Chapter 11

Nuclear Power

Nuclear reactors allow us to produce enormous amounts of thermal energy through fission chain reactions with the need for relatively small amounts of fuel compared to conventional fossil-fuel combustion reactions. Production of electricity and propulsion of ships are two major needs in our modern society for large thermal sources. Both needs, for which nuclear energy has served as an alternative to fossil fuels, are discussed in this chapter.

11.1 Nuclear Electric Power

During the last half of the twentieth century, many types of nuclear reactors have been designed and built to convert the thermal power produced from a fission chain reaction into electrical power. Such *power reactors* are today an important source of electrical energy in many countries with limited native resources of fossil fuels to use in conventional fossil-fired power plants. For example, France produces over 70% of its electrical energy from fission energy. Table 11.1 shows how various countries depend on nuclear power. Although many countries with large reserves of fossil fuels available to fire conventional electrical power plants have suspended expansion of nuclear capacity, many other countries are continuing to build and plan for new nuclear power plants. The existing and future nuclear power plant capacities for various countries are also shown in Table 11.1.

11.1.1 Electricity from Thermal Energy

Most of the energy produced by fission reactions in a nuclear reactor is quickly converted to thermal energy. This heat can be converted to electrical energy in a variety of ways. By far the most common method used to produce electricity from a thermal energy source is to use the thermal energy to produce a hot pressurized gas which is then allowed to expand through a turbine causing it to turn. The rotating turbine shaft is common with the shaft of a generator which converts the energy of rotation into electrical energy. The most common gas or *working fluid* used to transfer the thermal energy to rotational energy of the turbine is steam. Such a *steam cycle* is illustrated in Fig. 11.1. In a conventional fossil-fueled power plant, the steam is generated in a boiler in which oil, natural gas, or, more usually, pulverized coal is burned. In a nuclear power plant, thermal energy produced in a reactor is used, either directly or indirectly, to boil water.

Table 11.1. Percent of 1998 electrical energy generated from nuclear power in different countries. Also shown is the nuclear electric power capacity existing, under construction, and planned.

Country or Region	Percent Electricity from Nuclear in 2000	Gross Generating Capacity							
		Operating at start 2001		Under Construction		Planned		Total	
		GW(e)	units	GW(e)	units	GW(e)	units	GW(e)	units
U.S.A.	19.8	101.62	104					101.62	104
France	76.4	65.83	59					65.83	59
Japan	33.8	45.22	53	3.31	3	2.32	2	50.85	58
Germany	30.6	22.21	19					22.21	19
Russia	14.9	21.24	29	3.00	3	6.00	6	30.24	38
UK	27	14.16	35					14.16	35
S. Korea	40.7	13.72	16	4.00	4			17.72	20
Ukraine	47.3	11.84	13	4.00	4			15.84	17
Canada	11.8	10.62	14					10.62	14
Sweden	39.0	9.81	11					9.81	11
Spain	27.6	7.80	9					7.80	9
Belgium	56.8	6.00	7					6.00	7
Taiwan		5.14	6	2.70	2			7.84	8
Bulgaria	45.0	3.76	6					3.76	6
Switzerland	38.2	3.34	5					3.34	5
Finland	32.1	2.76	4					2.76	4
India	3.1	2.72	14	1.00	2	4.88	10	8.60	26
Lithuania	73.7	2.60	2					2.60	2
Slovak R.	53.4	2.58	6	0.86	2			3.44	8
Czech R.	20.1	2.57	5	0.98	1			3.70	6
China	1.2	2.27	3	6.52	8	4.20	4	12.99	15
Brazil	1.9	1.97	2			1.31	1	3.28	3
S. Africa	6.6	1.89	2					1.89	2
Hungary	40.6	1.87	4					1.87	4
Mexico	3.9	1.49	2					1.49	2
Argentina	7.3	1.01	2	0.75	1			1.75	3
Romania	10.9	0.71	1	0.71	1	1.93	3	3.35	5
Slovenia	37.4	0.71	1					0.71	1
Netherlands	4.0	0.48	1					0.48	1
Pakistan	1.7	0.46	2					0.46	2
Armenia	33.0	0.41	1					0.41	1
Iran	0			2.29	2	1.52	4	3.81	6
Cuba	0			0.88	2	1.52	4	0.88	2

Source: IAEA PRIS database.

Although the steam turbines are the principal devices used to generate electricity, there is no reason why other hot gases cannot be used to turn a turbo-generator. Indeed, the hot combustion gases from burning natural gas are sometimes used directly in a gas turbine. Because combustion gases are much hotter than the steam used in conventional steam turbines, these direct fired gas turbines are much smaller (and hence less expensive) than steam turbines. Capital costs for direct-fired gas turbine units are less than those for nuclear plants, and gas turbines have historically been used as peaking units to supply electrical energy during periods of high electrical demand.

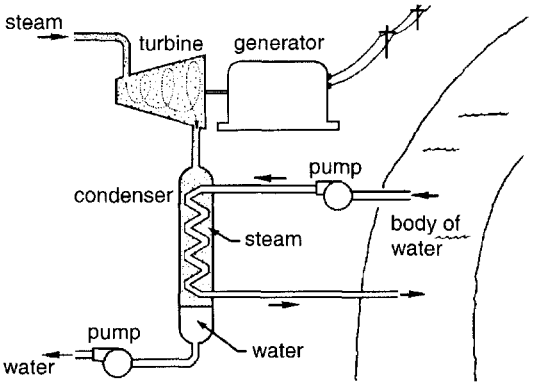

Figure 11.1. A source of steam is used to produce electricity.

11.1.2 Conversion Efficiency

Turbo-generator systems are *heat engines* that convert thermal energy to electrical energy. From the laws of thermodynamics, the maximum conversion efficiency of any heat engine is the *Carnot efficiency*, namely

$$\eta = \frac{T_{in} - T_{out}}{T_{in}}, \tag{11.1}$$

where T_{in} is the absolute temperature (K) of the gas entering the turbine and T_{out} is the absolute temperature of the gases leaving the turbine. Clearly, the higher the entering temperature and/or the lower the outlet temperature, more of the thermal energy is converted to electrical energy. The inlet temperature is limited by the water/steam pressure rating of the boiler or reactor vessel in a steam cycle, or by the temperature limitation of the turbine blades in a direct-fired gas turbine. The outlet temperature is usually limited by the ambient temperature of the cooling water used in the condenser of a steam cycle, while in a direct-fired gas turbine the exit temperature is determined by the exit pressure. An important measure of a power plant's performance is the conversion efficiency, i.e., the ratio of electrical power to thermal power, MW(e)/MW(t). In modern nuclear power plants conversion efficiencies of about 40% can be achieved, while fossil-fired units can achieve only slightly greater efficiencies. However, many older power plants have efficiencies in the range of 30-35%.

11.1.3 Some Typical Power Reactors

Many different designs for power reactors have been proposed and many different prototypes built. Most countries that have developed nuclear power started with

graphite or heavy-water moderated systems, since only these moderators allow criticality with natural uranium (0.711 wt% ^{235}U). However, most power reactors now use slightly enriched uranium (typically 3%). With such enrichments, other moderators, notably light water, can be used. The *steam supply system* for some important types of power reactors are illustrated in Figs. 11.2–11.7.

This section introduces different types of power reactors that have been used to generate electricity. Later in the chapter, the two most widely used types are discussed in detail.

Pressurized Water Reactor

Pressurized water reactors (PWR), Fig. 11.2, the most widely used type of power reactors, employ two water loops. The water in the primary loop is pumped through the reactor to remove the thermal energy produced by the core. The primary water is held at sufficiently high pressured to prevent the water from boiling. This hot pressurized water is then passed through a steam generator where the secondary-loop water is converted to high temperature and high pressure steam that turns the turbo-generator unit. The use of a two-loop system, ensures that any radioactivity produced in the primary coolant does not pass through the turbine.

Boiling Water Reactors

In a boiling water reactor (BWR) cooling water is allowed to boil while passing through the core. The steam then passes directly to the turbine. The low pressure steam leaving the turbine is then condensed and pumped back to the reactor. By having a single loop, the need for steam-generators and other expensive equipment in a PWR is avoided.

Heavy Water Reactors

In one design of a heavy water reactor (HWR), Fig. 11.4, pressurized heavy water in the primary loop is used to cool the core. The fuel is contained in pressure tubes through which the heavy water coolant passes. These pressure tubes pass through the moderator vessel, which is also filled also with heavy water. The heavy water in the primary loop then passes through steam generators to boil the secondary-loop light water. By using pressure tubes for the coolant, the need for an expensive high-pressure reactor vessel is avoided.

Another variation of the HWR uses light water in the primary loop, allowing it to boil as the coolant flows through the pressurized fuel tubes. In this design, no secondary water loops or steam generators are needed. The steam produced in the pressure tubes, after eliminating entrained moisture, flows directly to the turbo-generator

Gas Cooled Reactor

In a gas cooled reactor (GCR), Fig. 11.5, carbon dioxide or helium gas is used as the core coolant by pumping it through channels in the solid graphite moderator. The fuel rods are placed in these gas cooling channels. The use of graphite, which remains solid up to very high temperatures, eliminates the need for an expensive pressure vessel around the core. The hot exit gas then passes through steam generators.

In another design known as the high-temperature gas cooled reactor (HTGR), the fuel is packed in many fuel channels in graphite prisms. Helium coolant is

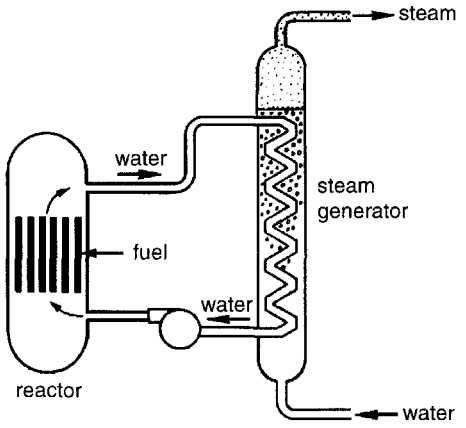

Figure 11.2. Pressurized water reactor (PWR).

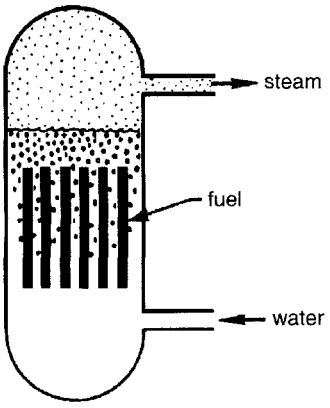

Figure 11.3. Boiling water reactor (BWR).

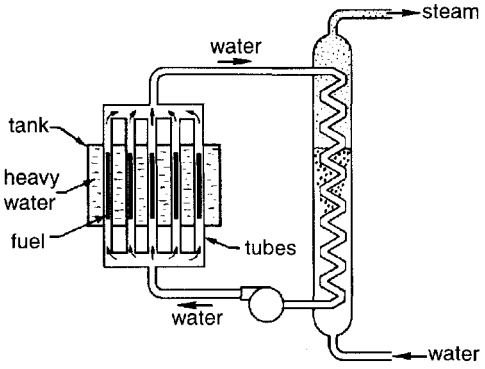

Figure 11.4. Heavy water reactor (HWR).

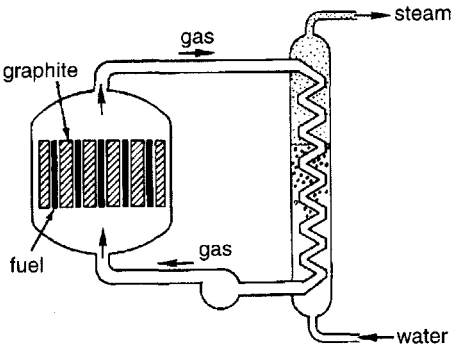

Figure 11.5. Gas cooled reactor (GCR).

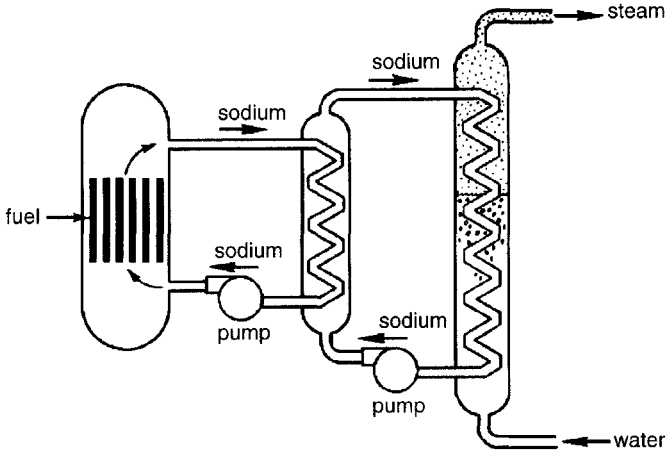

Figure 11.6. Liquid metal fast breeder reactor (LMFBR).

pumped through other channels bored through the graphite prisms. The hot exit helium then goes to a steam generator.

Liquid Metal Fast Breeder Reactors

In a fast reactor, the chain reaction is maintained by fast neutrons. Consequently, moderator materials cannot be used in the core. To avoid materials of low atomic mass, the core coolant is a liquid metal such as sodium or a mixture of potassium and sodium. Liquid metals have excellent heat transfer characteristics and do not require pressurization to avoid boiling. However, sodium becomes radioactive when it absorbs neutrons and also reacts chemically with water. To keep radioactive sodium from possibly interacting with the water/steam loop, an intermediate loop of non-radioactive sodium is used to transfer the thermal energy from the primary sodium loop to the water/steam loop (see Fig. 11.6).

The great advantage of such fast liquid-metal power reactors is that it is possible to create a breeder reactor, i.e., one in which more fissile fuel is produced than is consumed by the chain reaction. In such a breeder reactor, ^{238}U is converted to fissile ^{239}Pu or ^{232}Th into fissile ^{233}U (see Section 6.5.3). Although fissile fuel breeding also occurs in water and graphite moderated reactors, the ratio of new fuel to consumed fuel is less than one (typically 0.6 to 0.8).

