ref. [20]).

The scenario, as it has been recreated, puts the event about 1.8 billion years ago. At that time, the isotopic abundance of ^{235}U exceeded its present value by the factor $\exp(\Delta\lambda\, t)$, where $\Delta\lambda$ is the difference in the decay constants of the two isotopes and t is the time since the event. The decay constants of ^{235}U and ^{238}U are 0.985×10^{-9} yr^{-1} and 0.155×10^{-9} yr^{-1}, respectively (see Table 2.1), giving $\Delta\lambda\, t = 1.49$ for $t = 1.8 \times 10^9$ yr. Therefore. the isotopic abundance of ^{235}U was 4.4 times greater at the time of the Oklo event than it is today, putting the enrichment at slightly above 3%. (This is strikingly close to the enrichment used in modern LWRs.) The intrusion of water, acting as a moderator, apparently initiated a chain reaction, which appears to have simmered for at least several hundred thousand years. In this model of what took place, the reaction did not occur earlier because the concentrated uranium deposits had been only recently formed by the leaching of rocks and the precipitation of the dissolved uranium.

Aside from having posed an intriguing scientific puzzle, with a very interesting explanation, the Oklo event is considered by some to have significance as a test of the motion of fission products through the ground. For the most part, these products have moved very little over a period of more than 1 billion years. This could have implications for the rate of movement of fission products in buried nuclear wastes. The Oklo example cannot be used as an all-embracing guide because differences in the chemical form of the product and in the type of rock formation may vitiate an extrapolation from Oklo to the behavior of an individual modern waste disposal site. However, Oklo illustrates that at least under some circumstances, radionuclides do not migrate appreciably from their initial location.

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