Pressure-Tube Graphite Reactors

A widely used Russian designed power reactor is the (RBMK) *reactory bolshoi moshchnosti kanalnye* (translated: high-powered pressure-tube reactor). In this reactor (see Fig. 11.7), fuel is placed in fuel channels in graphite blocks that are stacked to form the core. Vertical pressure tubes are also placed through the graphite core and light water coolant is pumped through these tubes and into an overhead steam drum where the two phases are separated and the steam passes directly to the turbine.

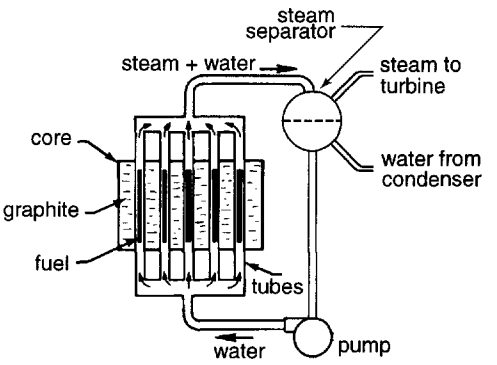

Figure 11.7. Graphite-moderated water-cooled reactor (RMBK).

11.1.4 Coolant Limitations

The thermal properties of a power reactor coolant greatly affect the reactor design. By far the most widely used coolant is water. It is inexpensive and engineers have a wealth of experience in using it as a working fluid in conventional fossil-fueled power plants. The great disadvantage of water as a coolant is that it must be pressurized to prevent boiling at high temperatures. If water is below the boiling point it is called *subcooled*. Water is *saturated* when vapor and liquid coexist at the boiling point, and it is *superheated* when the vapor temperature is above the boiling temperature. Above the *critical temperature* the liquid and gas phases are indistinguishable, and no amount of pressure produces phase transformation.

To maintain criticality in a water moderated core, the water must remain in liquid form. Moreover, steam is a much poorer coolant than liquid water. Thus, for water to be used in a reactor, it must be pressurized to prevent significant steam formation. For water, the critical temperature is 375 °C, above which liquid water cannot exist. Thus, in water moderated and cooled cores, temperatures must be below this critical temperature. Typically, coolant temperatures are limited to about 340 °C. This high temperature limit for reactor produced steam together with normal ambient environmental temperatures limit the thermal efficiency for such plants to about 34%.

Because steam produced by nuclear steam supply systems is saturated or very slightly superheated, expensive moisture separators (devices to remove liquid droplets) and special turbines that can operate with "wet steam" must be used. These turbines are larger and more expensive than those used in power plants that can produce superheated steam.

When gases or liquid metals are used as reactor coolants, liquid-gas phase transitions are no longer of concern. These coolants can reach much higher temperatures than in water systems, and can produce "dry" steam much above the critical temperature. Dry superheated steam at temperatures around 540 °C allows smaller, less expensive, turbines to be used and permits thermal efficiencies up to 40%.

11.2 Pressurized Water Reactors

Westinghouse Electric Company developed the first commercial pressurized water nuclear reactor (PWR) using technology developed for the U.S. nuclear submarine program. Many designs of various power capacities were developed by Westinghouse and built in several countries. The first, the Shippingport 60-MW(e) PWR, began operation in 1957 and operated until 1982. Following this first demonstration of commercial nuclear power, many other PWR plants, ranging in capacity from 150 MW(e) to almost 1500 MW(e), were built and operated in the U.S. and other countries. Other manufacturers of PWR plants included Babcock & Wilcox (B&W), Combustion Engineering (CE), France's Framatome, Germany's Siemens (KWU), Brown Boveri (BBR), Japan's Mitsubishi, and the Soviet Union's Atommash. Today, British Nuclear Fuel Ltd. (BNFL) owns both Combustion Engineering and Westinghouse reactor divisions and offers PWRs based on designs developed by these U.S. companies. Framatome many years ago acquired the B&W fuel/reactor division. Today BNFL and Framatome are the major vendors of PWRs, offering units ranging from 600 MW(e) to 1400 MW(e). The characteristics of a typical PWR power plant are given in Table 11.2.

11.2.1 The Steam Cycle of a PWR

The two-loop water/steam cycle of a PWR is shown in Fig. 11.8. In the primary loop, liquid water at pressures of about 2250 psi (15.5 MPa) is circulated through the core to remove the fission energy. The water leaving the core typically has a temperature of about 340 °C and the flow is regulated by the reactor coolant pumps. There are two to four separate primary coolant loops in PWRs, each with its own steam generator and recirculation pump. To avoid overpressures and pressure surges in the primary water, a *pressurizer*, which contains both liquid and saturated vapor, is included in the primary loop.

Table 11.2. Parameters for a typical 1000 MW(e) PWR sold in the early 1970s.

POWER	
thermal output	3800 MW
electrical output	1300 MW(e)
efficiency	0.34
CORE	
length	4.17 m
diameter	3.37 m
specific power	33 kW/kg(U)
power density	102 kW/L
av. linear heat rate	17.5 kW/m
rod surface heat flux	
average	0.584 MW/m^2
maximum	1.46 MW/m^2
REACTOR COOLANT SYSTEM	
operating pressure	15.5 MPa (2250 psia)
inlet temperature	292 °C
outlet temperature	329 °C
water flow to vessel	65.9×10^6 kg/h
STEAM GENERATOR (SG)	
number	4
outlet steam pressure	1000 psia
outlet steam temp.	284 °C
steam flow at outlet	1.91×10^6 kg/h

REACTOR PRESSURE VESSEL	
inside diameter	4.4 m
total height	13.6 m
wall thickness	22.0 cm
FUEL	
cylindrical fuel pellets	UO_2
pellet diameter	8.19 mm
rod outer diameter	9.5 mm
zircaloy clad thickness	0.57 mm
rod lattice pitch	12.6 mm
rods/assembly (17× 17)	264
assembly width	21.4 cm
fuel assemblies in core	193
fuel loading	115×10^3 kg
initial enrichment $\%^{235}U$	1.5/2.4/2.95
equil. enrichment $\%\ ^{235}U$	3.2
discharge fuel burnup	33 GWd/tU
REACTIVITY CONTROL	
no. control rod assemblies	68
shape	rod cluster
absorber rods per assembly	24
neutron absorber	Ag-In-Cd and/or B_4C
soluble poison shim	boric acid H_3BO_3

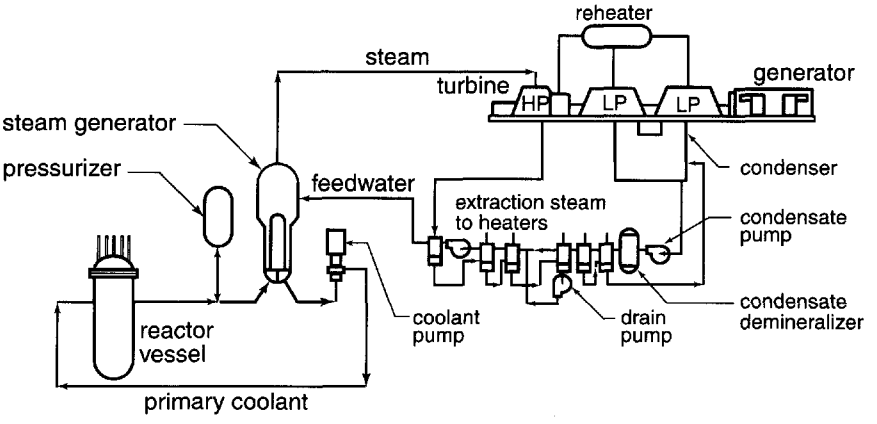

Figure 11.8. The steam cycle of a pressurized water reactor.

The primary water is passed through 2 to 4 *steam generators*, one for each primary loop, in which some of the thermal energy in the primary loop is transferred to the water in the secondary loop. The secondary water entering the steam generators is converted to saturated steam at about 290 °C (550 °F) and 1000 psi (7.2 MPa). This steam then expands through the turbine and, after leaving the turbine, is condensed back to the liquid phase in the *condenser*. The liquid condensate then passes through a series of 5 to 8 *feedwater heaters*, which use steam extracted from various stages of the turbine to heat the condensate before it cycles back to the steam generators. With this dual cycle, thermal energy efficiencies of about 34% can be achieved.

11.2.2 Major Components of a PWR

A nuclear power plant is a very complex system. Literally thousands of valves and pumps, miles of tubing and electrical wiring, and many tons of rebar and structural steel are needed. However, a few major components are of paramount importance. These include the pressurizer, steam generators, main recirculation pumps, reactor pressure vessel, turbo-generator, reheater, condenser, feedwater heaters, and the containment structure. Some of these items, unique to a PWR, are discussed below. The primary components of a 4-loop PWR are shown in Fig. 11.9.

The Pressure Vessel

A typical PWR pressure vessel is shown in Fig. 11.10. It is about 13 meters tall with a diameter of about 4 to 6 m. The vessel is built from low-alloy carbon steel and has a wall thickness of about 23 cm, which includes a 3-mm stainless steel clad on the inner surface. Such a thick wall is necessary to withstand the high operating pressure of about 2300 psi (158 bar). The primary coolant enters the vessel through two or more inlet nozzles, flows downward between the vessel and core barrel, flows upward through the reactor core removing the heat from the fuel pins, and then leaves the vessel through outlet nozzles. The fabrication and transportation of the roughly 500 tonne pressure vessel is a daunting task. Figure 11.11 shows a PWR being fabricated. Notice the size of the worker standing beside the pressure vessel.

Recirculation Pumps

Flow through the reactor core is controlled by the recirculation pump in each primary loop. These large pumps are vertical single-stage centrifugal type pumps designed to operate for the 30–40 year lifetime of the plant with minimal maintenance. All parts that contact the water are made from stainless steel. The pump is driven by a large (7000 HP), vertical, squirrel cage, induction motor. A flywheel is incorporated to increase the rotational inertia, thereby prolonging pump coast-down and assuring a more gradual loss of main coolant flow should power to the pump motor be lost.

The Pressurizer

The primary system of a PWR is very nearly a constant-volume system. As the temperature of the primary water increases or decreases, the water expands or contracts. However, water is almost incompressible and a small temperature change would lead to very large pressure changes, if the primary loop were totally filled

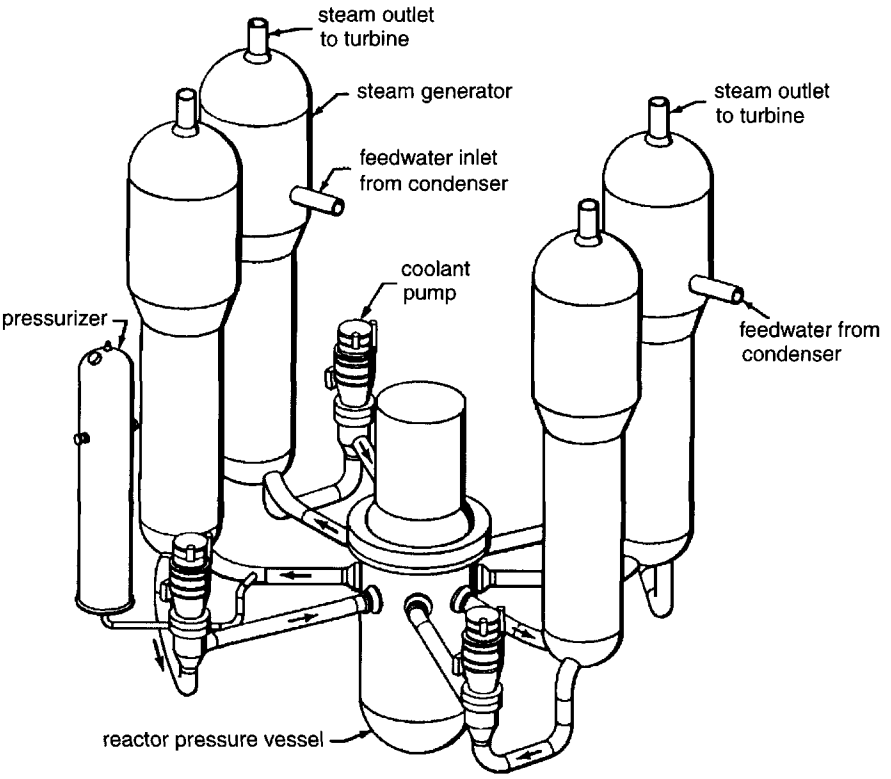

Figure 11.9. The steam cycle of a pressurized water reactor. [Westinghouse Electric Corp.].

with liquid water. To prevent such dangerous pressure surges, one primary loop of a PWR contains a pressure-regulating surge tank called a *pressurizer* (see Fig. 11.9).

The pressurizer is a large cylindrical tank with steam in the upper portion and water in the lower as shown in Fig. 11.13. The steam, being compressible, can absorb any sudden pressure surges. The pressurizer also is used to maintain the proper pressure in the primary system. If the primary water temperature should decrease (say from increased steam demand by the turbine) water would flow out of the pressurizer causing the steam pressure in it to drop. This drop in pressure in turn causes some of the water to flash to steam, thereby mitigating the pressure drop. At the same time, the pressure drop actuates electrical heaters in the base of the pressurizer to restore the system pressure. Likewise, an increase in water temperature causes the water to expand and to flow into the pressurizer and to increase the steam pressure, which, in turn, actuates values to inject spray water into the top of the pressurizer. This cooling spray condenses some of the steam thereby reducing the system pressure to normal.

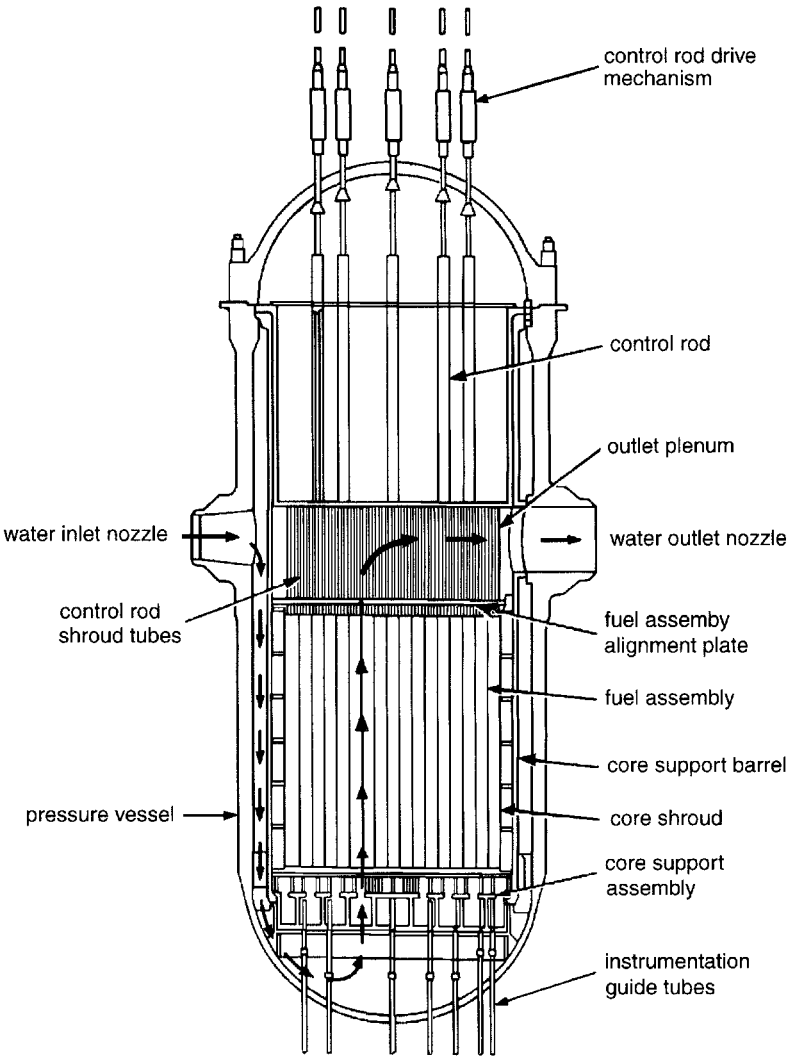

Figure 11.10. PWR pressure vessel. [Westinghouse Electric Corp.]

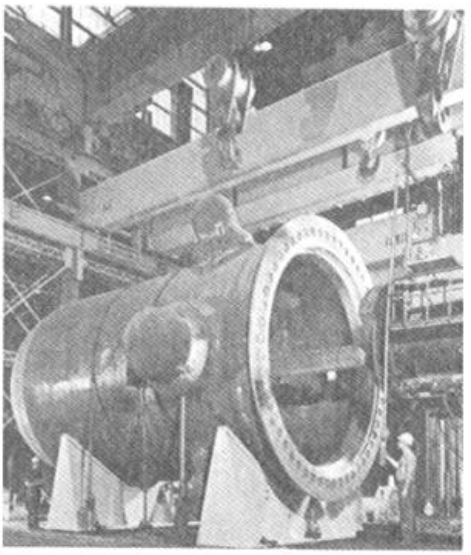

Figure 11.11. A PWR pressure vessel during manufacture. Source: [CE 1974].

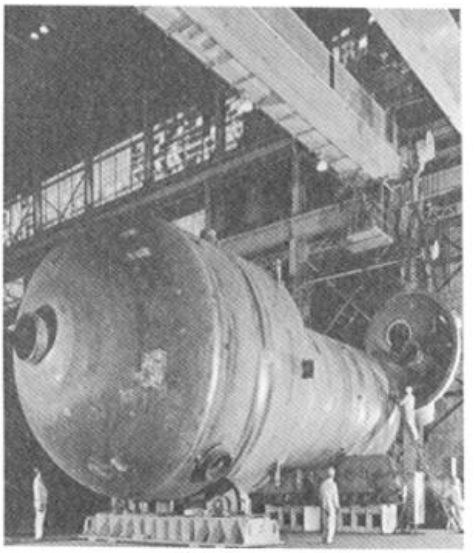

Figure 11.12. A PWR steam generator during fabrication. Source: [CE 1974].

Steam Generators

Steam generators are very large devices in which thermal energy of the primary water is transferred to the secondary water to produce steam for the turbo-generator. Several different designs have been used, although the most common type of steam generator is that evolved from a Westinghouse design. These steam generators, which are larger than the pressure vessel (compare Figs. 11.11 and 11.12), are nearly 21-m tall and 5.5 m in diameter at the upper end. The internal structure of such a steam generator is shown in Fig. 11.14. The hot high-pressure water of the primary loop (2250 psia, 345 °C, 262 tonnes/minute) is passed through a bank of U-shaped pressure tubes where some of the thermal energy of the primary water is transferred to the secondary water at 1000 psi on the outside of the pressure tubes and allowed to boil. The saturated steam above the tubes contains many small water droplets and the bulbous top portion of the steam generator contains cyclone separators to remove this entrained water and to allow only "dry" hot steam (0.25% moisture, 290 °C) to reach the high-pressure turbine. The primary water leaves the steam generator at a temperature of about 325 °C and reenters the reactor.

Nuclear Fuel

The uranium fuel used in a PWR is contained in many thousands of thin long fuel rods or *fuel pins*. Slightly enriched (3.3%) UO_2 pellets about 9 mm in diameter are stacked inside a Zircaloy tube 3.8-m long with a wall thickness of about 0.64 mm. The small diameter is needed to allow fission thermal energy produced in the UO_2 pellets to transfer rapidly to the water surrounding the fuel pins.

The fuel pins are assembled into *fuel assemblies* each containing an array of typically 17 × 17 pin locations. However, in many of these assemblies, 24 locations are occupied by guide tubes in which control-rod "fingers," held at the top by a "spider," move up and down in the assembly to provide coarse reactivity control. A

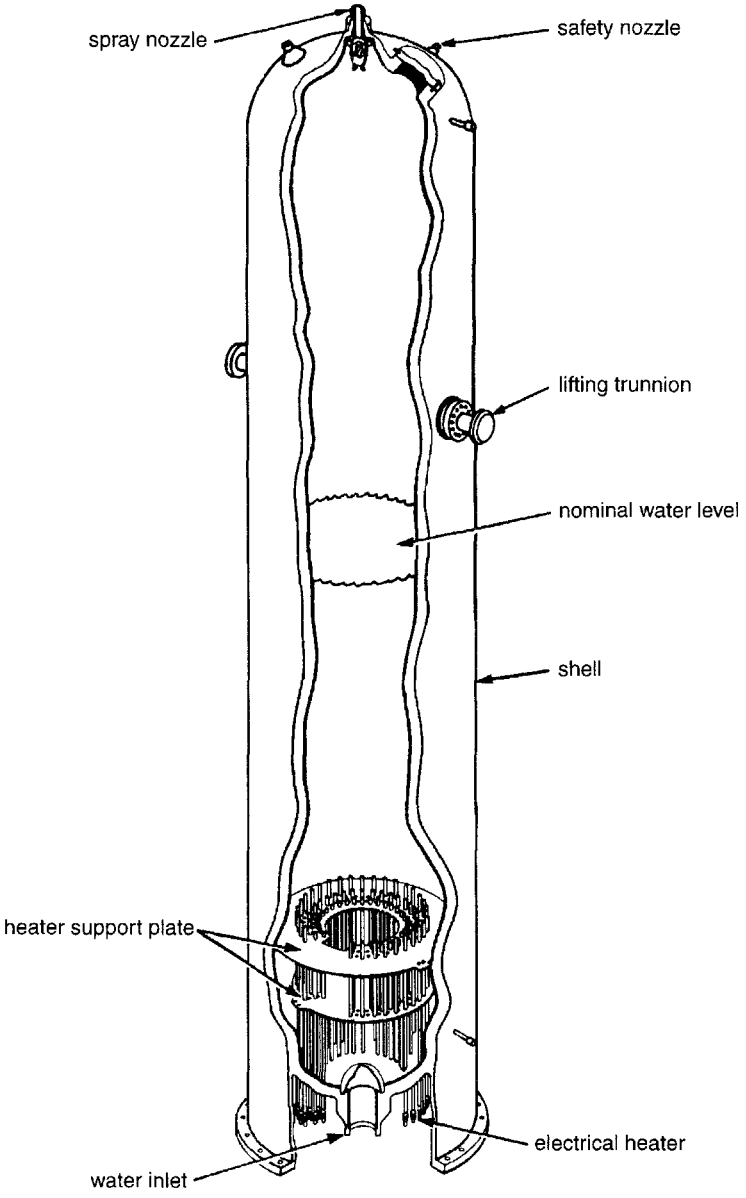

Figure 11.13. PWR pressurizer. [Westinghouse Electric Corp.]

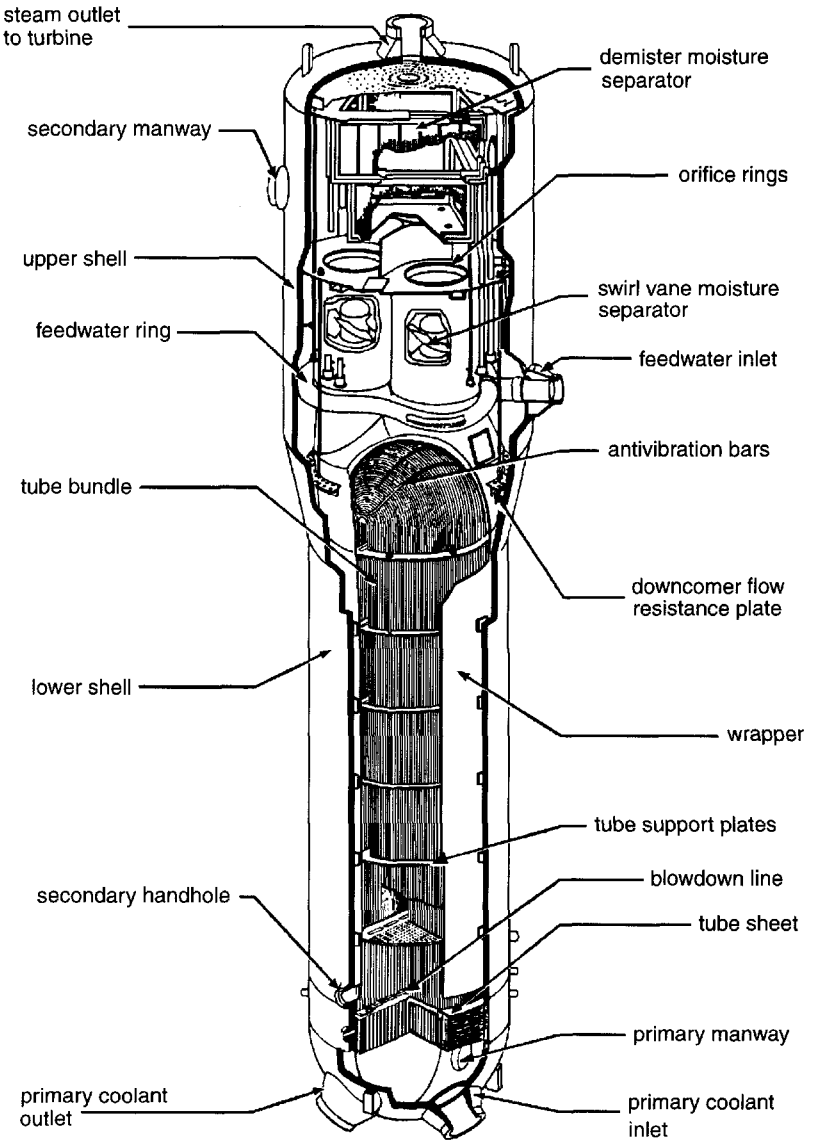

Figure 11.14. PWR steam generator. [Westinghouse Electric Corp.]

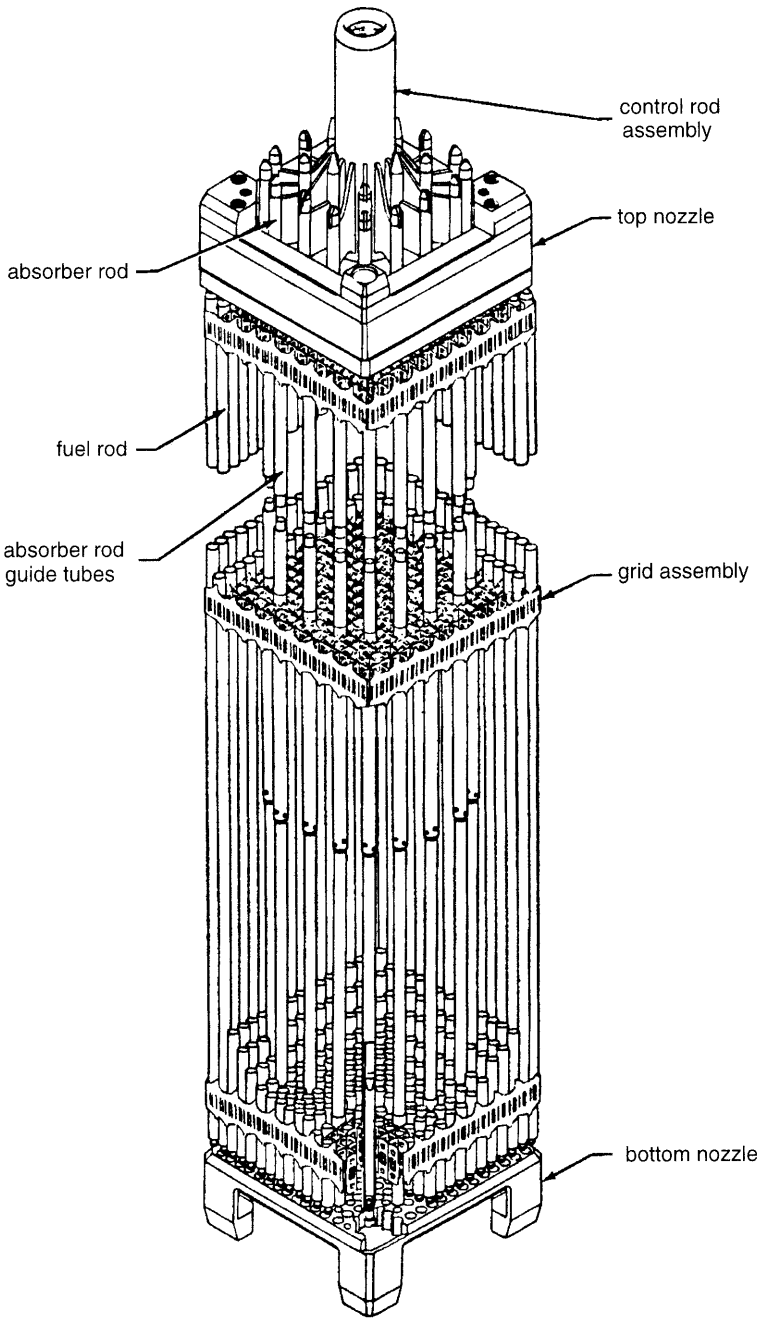

Figure 11.15. PWR fuel assembly with control rods. [Westinghouse Electric Corp.]

typical fuel assembly is shown in Fig. 11.15. Fuel assemblies with 15×15 and 16×16 arrays with fewer control rod locations have also been used.

Some 200 to 300 of these fuel assemblies are loaded vertically in a cylindrical configuration to form the reactor core. The assemblies typically spend three years in the core before they are replaced by new assemblies.

Reactivity Control

Short term or emergency reactivity control is provided by the 24 control rod fingers in many of the assemblies. These control fingers usually contain B_4C or, more recently, a mixture of silver (80%), indium (15%), and cadmium (5%) to produce slightly weaker absorbers. Generally, 4–9 adjacent control-rod spiders (which connect all the control rod fingers in an assembly) are grouped together and moved together as a single *control-rod bank*. The various control rod banks then provide coarse reactivity control.

Intermediate to long term reactivity control is provided by varying the concentration of boric acid in the primary water. Such a soluble neutron absorber is called a *chemical shim*. By varying the boron concentration in the primary water, excessive movement of control rods can be avoided.

For long-term reactivity control, *burnable poisons* are placed in some of the lattice positions of the fuel assemblies. These *shim rods*, from 9 to 20 per assembly, are stainless steel clad boro-silicate glass or Zircaloy clad diluted boron in aluminum oxide pellets.

The Containment Building

Of paramount importance in the safe use of nuclear power is the isolation of the radioactive fission products from the biosphere. To prevent the leakage of fission products into the environment from a nuclear power plant, three principal isolation barriers are used in every nuclear power plant. First the cladding of the fuel pins prevent almost all of the fission products from leaking into the primary coolant. However, with many thousands of fuel rods in a reactor, a few have small pin-holes through which some radioactive fission products escape. Elaborate clean-up loops are used to continuously purify the primary coolant and collect the fission products that leak from the fuel rods.

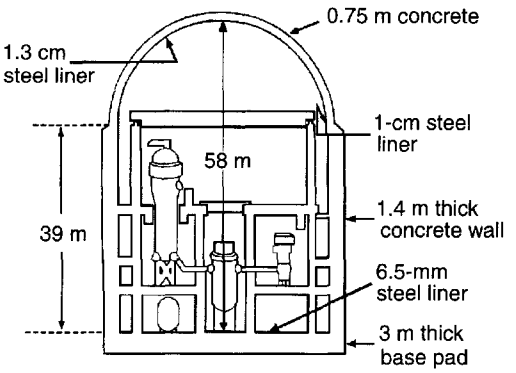

Figure 11.16. PWR containment building.

A second level of fission product confinement is provided by the pressure vessel and the isolated primary loop. Finally, should there be a leak in the primary system, the reactor and all the components through which the primary water flows are enclosed in a *containment building* designed to withstand tornados and other natural phenomena, as well as substantial overpressures generated from within the containment from accidental depressurization of the primary coolant. A typical confinement structure is shown in Fig. 11.16.

11.3 Boiling Water Reactors

The boiling water reactor (BWR) was developed by General Electric Company (GE) which, since its first 200 MW(e) Dresden unit in 1960, has built units as large as 1250 MW(e) in both the USA and in many other countries. In partnership with GE, Hitachi and Toshiba in Japan have developed an advanced BWR. Other designs of BWRs have been proposed by Germany's Kraftwerk Union (KWU) and Sweden's ASEA-Atom. Some operating parameters of a large BWR, typical of those now in operation, are given in Table 11.3.

11.3.1 The Steam Cycle of a BWR

A BWR uses a single direct-cycle steam/water loop as shown in Fig. 11.17. Its flow is regulated by the recirculation jet pumps, and feedwater entering the reactor pressure vessel is allowed to boil as it passes through the core. After passing through a complex moisture separation system in the top of the pressure vessel, saturated steam at about 290 °C and 6.9 MPa (1000 psi) leaves the reactor pressure vessel and enters the high-pressure (HP) turbine. The exit steam from the HP turbine then is reheated and moisture removed before entering low-pressure (LP) turbines from which it enters a condenser, is liquefied, and pumped through a series of feedwater heaters back into the reactor vessel. Such a conventional regenerative cycle typically has a thermal efficiency of about 34%.

11.3.2 Major Components of a BWR

The basic layout of a BWR is considerably simpler than that of a PWR. By producing steam in the reactor vessel, a single water/steam loop can be used, eliminating the steam generators and pressurizer of the PWR. However, radioactivity produced in the water as it passes through the core (notably ^{16}N with a 7-s half life) passes through the turbine, condenser and feedwater heaters. By contrast the same radioactivity in a PWR is confined to the primary loop, so that non-radioactive steam/water is used in the secondary loop. Thus, in a BWR plant considerably greater attention to radiation shielding is needed. Because the pressure-vessel containment in a BWR serves also as a steam generator, the internals of a BWR vessel are somewhat more complex.

The BWR Pressure Vessel

A typical BWR pressure vessel and its internal structures are shown in Fig. 11.18. The 3.6-m high core occupies a small fraction of the 22-m high pressure vessel. The pressure vessel is about 6.4 m in diameter with a 15-cm, stainless-steel clad wall of carbon steel. Above the core are moisture separators and steam dryers designed to remove almost all of the entrained liquid from the steam before it leaves the top of the pressure vessel. Because of the steam conditioning systems in the top half of the vessel, the reactor control rods must be inserted through the bottom of the vessel. Hydraulic force rather than gravity must be relied upon to ensure that the control rods are fully inserted into the core if electrical power to the plant is lost.

Jet Pumps and Recirculation Flow

A unique feature of BWRs is the recirculation flow control provided by the jet pumps located around the periphery of the core. By varying the flow of water

Table 11.3. Parameters for a typical 1000 MW(e) BWR sold in the early 1970s.

POWER	
thermal output	3830 MW
electrical output	1330 MW(e)
efficiency	0.34
CORE	
length	3.76 m
diameter	4.8 m
specific power	25.9 kW/kg(U)
power density	56 kW/L
av. linear heat rate	20.7 kW/m
rod surface heat flux	
average	0.51 MW/m^2
maximum	1.12 MW/m^2
REACTOR COOLANT SYSTEM	
operating pressure	7.17 MPa (1040 psia)
feedwater temperature	216 °C
outlet steam temperature	290 °C
outlet steam flow rate	7.5×10^6 kg/h
core flow rate	51×10^6 kg/h
core void fraction (av.)	0.37
core void fraction (max.)	0.75
no. in-core jet pumps	24
no. coolant pumps/loops	2

REACTOR PRESSURE VESSEL	
inside diameter	6.4 m
total height	22.1 m
wall thickness	15 cm
FUEL	
cylindrical fuel pellets	UO_2
pellet diameter	10.57 mm
rod outer diameter	12.52 mm
zircaloy clad thickness	0.864 mm
rod lattice pitch	16.3 mm
rods/assembly (8× 8)	62
assembly width	13.4 cm
assembly height	4.48 m
fuel assemblies in core	760
fuel loading	168×10^3 kg
av. initial enrichment $\%^{235}U$	2.6%
equil. enrichment $\%\ ^{235}U$	1.9%
discharge fuel burnup	27.5 GWd/tU
REACTIVITY CONTROL	
no. control elements	193
shape	cruciform
overall length	4.42 m
length of poison section	3.66 m
neutron absorber	boron carbide
burnable poison in fuel	gadolinium

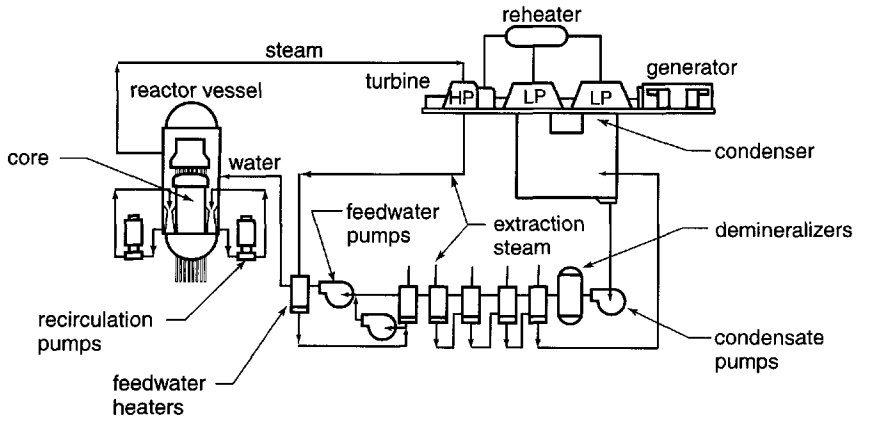

Figure 11.17. The steam cycle of a boiling water reactor.

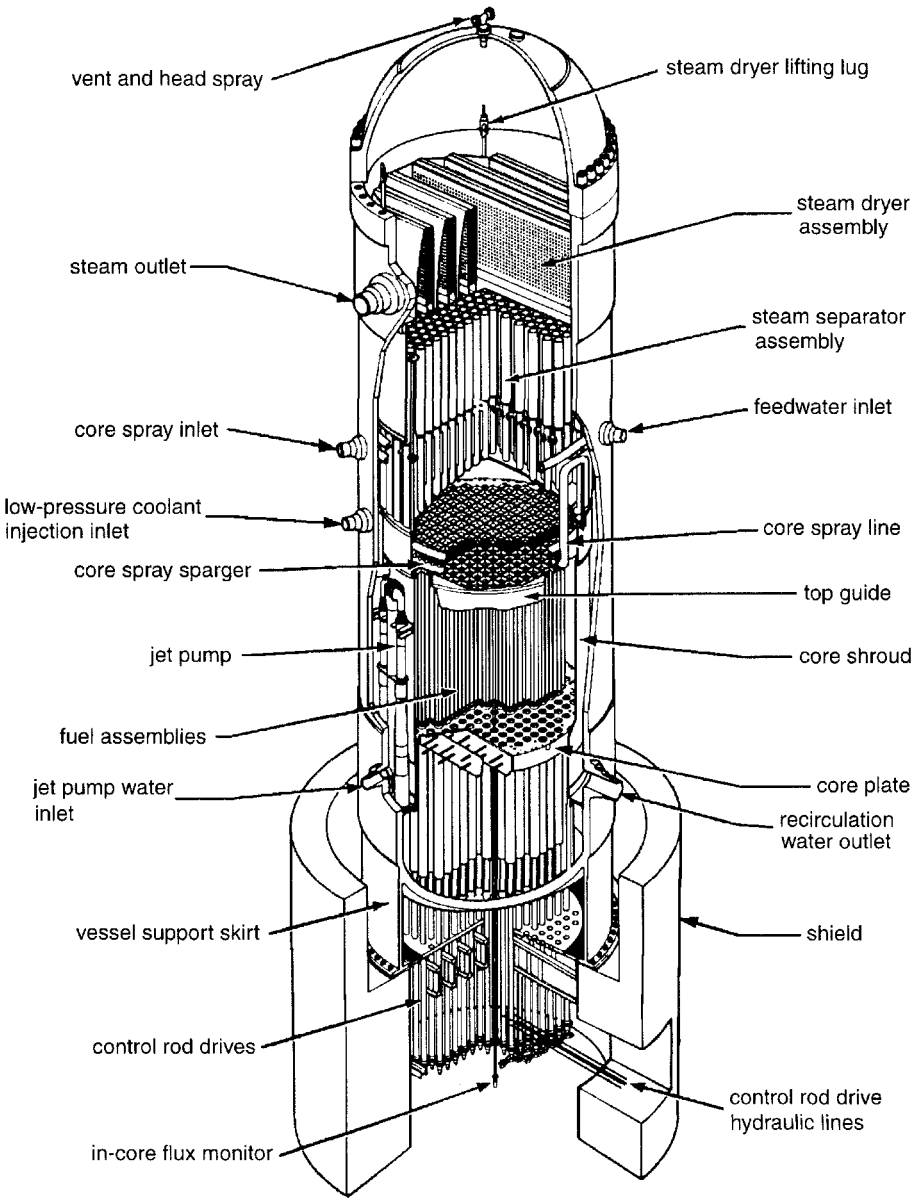

Figure 11.18. BWR pressure vessel components. [General Electric Co.]

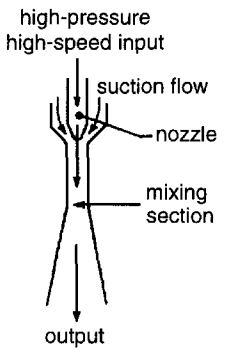

Figure 11.19. BWR jet pump.

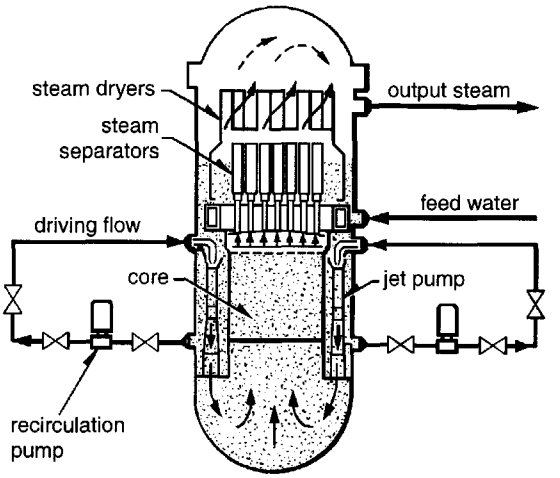

Figure 11.20. Recirculation flow in a BWR.

through the core, the amount of steam, or void fraction of the water, in the core can be controlled. This allows fine control of the reactivity of the core, since increasing steam production decreases the amount of liquid water and, thereby, the amount of neutron moderation and hence the reactivity. During normal operation, the water flow through the core is normally used to control reactivity rather than the coarse reactivity control provided by the 100–150 control rods.

The jet pump recirculation system used to control the water flow through the core is shown in Fig. 11.19. The recirculation pumps control the injection of high pressure and high-velocity water to the venturi nozzles of the jet pumps located around the periphery of the core (up to 21). This forced, high-velocity, injection water flow (see Fig. 11.19) creates a suction flow of vessel water downward between the vessel wall and the core shroud and then upward through the core. This reactivity controlling water flow through the recirculation loop is shown in Fig. 11.20. Up to about 30% of the feedwater to the BWR vessel is diverted from the vessel to the two recirculation loops used to operate the jet pumps around the periphery of the vessel.

BWR Fuel

The thousands of fuel pins, composed of enriched UO_2 pellets in Zircaloy tubes, are much like those in a PWR. However, they are arranged in square subassembly arrays of 8×8 to 10×10 pins (see Fig. 11.21). Four subassemblies, each contained in a Zircaloy shroud, make up a *fuel module* (see Fig. 11.22). Subassemblies are individually orificed to control water/steam flow and assure a uniform elevation at which boiling commences and a uniform steam quality as coolant leaves the core. In contrast, PWR fuel assemblies are open. Cross flow is minimal because of the lack of driving force, and, absent boiling, the mass flow rate throughout the core cross section is very uniform.

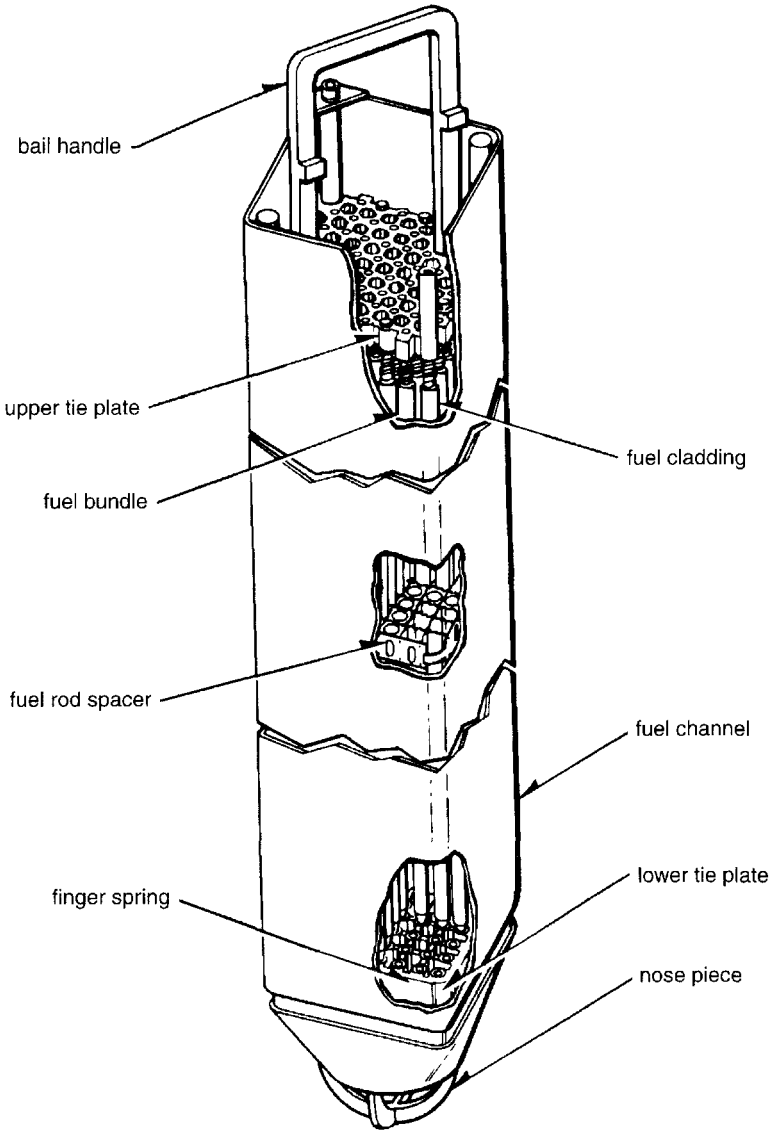

Figure 11.21. A BWR fuel subassembly. [General Electric Co.]

The four subassemblies in a fuel module are separated by water gaps in which a cruciform control blade moves up and down to control the core reactivity (see Fig. 11.22). Because the thermal flux peaks near water channels, lower enrichment fuel pins are used near the water channels. Pins of up to four different enrichments are used in each subassembly to flatten the power profile across the assembly.

Reactivity Control

In a BWR three different mechanisms are used to control the core's reactivity. Short term reactivity changes are made by adjusting the recirculation flow through the jet pumps. As the water flow through the core is increased (decreased), the amount of boiling and voiding in the core decreases (increases) and the reactivity increases (decreases) because of increased (decreased) neutron moderation. Flow modulation can accommodate power variations of as much as 25%.

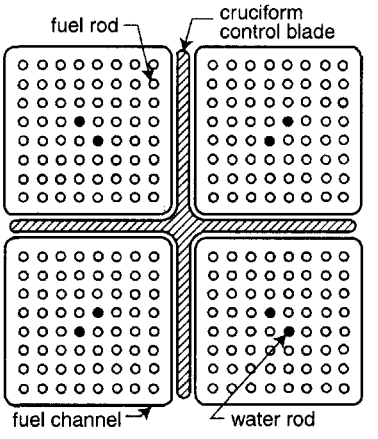

Figure 11.22. A BWR fuel module.

Longer term reactivity control is provided by the cruciform control blades that are raised and lowered from the bottom of the vessel, both to avoid the steam separator/dryers in the top of the vessel and to use their greater effectiveness in the liquid water in the bottom part of the core than in the vapor region in the top part.

Reactivity control to compensate for fuel burnup is also provided by burnable gadolinium oxide (GdO_2) mixed in the UO_2 pellets of the fuel pins. As the gadolinium absorbs neutrons, it is transformed into an isotope with a low neutron absorption cross section and thus allows more neutrons to be absorbed in the remaining fuel.

Although PWRs use soluble boron, a strong thermal neutron absorber, in the primary cooling water to regulate the core's reactivity, this method of reactivity control is not available to BWRs. Boiling would cause boron to be precipitated as a solid on the fuel pin surfaces, thus making reactivity control impossible.

11.4 New Designs for Central-Station Power

During the 1990s a major effort was made in the United States to design, license, and install a new generation of nuclear reactors for central-station power generation.[1] Efforts proceeded along three avenues. Electric utility requirements and specifications were developed under the auspices of the Electric Power Research Institute as the Advanced Light Water Reactor (ALWR) Program. Design work was sponsored

[1]This effort is described in a review article "New nuclear generation—in our lifetime," by E.L. Quinn, in the October 2001 issue of *Nuclear News*. The case for new technology is made in the May 2001 report of the National Energy Policy Development Group, "Reliable, Affordable, and Environmentally Sound Energy for America's Future," available on line or from the U.S. Government Printing Office.

by the U.S. Department of Energy in partnership with vendors and utilities. Design certification was done by the Nuclear Regulatory Commission under streamlined licensing procedures. A path similar to that of the USA was followed in the European Pressurized Water Reactor (EPR) development program.

Two design scopes were followed. One, evolutionary design, built directly upon an installed capacity of about 100,000 MW(e) from 103 nuclear plants, stressing safety, efficiency and standardization. The other, stressing ultra-safe features, called for passive safety features involving gravity, natural circulation, and pressurized gas as driving forces for cooling and residual heat removal. Two evolutionary plant designs and one passive design have so far been certified by the U.S. Nuclear Regulatory Commission.

11.4.1 Certified Evolutionary Designs

GE Advanced Boiling Water Reactor

The design effort for this plant was accomplished by GE Nuclear Energy, in cooperation with Hitachi, and Toshiba. It is a 1350-MW(e) plant operable with a mixed-oxide (PuO_2/UO_2) fuel cycle. The Tokyo Electric Power Company has two ABWRs in operation, the second of which was constructed in a record 48 months. Two similar plants are in construction in Taiwan for the Taiwan Power Company. The ABWR is adapted to USA utility needs and conforms with the EPRI evolutionary design requirements. The ABWR was the first such design to receive the NRC final design approval.

Westinghouse-BNFL System 80+

The System 80+ plant offered by Westinghouse-BNFL (British Nuclear Fuels), originally conceived by Combustion Engineering and ASEA Brown Boveri (ABB), is a 1350-MW(e) pressurized water reactor. Features of the System 80+ design are incorporated in eight units completed or under construction in the Republic of Korea. Important design features include a dual spherical steel confinement for accident mitigation and a cavity-flood system with an in-containment refueling-water storage supply.

11.4.2 Certified Passive Design

Westinghouse-BNFL AP600

The AP600, the only certified passive design, is a 600-MW(e) modular pressurized water reactor that relies on passive systems for both emergency core cooling and residual heat removal. Implementation of the passive safety features greatly reduces the operation, maintenance and testing requirements of the AP600. Through the use of modular construction techniques similar to those applied in ship construction, the design objective is a 36-month schedule from first concrete pour to the fuel load.

11.4.3 Other Evolutionary LWR Designs

Several advanced LWRs are in the process of acquiring NRC certification. These are the Westinghouse-BNFL AP1000, the GE Nuclear ESBWR, and the IRIS systems. The AP1000 is a 1000-MW(e) uprated AP600 system that retains the passive safety features, with an estimated per-kW cost reduction of 30 percent. The Evolutionary Simplified Boiling Water Reactor (ESBWR) is a 1380-MW(e) natural

circulation plant with no recirculation or reactor internal pumps. It makes extensive use of components developed for the ABWR. The IRIS system (International Reactor Innovative and Secure) is a modular system, 100 to 350-MW(e) per module, being developed by an international design team headed by Westinghouse. The design emphasizes proliferation resistance and safety enhancements. IRIS uses LWR technology, but is newly engineered to include a five-to-eight-year core reloading schedule.

11.4.4 Gas Reactor Technology

Two designs are being pursued actively. One is the Pebble Bed Modular Reactor (PBMR) under development in South Africa by the utility Eskom, by the Industrial Development Corporation of South Africa, by British Nuclear Fuels, and the U.S. firm, Exelon. The other is the General Atomics Gas Turbine-Modular Helium Reactor (GE-MHR).

The PBMR is an evolutionary design based on HTR (helium cooled high temperature reactor) technology developed jointly by Siemens and ABB. It is a modular system, 120-MW(e) per module, with an estimated 18 to 24 month construction time. The core of the reactor is a cylindrical annulus, reflected on the outside by a layer of graphite bricks and on the inside by approximately 110,000 graphite spheres. The core contains approximately 330,000 60-mm diameter fuel spheres. Each has a graphite-clad, 50-mm diameter, graphite matrix containing 0.5-mm diameter UO_2 fuel particles surrounded by refractory layers of graphite and silicon carbide. The fuel is stable at temperatures as high as 2000 °C, well above core temperatures even in a worst-case loss-of-coolant accident. Helium flows through the pebble bed and drives a gas turbine for power generation.

General Atomics (GA) is the industrial pioneer of the Gas Turbine–Modular Helium Reactor (GT-MHR), an ultra-safe, meltdown-proof, helium-cooled reactor, with refractory-coated particle fuel. In early 1995, GA and Russia's Ministry of Atomic Energy (MINATOM), in cooperation with Framatome and Fuji Electric, began a joint program to design and develop the GT-MHR for use in Russia in the destruction of weapons-grade plutonium and replacement of plutonium production reactors in the Russian Federation. A typical GT-MHR module, rated at 600 MW(t), yields a net output of about 285-MW(e). The reactor can be fueled with uranium or plutonium.

11.5 The Nuclear Fuel Cycle

The several stages involved in the processing of nuclear fuel from its extraction from uranium ore to the ultimate disposal of the waste from a reactor is called the *nuclear fuel cycle*. A schematic of this cycle for LWRs is shown in Fig. 11.23. In this figure, the cycle shown in by the solid boxes and arrows (i.e., ignoring the dashed line components) is the *once-through* cycle, currently used by all power plants in the United States.

The once-through cycle begins with the mining of uranium ore either by shaft mining and, more commonly, by open-pit mining. The uranium is extracted from the ore in a milling process to produce "yellow cake" which is about 80% U_3O_8. Before the uranium can be used in modern LWRs, the ^{235}U content must be enriched from its natural isotopic abundance of 0.720 a% to about 3 a%. The first step

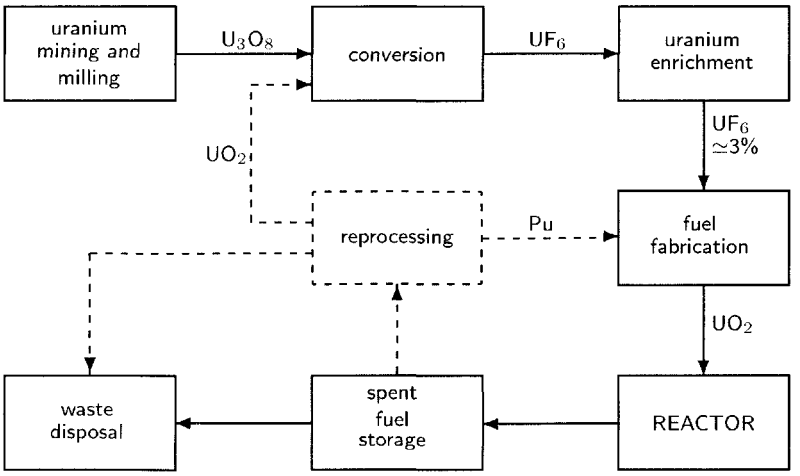

Figure 11.23. The nuclear fuel cycle for LWRs without fuel recycle (solid boxes) and with uranium and plutonium recycle (solid plus dashed elements).

in the enrichment process is the conversion of U_3O_8 to UF_6, a substance that becomes gaseous at relatively low temperatures and pressures. The gaseous UF_6 is then processed to separate $^{235}UF_6$ molecules from the far more abundant $^{238}UF_6$ molecules. The enrichment techniques are discussed later in Section 11.5.2. After UF_6 has been isotopically enriched to about 3 a%, it is converted to ceramic UO_2 pellets which are then used in the fuel rods of LWR reactors.

Every one to two years, a LWR is shutdown for several weeks, during which time about one-third of a PWR's fuel and about one-quarter of a BWR's fuel is replaced and the older remaining fuel is shuffled inward towards the center of the core. The fuel assemblies removed from the core are initially submerged for a few years in a spent-fuel storage pool where the water safely removes the decay heat produced by the radioactive decay of the fission products. After several years, the spent fuel assemblies may be transferred to a long-term spent-fuel storage pool or to dry spent-fuel storage casks at a facility outside the plant. By now the greatly reduced decay heat can be convectively removed by only the gas in the casks. Eventually, it is planned to place the waste in these spent-fuel assemblies into a permanent waste repository (see Section 11.5.3). The annual uranium needs and production of other elements in a typical

Table 11.4. The annual material requirements and production (in kg) for a typical 1000 MW(e) PWR. Source: Lamarsh and Baratta [2001].

Mining/Milling output:	
U in U_3O_8	150,047
Conversion output:	
U in UF_6	149,297
Enrichment output:	
^{235}U	821
^{238}U	27,249
U tails (0.2 %)	121,227
Reactor Output:	
^{235}U	220
total U	25,858
Pu (fissile)	178
total Pu	246
total U+Pu	26,104
fission products	873

1000 MW(e) PWR are listed in Table 11.4. The data in this table are based on an assumed plant capacity factor of 0.75, i.e., the production of 750 GWy of electrical power, and on an assumed 0.2% enrichment in the tailings discarded by the enrichment process.

The fuel removed from a reactor, besides containing radioactive fission products also contains significant amounts of residual ^{235}U and fissile ^{239}Pu and ^{241}Pu (see Table 11.4). In principle, these fissile isotopes can be extracted from the spent fuel and recycled back into new *mixed oxide* fuel for a LWR. The fuel cycle with such recycling is shown in Fig. 11.23 with the addition of the dashed components. The use of recycling of fissile isotopes in spent fuel can reduce the lifetime U_3O_8 requirements by about 45%. However, the economics of such recycling are uncertain and political concern over the possible diversion of recycled plutonium for terrorist bombs so far has prevented recycling in the United States. Nevertheless, the annual discarding from a LWR of 220 kg of ^{235}U and the 180 kg of fissile plutonium isotopes represents an energy equivalent of about 1.3 million tons of coal. Thus there are strong energy incentives to adopt recycling.

11.5.1 Uranium Requirements and Availability

To fuel existing nuclear power plants, uranium must first be extracted from natural deposits and converted into a form suitable for use in a reactor. In this section the uranium needs of LWRs and the availability of uranium are discussed.

Uranium Needs for LWRs

Currently, uranium is extracted from ores containing uranium bearing minerals of complex composition. High grade ores contain about 2% uranium with medium-grade ores ranging from 0.1 to 0.5% uranium. To extract uranium from its ore, mills near the mining areas use either chemical leaching or solvent extraction techniques to produce yellow cake, which is about 80% U_3O_8. The yellow cake is then shipped to other facilities where it is purified and converted into uranium dioxide (UO_2) or uranium hexafluoride (UF_6). The tailings left at the milling site still contain considerable uranium and, consequently, emit relatively large amounts of radon. To mitigate radon releases, the tailing can be either placed underground or capped with a thick earthen barrier.

From the data in Table 11.4, it is seen that a 1000-MW(e) LWR with a 75% capacity factor requires about 150 tonnes of new natural uranium each year if the once-through fuel cycle is used. This amounts to 4500 tonnes of uranium over the 30-y lifetime of such a plant. With plutonium and uranium recycling, about 80 tonnes of new uranium are needed per year or 2400 tonnes over the lifetime of the plant.[2] By contrast, a liquid-metal fast breeder reactor of the same capacity has a lifetime requirement for natural or depleted uranium of about only 40 kg (assuming mixed oxide fuel recycling), since such a reactor produces more fissile fuel than it consumes.

[2]These uranium requirements are based on the depleted uranium produced in the enrichment process having a ^{235}U enrichment of 0.2 wt%. Less new uranium would be required if a lower, but more expensive, tailing enrichment were used.

Uranium Availability

In 2000 there was a worldwide installed nuclear power capacity of about 340 GWe. Since a 1 GWe LWR has a lifetime need of about 4500 tonnes of uranium, and since most reactors are LWRs, this installed capacity represents a uranium need of about $340 \times 4500 = 1.5 \times 10^6$ tonnes The global nuclear capacity is expected to increase to between 415 and 490 GWe by 2010, and even more uranium will be needed. An obvious question is how much uranium will be available for the future needs of nuclear power.

Most of the uranium needed for reactor fuel during the next 20 to 30 years will come from reasonable assured deposits from which uranium can be produced for no more than \$130 per kg. Such resources as estimated to total about 4×10^6 tonnes, 80% of which are in North America, Africa and Australia [Lamarsh and Baratta 2001]. In addition, there are resources that are more speculative whose estimate is based on geological similarities of similar ore deposits and other indirect evidence, that suggest additional uranium resources can be realized. Such speculative resources have been estimated (for production costs of less than \$130 per kg) at about 11×10^6 tons worldwide with about 1.4×10^6 tons in the United States [Lamarsh and Baratta 2001]. However, few of these speculative reserves will be developed within the next thirty decades.

At higher production costs, much more uranium can be obtained since ore with more dilute uranium concentrations can be exploited. There are many low grade uranium deposits containing enormous quantities of uranium. For example, a shallow geological formation known as Chattanooga shale underlies six Midwestern states. This shale contains an estimated 5×10^6 tonnes of uranium with concentrations up to 66 ppm. Indeed, an area of just 7 miles square of this shale contains the energy equivalent of all the world's oil [Lamarsh and Baratta 2001]. Other similar formations containing significant amounts of low-concentration uranium exist in many other countries. However, none of these resources are presently economically useful.

From the elemental abundances listed in Table A.3, we see that uranium is almost 700 times more abundant than gold and 6 times more abundant than iodine in the earth's crust. Within the first 10 km of the earth's crust, there are about 5×10^{13} tonnes of uranium. Of course, only a small fraction of this can be recovered economically. Uranium, because of its high solubility in an oxidizing environment, is also relatively abundant in sea water (0.0032 mg/L). All the oceans of the world contain about 4×10^9 tonnes of uranium. However, no economically feasible method has yet been devised to extract uranium from the sea.

11.5.2 Enrichment Techniques

There are many methods by which elements can be enriched in a particular isotope. Many techniques have been proposed for enrichment or separation of an isotope from its element. Several have been developed and demonstrated to be economically feasible. Generally, the lighter the element, the easier (and less costly) it is to separate the isotopes. In this section several of the more important methods for enriching uranium in ^{235}U are summarized.

Gaseous Diffusion

The gaseous diffusion method was the first method used to enrich uranium in ^{235}U to an exceedingly high level (> 90%) so that atomic bombs could be constructed. Today it is the primary method used to enriched uranium to about 3% as needed for LWRs.

Uranium, whose isotopes are to be separated, must first be incorporated into a molecule such as UF_6, which is gaseous at only slightly elevated temperatures. Fluorine has only the single stable isotope ^{19}F so that the mass difference of $^{235}UF_6$ and $^{238}UF_6$ is due only to the mass difference in the uranium isotopes. The basis of gaseous diffusion enrichment is to find a porous membrane with pores sufficiently large to allow passage of the molecules but prevent bulk gas flow. In a container with two regions separated by such a membrane, the UF_6 molecules are pumped into one region. The $^{235}UF_6$ molecules travel faster than the $^{238}UF_6$ molecules, strike the membrane more often, and preferentially are transmitted through the membrane into the second region. The gas extracted from the second region is thus enriched in ^{235}U.

In gaseous diffusion enrichment of UF_6 membranes of nickel or of austenitic stainless steel are used. However, because the mass difference between $^{235}UF_6$ and $^{238}UF_6$ is so small, each gaseous diffusion cell or *stage* has but a small enrichment capability. Thus, hundreds of such diffusion-separation stages must be interconnected such that the output UF_6 from one cell becomes the input of another cell. This cascade technology, which requires enormous amounts of electricity to pump and cool the gas, has been used by France, China, and Argentina as well as the United States to obtain enriched uranium.

Gas Centrifuge

Gas molecules of different masses can be separated by placing the gas in a rotor container and spinning it at high speed. The centrifugal force causes the molecules to move toward the outer wall of the rotating container where the lighter molecules are buoyed up and moved slightly away from the rotor wall by the heavier molecules adjacent to the wall. This technique works more effectively when there is a large percentage difference between the molecular masses. Indeed, it was first used in the 1930s to separate isotopes of chlorine.

Because of the small mass difference between $^{235}UF_6$ and $^{238}UF_6$, the capability of a single centrifuge to separate $^{235}UF_6$ from $^{238}UF_6$ is not large. Like gaseous diffusion cells, many gas centrifuges must be connected together in cascades to achieve the necessary enrichment for nuclear fuel. The great advantage of uranium enrichment by gas centrifuges is that it requires only a few percent of the electrical energy needed by gaseous diffusion plants of the same enrichment capacity. At least nine countries have developed uranium enrichment facilities using gas centrifuges.

Aerodynamic Separation

In this enrichment process, a mixture of hydrogen and uranium hexafluoride is subjected to strong aerodynamic forces to separate $^{235}UF_6$ from $^{238}UF_6$. In one variation, the gas flows at high speed through a curved nozzle. During passage through this curved nozzle, the heavier $^{238}UF_6$ preferentially moves towards the outer wall (surface with the larger radius) of the nozzle. An appropriately placed sharp divider at the nozzle exit then separates the two uranium isotopes. Although

this curved nozzle technique has a better separation ability than a gas centrifuge cell, it has not yet proven to be economically superior.

An alternative aerodynamic technique introduces a mixture of hydrogen and uranium hexafluoride at high speed through holes in the side wall of a tube. The tube narrows towards the exit, and as the gas flows down the tube, it spirals with increasing angular speed so that $^{238}UF_6$ is preferentially forced to the tube surface where it is extracted. The lighter $^{235}UF_6$ is left to exit the tube. A plant using this technique has been successfully operated in South Africa, but was found to be economically impractical.

Electromagnetic Separation

In this process, ionized uranium gas is accelerated by an electric potential through a perpendicular magnetic field. The magnetic field deflects the circular trajectories of $^{235}UF_6$ and $^{238}UF_6$ to different extents. Two appropriately placed graphite catchers receive the two uranium isotopes. The great advantage of this method is that complete separation can be achieved by a single machine.

This technique was developed at Oak Ridge, TN during World War II for obtaining highly enriched uranium for the atom bomb project. However, it has proven to be more costly than gaseous diffusion enrichment and is no longer used for enriching nuclear reactor fuel. However, such magnetic separation devices, known as *cyclotrons* are routinely used today in medical facilities to extract radioactive isotopes of lighter elements for use in nuclear medicine.

Laser Isotope Separation

A recent enrichment technique uses the small difference in the electron energy levels of ^{235}U and ^{238}U caused by the different masses of their nuclei. For excitation of a particular electron energy level in ^{235}U and ^{238}U, this *isotope shift* is about 0.1×10^{-8} cm (10 nm) for light of wavelength 5027.3×10^{-8} cm, corresponding to a photon energy difference of 49.1 μeV about an energy of 0.0246 eV. Lasers with a bandwidth of 10 pm at this frequency are available, so that one of the isotopes can be selectively excited.

To separate ^{235}U from ^{238}U, uranium in a vacuum chamber is first vaporized with an energetic electron beam. Then a laser beam of precisely the correct frequency to excite ^{235}U, but not ^{238}U, is passed back and forth through the chamber. A second laser beam is then used to ionize the excited ^{235}U whose ion is subsequently removed by electric fields in the chamber without disturbing the vaporized, but still neutral, ^{238}U atoms. The same technique can also be applied to the selective ionization of UF_6 gas. Laser enrichment technologies are currently being actively developed.

11.5.3 Radioactive Waste

Radioactive waste is generated in all portions of the nuclear fuel cycle, ranging from slightly contaminated clothing to highly active spent fuel. The goal of radioactive waste management is to prevent any significant waste activity from entering the biosphere before the radionuclides have decayed to stable products. Clearly, the sophistication of the technology required for the safe sequestration of the radwaste depends on both the number, or activity, and the half-lives of the radionuclides in the waste. For example, low-activity radwaste with half-lives of a few days can be retained for several weeks to allow the activity to decay to negligibly levels and

then disposed of as ordinary (non-radioactive) waste. By contrast, some waste from spent fuel must be contained safely for hundreds of thousands of years.

Classification of Radioactive Wastes

There is no universally accepted classification of the many different types of radioactive wastes generated in our modern technological world. Several different national standards exist. However, for our discussion of the nuclear fuel cycle, the following classification scheme is useful to distinguish among the different radioactive wastes encountered.

High Level Waste (HWL): These are the fission products produced by power reactors. They are separated from spent fuel in the first stage of fuel reprocessing and are appropriately named because of their very large activity. In the once-through fuel cycle, spent fuel itself is discarded as waste and hence also classified as HLW, even though it also contains fissile fuel and transuranic isotopes.

Transuranic Waste (TRU): These wastes are composed of plutonium and higher Z-number actinides and have an activity concentrations greater than 100 nCi/g. Such wastes are generated primarily by fuel reprocessing plants where transuranic fissile isotopes are separated from the fission products in spent fuel.

Mine and Mill Tailings: These are wastes from mining and milling operations and consist of low levels of naturally occurring radioactivity. The primary concern is the radioactive radon gas emitted from these wastes.

Low Level Waste (LLW): This is waste that has low actinide content ($<$ 100 nCi/g) and sufficiently low activity of other radionuclides that shielding is not required for its normal handling and transportation. This waste can have up to 1 Ci activity per waste package, but is generally of lesser activity distributed over a large volume of inert material. Such waste is usually placed in metal drums and stored in near-surface disposal sites.

Intermediate Level Waste (ILW): This category is rather loosely defined as wastes not belonging to any other category. This waste may contain $>$ 100 nCi/g of transuranic actinides and, generally, requires shielding when handled or transported. Such wastes typically are activated reactor materials or fuel cladding from reprocessing.

11.5.4 Spent Fuel

By far the most problematic of all radioactive wastes is that of spent fuel. During the three to four years a uranium fuel rod spends in a power reactor, much of the ^{235}U and a small amount of the ^{238}U are converted into fission products and transuranic isotopes. The typical conversion of uranium to these products in a LWR is summarized in Table 11.5.

Of the hundreds of different radionuclides produced as fission products, only seven have half-lives greater than 25 years: ^{90}Sr (29.1 y), ^{137}Cs (30.2 y), ^{99}Te (0.21 My), ^{79}Se (1.1 My), ^{93}Zr (1.5 My), ^{135}Cs (2.3 My), and ^{129}I (16 My). The

Table 11.5. Composition (in atom-%) of the uranium in LWR fuel before use and the residual uranium and nuclides created by fission and transmutation after use. After Murray [2001].

New Fuel		Spent Fuel	
Nuclide	Percent	Nuclide	Percent
^{238}U	96.7	^{238}U	94.3
^{235}U	3.3	^{235}U	0.81
		^{236}U	0.51
		^{239}Pu	0.52
		^{240}Pu	0.21
		^{241}Pu	0.10
		^{242}Pu	0.05
		fiss. products	3.5

latter five, with such long half-lives, are effectively stable, and thus the long-term activity of fission-product waste is determined solely by ^{137}Cs and ^{90}Sr (in secular equilibrium with its 54-hour half-life daughter ^{90}Y). After 1000 years, the fission-product activity will have decreased by a factor of $\exp[-(1000\text{ y}\ln 2)/30\text{ y}] \simeq 10^{-10}$, an activity less than the ore from which the uranium was extracted.

However, some transuranic isotopes in the spent fuel have much longer half-lives than ^{90}Sr and ^{137}Cs, notably ^{239}Pu with a 24,000 y half-life. It is these transuranic actinides that pose the greatest challenge for permanent disposal of spent fuel, requiring isolation of this HWL from the biosphere for several hundred thousand years.

Spent fuel reprocessing, as currently practiced by some countries such as France and England, allows fissile isotopes to be recovered and used in new fuel. In addition, this reprocessing of spent fuel allows the fission products to be separated from the transuranic radionuclides. As discussed above, storage of the fission products requires isolation for only about a thousand years; the transuranic nuclides can be recycled into new reactor fuel and transmuted or fissioned into radionuclides with much shorter half-lives. Although spent-fuel reprocessing can greatly reduce the length of time the radioactive waste must be safely stored, it poses nuclear weapon proliferation problems since one of the products of reprocessing is plutonium from which nuclear weapons can be fabricated. By keeping the plutonium in the highly radioactive spent fuel, it is less likely that it will be diverted and used for weapons. For this reason the U.S. currently elects not to reprocess its spent fuel.

In addition to fission products and transuranic nuclides, spent fuel accumulates radioactive daughters of the almost stable ^{238}U and ^{235}U, which were also present in the original uranium ore (see Figs 5.19 and 5.20).

HLW Disposal

If the spent fuel is reprocessed, the moist chemical slurries containing the separated fission products are first solidified by mixing the waste with pulverized glass, heating and melting the mixture, and pouring it into canisters where it solidifies into a glass-like substance from which the radionuclides resist leaching by water. If the spent fuel

rods are not to be reprocessed, the fuel assemblies can be placed in a container for ultimate disposal or, alternatively, the rods can be bundled together in a container and consolidated into a single mass by pouring in a liquid metal such as lead to fill the void regions.

The resulting solidified HLW is then placed in a permanent waste repository where it will be safely contained until virtually all the radionuclides have decayed into stable products. How such isolation may be achieved for the many tens of thousand of years has been the subject of much study and public debate. Some proposed HLW disposal techniques are summarized in Table 11.6.

Although no permanent HLW repository has yet been placed in service, most countries, including the USA, are planning on using geological isolation in mines. In the USA several sites around the country were investigated for geological suitability as a national HLW repository. Congress mandated in 1987 that the national USA HLW repository is to be established at Yucca Mountain about 160 km north of Las Vegas, Nevada, near the Nevada test site for nuclear weapons. Favorable characteristics of this site include a desert environment with less than seven inches of rain a year, a very stable geological formation, with the repository 2000 ft above the water table, and a very low population density around the site. Although no HLW waste has been stored at the Yucca Mountain repository, extensive site characterization and numerous experiments have been performed to validate the suitability of this site.

As presently envisioned, spent fuel assemblies would arrival by rail at Yucca Mountain, be placed in storage containers consisting of a 2-cm inner shell of a nickel alloy that is very resistant to corrosion and a 10-cm thick outer shell of carbon steel. The waste containers would then be placed in concrete-lined horizontal tunnels in the repository atop support piers that allow uniform heat flow from the containers. Initially, the waste packages could be retrieved; but, at some future date, the storage chambers would be backfilled, after possibly coating the packages with a ceramic shield. With these multiple barriers around the spent fuel, it is expected that no water would reach the waste for at least 10,000 years.

Disposal of LLW and ILW

Besides the HWL of spent fuel rods, a nuclear power plant also generates a much greater volume of solid LLW, about 1 m^3 per 10 MW(e)y of electrical energy production. This waste consists of slightly contaminated clothing, tools, glassware, and such, as well as higher activity waste from resins, demineralizers, air filters and so on. This solid waste is usually placed in drums and transported to a LLW repository where the drums are placed in near-surface trenches designed to prevent surface water from reaching the waste. A 1000 MW(e) nuclear power plant typically generates several hundred LLW drums a year.

Nuclear power plants also produce liquid LLW containing primarily tritium, which is readily incorporated into water as HTO molecules. It is not economically feasible to concentrate or separate the very small amount of HTO involved, and consequently such tritiated waste water is usually diluted to reduce the activity concentration and then dispersed safely into the environment since the amounts involved are dwarfed by the natural production of tritium.

Table 11.6. Possible permanent HLW disposal strategies.

HLW Disposal Concept	Comment
Geological Disposal in Mines: Put waste in underground mined chambers, backfill chambers, and eventually backfill and close the mine.	This is the current U.S. planned disposal method. It requires long-term seismically stable geological formation and the exclusion of ground water from the waste.
Seabed Disposal: Let waste canisters fall into the thick sediment beneath the seabed floor in deep ocean waters. Modifications include placing waste in deep holes drilled in the seabed or in subduction zones at edges of tectonic plates so that the waste is eventually drawn deep into the earth's mantle.	Tests in the 1980s showed such seabed disposal is feasible with very low diffusion of radionuclides in the sediment. There are obvious environmental concerns. Also using international waters presents legal/political difficulties, and inaccessibility makes monitoring or recovery difficult.
Deep Hole Disposal: Place waste in deep holes, e.g., 10 km deep, so the great depth will isolate the waste from the bioshpere.	Drilling such deep holes is likely to be very expensive and is currently beyond current drilling technology.
Space Disposal: Launch waste into interstellar space, into a solar orbit, or into the sun.	Weight of the encapsulation to prevent vaporization in the atmosphere should launch fail makes this option very expensive.
Ice Cap Disposal: Place waste canisters on the Antartic ice cap. The decay heat will cause the canister to melt deep into the ice cap coming to rest on the bed rock. Refreezing behind the canister isolates the waste from the biosphere.	There are important economic uncertainties and obvious environmental concerns. Moreover, the use of Antartica poses difficult political problems.
Rock Melting Disposal: Place waste in a deep hole where the decay heat will melt the surrounding rock and waste. Upon eventual refreezing, the waste will be in a stable solid form.	This technology is not well developed. Geological and environmental concerns are not yet addressed.
Injection into Wells: Inject wastes as liquids or slurries into deep wells using technology similar to that used in the oil and gas industry.	This scheme is used by some countries for LLW; but liquid waste can migrate in underground formations and their long-term safe isolation from the biosphere is not certain.
Waste Processing and Transmutation: Chemically separate the fission products from the actinides (TRU). Then use a reactor or accelerator to transmute the TRU into higher actinindes that decay more rapidly by spontaneous fission into relatively short-lived fission products.	This process converts the long-lived waste into fission products that need be stored safely for only several hundred years. This technology requires fuel reprocessing and the economic costs may be prohibitive.

Finally, it should be mentioned that a modern hospital using nuclear medical procedures also generates large quantities of LLW, often more than a nuclear power plant. Such hospital wastes consist of ^{3}H, ^{14}C and other radioisotopes that decay rapidly. Thus, after storage for a few weeks, only ^{3}H and ^{14}C remain. The amounts of these radioisotopes are negligibly small compared to their natural occurrence in the environment and could safely be disposed of by incineration and dilution of the exhaust gases. However, they are usually disposed of by shipping them to a LLW repository.

11.6 Nuclear Propulsion

Small power sources that can operate for extended periods without refueling are ideal for propulsion of vehicles that must travel large distances. Nuclear reactors are such power sources, and much development has gone into their use for propulsion. In the 1950s, the U.S. developed many designs and even some prototype reactors for nuclear powered aircraft. However, the obvious hazards posed by airborne nuclear reactors precluded the construction of such aircraft and this effort is now remembered as an interesting footnote in the history of nuclear power.

Today, nuclear power reactors are being planned as power sources for space missions, both as an electrical power source (see the next chapter) as well as a source of propulsion for deep space missions. However, by far the most successful use of nuclear power for propulsion has been in ships, particularly in modern navies.

11.6.1 Naval Applications

The potential of nuclear power for ships was immediately recognized. The ability of a nuclear powered ship to travel long distances at high speeds without refueling was highly attractive to the military. Moreover, a nuclear submarine could remain submerged almost indefinitely since the reactor needed no air, unlike the diesel engines used to charge batteries in a conventional submarine.

In 1946 the legendary Admiral Rickover assembled a team to design and build the first nuclear-powered ship, the submarine *Nautilus*. This prototype submarine used a small water-moderated pressurized reactor with highly enriched uranium fuel. In this design, the primary pressurized liquid water passes through a steam generator where secondary water boils and the steam is used to turn a turbine which, in turn, drives the propeller shafts of the submarine. The *Nautilus* was launched in 1954 and soon broke many submarine endurance records. It was the first submarine to reach the north pole by traveling under the Arctic ice cap, it travelled extended distances at speeds in excess of 20 knots, and travel almost 100,000 miles on its second fuel loading.

Nuclear submarines can travel faster underwater than on the surface and can travel submerged at speeds between 20–25 knots for weeks on end. By comparison, conventional World War II submarines could travel only a maximum speed of eight knots submerged and do so for only an hour before needing to surface to recharge their batteries. In 1960, the nuclear submarine *Triton* followed the route taken by Magellan in the sixteenth century to circumnavigate the world. The 36,000 mile voyage took Magellan nearly three years; the *Triton* completed the trip, entirely submerged, in 83 days! Nuclear power has revolutionized the strategic importance of submarines with their capability to launch missiles while submerged and to hunt

submerged enemy submarines. At the end of the cold war in 1989, there were 400 nuclear-powered submarines, either in operation or being built. Russia and the U.S. had in operation over 100 each, with the UK and France less than 20 each and China six. About 250 of these submarines have been scrapped or their construction cancelled as a result of weapons reduction programs. Today there are about 160 nuclear submarines in operation.

Nuclear powered surface vessels have also been added to the navy. Both the Soviet Union and the U.S. have deployed nuclear-powered cruisers, and several countries, including the U.S., have nuclear-powered aircraft carriers. The U.S. has the most nuclear-powered aircraft carriers, the first being the 1960 USS *Enterprise* powered by eight reactors, followed by nine other carriers with two reactors each.

All naval reactors are PWRs with compact cores fueled by rods composed of a uranium-zirconium alloy using highly enriched uranium (originally about 93% but today about 20–25% in the U.S. cores and about 50% in Russian cores). The cores can operate for 10 years without refueling, and newer designs produce cores with lifetimes of 30-40 years in submarines and 50 years in aircraft carriers. Maximum thermal power of these cores ranges up to 190 MW in large submarines and surface ships. The Russian, U.S., and British vessels use secondary-loop steam to drive a turbine which is connected, through a gearbox, to the propeller shafts. By contrast, the French and Chinese use the turbine to generate electricity for motor driven propeller shafts. All surface vessels since the *Enterprise* and all Russian submarines use two reactors; all other submarines are powered by a single reactor.

11.6.2 Other Marine Applications

The same benefits that nuclear power gives naval ships also apply to civilian ships. For cargo ships, nuclear power eliminates the need for oil fuel tanks or coal bins thereby making more space available for cargo. Also the higher cruising speeds and the greatly reduced time needed for refueling allow better ship utilization. Three nuclear merchant ships have been built and commissioned.

In 1959 the U.S. launched the *NS Savannah* a demonstration freighter which could also carry passengers (60 cabins). The *Savannah* was almost 600 feet long with a displacement, when fully loaded, of 20,000 tons. Her cruising speed was 21 knots and was powered by a pressurized reactor using 4.4% enriched fuel and with a maximum thermal power of 80 MW. This first nuclear-powered commercial vessel was intended as a demonstration of the peaceful uses of nuclear energy and made many goodwill voyages to ports around the world for several years in the 1960s. It was decommissioned in 1970.

Germany built and operated Western Europe's first nuclear merchant ship, the *Otto Hahn*. This demonstration vessel used a pressurized water reactor, which was very similar to that used in the *Savannah*. The *Otto Hahn* sailed some 650,000 miles on 126 voyages over ten years without experiencing any technical problems. However, because of its high operation expense, it was converted to diesel power.

Japan launched the nuclear-power merchant ship *Mutsu* in 1962. This merchant ship also used a pressurized water reactor and was operated for several years. However, it suffered both technical and political problems, and was prematurely decommissioned and now resides at a naval museum.

From these three demonstration merchant vessels, several other advanced and larger nuclear-powered merchant marine vessels were designed. However, no nuclear-powered merchant ship operates today, primarily because of the large capital costs associated with nuclear ships compared to diesel-powered ships.

There is one country that has found nonmilitary ships to be technically and economically feasible. To keep its northern shipping lanes open in the winter, Russia operates several icebreakers. Ice breaking requires powerful ships which consume large amounts of energy. Diesel-powered icebreakers need frequent refueling and thus cannot navigate the entire Arctic basin. In 1959 the world's first nuclear-powered icebreaker, the *Lenin*, joined the Arctic fleet and remained in service for 30 years, although new reactors were fitted in 1970. Russia has since built several other nuclear-powered icebreakers. The large, two-reactor, Arktika-class icebreakers are used in deep Arctic waters. Such an icebreaker was the first surface ship to reach the North Pole. For shallower waters, Russia is now building the one-reactor, Taymyr-class icebreakers.

Although today non-military marine nuclear propulsion is not economically feasible, the U.S. nuclear navy has benefited the nuclear-electric industry in two important ways. Much of the technology developed for naval reactors has been widely used in the design of civilian pressurized-water power reactors. Also many highly skilled personnel in the civilian nuclear power industry have obtained their nuclear background from earlier service in the nuclear navy.

11.6.3 Nuclear Propulsion in Space

Because the energy content of nuclear fuel in a reactor is about 10^8 times that in an equal mass of chemical reactants, nuclear fission power offers far greater propulsive capability than conventional chemical rockets, and, consequently, is ideally suited for deep space missions. Although no nuclear space propulsion systems have yet been launched, there have been extensive design studies and even ground tests of various ways of converting nuclear fission energy into propulsive thrust.

Two basic approaches are being pursued. The first called *nuclear thermal propulsion* mimics conventional chemical rockets in which reactants are combined to produce high-temperature gases which are allowed to escape at high speed from the rear of the rocket. An alternative, called *electric propulsion*, is first to convert nuclear thermal energy into electrical energy and then to use this energy in electromagnet devices to eject atoms at very high speeds from the rear of the spacecraft.

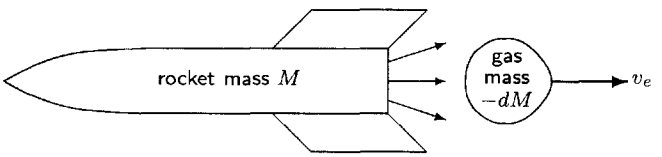

Figure 11.24. Rocket of mass M ejecting a mass of $-dM$ of hot gas with speed v_e in a time interval dt.

Thermal Propulsion

In thermal propulsion, a hot gas, whose atoms or molecules are moving with high speeds, is allowed to expand through the rocket nozzle and escape into space behind the rocket, thereby, providing a forward thrust. To understand the basic physics, consider a rocket of mass M that emits, in time dt, a mass $-dM$ from its rear with a speed v_e (see Fig. 11.24). Here dM is the mass *loss* of the rocket, a negative quantity. This gas release increases the rocket's speed by dv. From the principle of conservation of linear momentum, the gain in momentum of the space craft, $M\,dv$, must equal the momentum of the emitted gas, $(-dM)v_e$, namely, $M\,dv = (-dM)v_e$. From this relation we obtain

$$-\frac{dM}{M} = \frac{1}{v_e}dv. \tag{11.2}$$

Integration of this differential equation from the rocket's initial state, when it had mass M_o and speed v_o, to the time it has mass $M(v)$ and speed v, yields

$$-\int_{M_o}^{M(v)} \frac{dM}{M} = \frac{1}{v_e}\int_{v_o}^{v} dv, \tag{11.3}$$

from which we obtain

$$\frac{M(v)}{M_o} = \exp[-(v - v_o)/v_e]. \tag{11.4}$$

The fuel mass consumed to give the rocket a speed v is

$$M_f(v) = M_o - M(v) = M_o\left\{1 - \exp[-(v - v_o)/v_e]\right\}. \tag{11.5}$$

From this result we see that the fuel mass needed for a given increase in speed of the rocket, $v - v_o$, decreases as the speed of the ejected exhaust atoms or molecules increases. The average molecular speed in a gas in thermal equilibrium at an absolute temperature T is $v_e = \sqrt{(8kT)/(\pi M_e)}$, where M_e is the mass of the exhaust gas molecules and k is Boltzmann's constant ($1.3806503 \times 10^{-23}$ J K^{-1}). Because v_e is proportional to $\sqrt{T/M_e}$, to reduce the amount of fuel needed to achieve a rocket speed v, the temperature of the ejected gas must be increased and/or the molecular weight of the exhaust molecules must be decreased.

Present chemical rockets used for space launches combine liquid hydrogen with the oxidizer O_2 to produce water (H_2O) as the exhaust gas with a molecular weight of 18. By using nuclear reactors to heat hydrogen molecules (molecular weight 2) to the same temperature, the mass of the escaping hydrogen molecules would be nine times smaller, requiring almost three times less fuel for the rocket to achieve the same speed.

A very useful figure of merit for a rocket engine is the *specific impulse* I_{sp} defined as the thrust or force F exerted by the engine to accelerate the rocket divided by the fuel mass consumption rate (equal to the rate at which mass is lost by the rocket $-dM/dt$). The thrust is found from Eq. (11.2) by dividing by dt, namely,

$$F \equiv M\frac{dv}{dt} = \left(-\frac{dM}{dt}\right)v_e. \tag{11.6}$$

Hence the specific impulse is

$$I_{sp} \equiv \frac{F}{-dM/dt} = v_e. \tag{11.7}$$

To create a thermal nuclear rocket engine, a reactor core composed of heat resistant material such as graphite is cooled by pumping liquid hydrogen through components around the core to keep them cool, and then the resulting hydrogen gas enters channels in the core where it is heated to a temperature as high as the core can withstand. The hot hydrogen gas then expands through a nozzle to produce the rocket thrust.

Thermal nuclear engines were considered as early as 1946 for the initial designs of Intercontinental Ballistic Missiles (ICBMs); however, conventional chemical rockets were selected. In the early 1950s nuclear engines were again considered by the U.S. Air Force for nuclear powered aircraft as well as for rocket use. In 1956 project Rover was begun at Los Alamos National Laboratory culminating in several tests of the Kiwi reactor engine. In the late 1950s, the Air Force lost interest in nuclear rockets since the chemically propelled ICBMs had proven themselves. With the launching by the Soviet Union in 1957 of Sputnik I, the National Aeronautics and Space Administration (NASA) developed an interest in nuclear rockets and the NERVA (Nuclear Engine for Rocket Vehicle Application) Program was started in 1960. A series of rocket engines, named Kiwi, NRX, Phoebus, Pewee, and XE′, were built and tested throughout the 1960s at the Nuclear Rocket Development Station in Nevada. The NERVA program convincingly demonstrated the technical feasibility of thermal nuclear rocket engines; but it was terminated in 1973 when NASA withdrew its support.

During the NERVA program several fuels were developed for graphite-moderated, once-through, hydrogen-cooled reactor cores.

- Beaded loaded graphite consisted of a graphite matrix containing a multitude of very small spheres of fuel coated with pyrocarbon. Reactor tests with this fuel achieved a temperature of 2500 K for 1 hour.
- A composite fuel of 30–35 volume% of UC.ZrC dispersed in graphite. This fuel could sustain temperatures of 2700 K for at least an hour.
- A pure carbide fuel such as UC.ZrC could maximize the reactor's time-temperature performance. However, this fuel is difficult to fabricate and insufficient testing was done. Temperatures around 3000 K are thought possible with this fuel, yielding a specific impulse I_{sp} of nearly 9.5 km/s, compared to chemical rocket engines which have specific impulses ranging from 1.5 to 4.5 km/s.

Although nuclear rockets are not used today, partially because of the present emphasis on near-earth manned space missions, NASA has a manned mission to Mars scheduled for 2014. To move the spacecraft between Earth and Mars, a NERVA-type nuclear engine would greatly reduce the transit time to Mars and back compared to using chemical engines.

Electric Propulsion

An alternative to thermal nuclear rockets is first to convert thermal energy from a reactor to electrical energy (see the next chapter), and then use this electrical energy to accelerate ions to very high speeds, passing them through a neutralizer in the rocket nozzle, to produce a beam of very fast moving neutral atoms leaving

the rocket engine and producing a forward thrust. Several technologies have been proposed for creating the fast ions: (1) thermoelectric heating in which electricity is passed through the propellant to ionize and heat the gas, (2) electromagnetic devices to confine, heat and accelerate a plasma of the propellant ions, and (3) electrostatic devices that accelerate the ions between two electric grids as in an ion accelerator.

Table 11.7. Specific impulses of different rocket engines. After [Niehoff and Hoffman].

Engine Type	I_{sp} (km/s)
chemical	1.5–4.5
thermonuclear	8.3–9.2
electrothermal	8–12
electromagnetic	20–50
electrostatic	35–100

Such electric rocket engines produce very low thrusts but also have very small fuel flows and high exhaust speeds v_e and, hence, high specific impulses I_{sp} (see Table 11.7). Design and construction of electric propulsion systems began in the 1940s and by 1990 more than 80 such systems were tested in orbital missions, the majority by the Soviet Union.

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PROBLEMS

1. In a BWR or PWR, steam is generated with a temperature of about 290 °C. If river water used to receive waste heat has a temperature of 20 °C, what is the maximum possible (ideal) conversion efficiency of the reactor's thermal energy into electrical energy? Nuclear power plants typically have conversion efficiencies of 34%. Why is this efficiency less than the ideal efficiency?

2. A 1000 MW(e) nuclear power plant has a thermal conversion efficiency of 33%. (a) How much thermal power is rejected through the condenser to cooling water? (b) What is the flow rate (kg/s) of the condenser cooling water if the temperature rise of this water is 12 °C? Note: specific heat of water is about 4180 J kg^{-1} C^{-1}.

3. What are the advantages and disadvantages of using (a) light water, (b) heavy water, and (c) graphite as a moderator in a power reactor.

4. Why are the blades of a low-pressure turbine larger than those of a high-pressure turbine?

5. Why can a heavy-water moderated reactor use a lower enrichment uranium fuel than a light-water moderated reactor?

6. Explain whether the turbine room of a BWR is habitable during normal operation.

7. If the demand on the generator increases (i.e., a greater load is placed on the turbine), explain what happens to the reactor power of (a) a PWR and (b) a BWR if no operator-caused reactivity changes are made. Which reactor *follows* the load?

8. Although the steam cycle is simpler in a BWR, explain why the capital costs of BWR and PWR plants are very competitive.

9. Explain the advantages and disadvantages of using helium instead of water as a coolant for a power reactor.

10. During the April 1986 accident in the 1000-MW(e) RMBK Chernobyl reactor, the water in the cooling tubes of the graphite-moderated reactor was allowed, through operator error, to boil into steam and cause a supercritical, run-away chain reaction. The resulting energy excursion resulted in the destruction of the reactor containment and a large amount of the fission products in the fuel elements were released to the environment as the reactor containment ruptured. Explain why the boiling in the cooling tubes led to supercriticality.

11. How many years are required for the initial activity to decrease by a factor of 10^{10} for (a) ^{137}Cs, (b) ^{90}Sr, and (c) ^{239}Pu?

12. Over a period of one year what mass (in kg) of fission products is generated by a 1000 MW(e) power reactor?

13. Discuss possible environmental, technical and politcal problems associated with each of the disposal options listed in Table 11.6.

14. A nuclear drive in a submarine delivers 25,000 shaft horse power at a cruising speed of 20 knots (1 knot = 1.15 miles/h). If the power plant has an efficiency of 25%, how much (in kg) of the ^{235}U fuel is consumed on a 40,000 mile trip around the world?

15. Reactors for naval vessels are designed to have very long lifetimes without the need to refuel. Discuss possible techniques that can be used to maintain criticality over the core lifetime as ^{235}U is consumed.

16. A thermal nuclear rocket using hydrogen as the propulsive gas operates for one hour at a thermal power of 4000 MW and a temperature of 2700 K. Estimate (a) the amount (in grams) of fissile material consumed, (b) the specific impulse of the engine.

17. Estimate the specific impulse I_{sp} for (a) a chemical rocket burning hydrogen and oxygen at a temperature of 4000 K, and (b) a thermal nuclear rocket emitting hydrogen at 3000 K.

18. A nuclear rocket propulsion system uses an ion drive to accelerate mercury atoms to energies of 50 keV. Estimate the specific impulse of this drive.