CHAPTER 10

Nuclear Power Plants*

10.1 Introduction

Nuclear power is universally controversial. Many would say that it is also universally needed–as an alternative or supplement to power generated by fossil fuels.

The combustion of fossil fuels produces carbon dioxide, now notorious for the threat of global warming. Nuclear power plants produce neither carbon dioxide nor oxides of sulfur and nitrogen, as does the burning of fossil fuels. Thus nuclear power reduces the global production of carbon dioxide and other pollutants, and helps to alleviate many of the pervasive problems of fossil fuel supply.

Petroleum is least available in regions of widest use; natural gas is, for the time being, plentiful and sought after by all; and widely abundant coal has come to be regarded as the great Satan of air pollution. Water power is important, but it offers limited possibility for growth. Solar energy, while promising, is far from being a mainstay of the world's energy supply. Thus sources other than fossil fuels and nuclear power offer little hope to become major suppliers during our lifetimes.

Nuclear power, in stasis for many years, may make a comeback. Engineers have been quietly working on new and safer designs for nuclear power plants, and the political climate may be swinging slowly back in favor of nuclear power. According to references 31 and 34, there were 434 operating nuclear plants producing 17% (350,000 megawatts) of the world's electricity in 1998. Regardless of one's position towards it, nuclear power is a major factor in world power production.

Knowledge of nuclear power is not American, French, Indian, or British; it is virtually universal. Nuclear power plants such as those shown in Figure 10.1 are producing power in many nations around the world. Blockage of the growth of nuclear power in the United States did not prevent the development and extensive use of nuclear-generated electricity in France or Canada. Should developing countries with minimal fossil resources not use nuclear power? Should a country that has seen the terror wrought by nuclear weapons be denied the benefits of electricity from the nucleus? Can attempts to halt the growth of nuclear power stop the proliferation of nuclear weapons? These and many other issues around nuclear power, which obviously extend far beyond the bounds of engineering, are much too broad to be pursued here. Nuclear power is controversial, but it is here; it is important and likely to remain so.

*Thanks to Dr. Andrew A. Dykes for his valuable inputs and comments on this chapter.

FIGURE 10.1 Bruce Nuclear Power Development in Western Ontario. Bruce A (3300 MW) is at top, and Bruce B (3660 MW) is in the foreground. The Douglas Point Nuclear Station is at far left, and the Bruce heavy-water plants are just to the right. (Courtesy of Ontario Hydro, Toronto, Canada.)

This chapter is concerned largely with nuclear fission power reactors. Most nuclear power plants use steam cycles that differ little from those of fossil fuel plants except for the source of heat for the steam generator and for steam supply conditions. Steam turbine cycles were considered in some detail in Chapter 2. Thus this chapter will focus primarily on (a) the characteristics of nuclear reactor steam supply systems and, to a lesser extent, on (b) the climate for future nuclear development. In preparation for this study let us first review relevant aspects of atomic and nuclear structure.

10.2 Review of Atomic Structure

Atoms and Molecules

Somewhat more than a hundred *elements* are known and are thought to be the building blocks of everything in the universe. The *atom* is the basic unit of structure for each element. An important connection between the microscopic world of the atom and the macroscopic world of experience is given by *Avogadro's number*. A gram-mole of any element has Avogadro's number (6.023 x 10^{23}) of atoms.

The atom may be considered as consisting of a positively charged *nucleus* at its center and one or more negative charges around the nucleus called electrons that make the atom electrically neutral. The *electron* is the fundamental unit of negative charge. It may be viewed as a particle which is much smaller than the nucleus and which orbits around the nucleus as a planet orbits the sun, or it may be viewed as a diffuse electron cloud around the nucleus. Still another concept is that of a particle whose location is not known but which is more likely to be in some places than others, its position being given by a probability distribution. We will not concern ourselves with the rationales for these views.

Thus atoms consist of nuclei surrounded by electrons. The sizes of atoms are conveniently measured in *Angstroms* (10^{-8} cm). The nucleus typically is of the order of 10^{-5} Angstroms. Thus the volume of the atom is largely due to the size of the outer electron's orbit or to the atom's electron cloud.

Molecules are collections of atoms held together by electromagnetic forces between the nuclei and the electrons. Atoms and molecules can exist in a variety of energy states associated with their electron distributions. These microscopic states and their macroscopic influences are dealt with theoretically in the fields of quantum mechanics and statistical thermodynamics.

Molecules and atoms can interact with each other to form different molecules in ways that are controlled by their electron structures. These interactions, called *chemical reactions,* have little to do with the nucleus. In Chapter 3, we considered aspects of these reactions that are relevant to combustion. The magnitudes of the energy associated with these chemical changes, while of great importance in thermal engineer-ing, are so small that they have no significant influence on the nuclei of the reacting atoms. Thus the nuclei may be thought of as merely going along for the ride when a chemical reaction occurs.

An electrically neutral particle, however, can penetrate an atom's electromagnetic field and approach the nucleus, where it interacts via short range but powerful nuclear forces. Then the electrical forces holding the nucleus together may be overcome, resulting in changes in the nucleus. In these cases the interatomic forces are largely irrelevant and are overpowered by nuclear events. It is these changes that are the concern of this chapter.

The Nucleus

The nucleus, for our purposes, may be thought of as being made up of integral numbers of protons and neutrons. The *proton* is a particle with a positive charge of the same magnitude as that of the electron, so that pairing a proton with an electron produces exact electrical neutrality. Thus protons account for the charge of the nucleus, and a like number of electrons ensures the electrical neutrality of the atom. Compared with the electron, the proton is a massive particle, having a mass which is about 1800 times the mass of the electron. Thus the hydrogen atom, which consists of one proton in the nucleus and a single electron in orbit around the nucleus, is electronically neutral and has a mass only slightly larger than that of the nucleus.

Atoms larger than the hydrogen atom have more than one proton in the nucleus and have one or more neutrons as well. A *neutron,* as the name suggests, is an electrically neutral particle with a mass only slightly larger than that of the proton. As components of the nucleus, protons and neutrons are called *nucleons*, and are thought of interchangeably with respect to mass because their masses differ so little from each other.

The number of protons in an atom of an element is called the *atomic number* of the element. Thus hydrogen has an atomic number of 1. The atomic number of a given element is unique to that element. Thus we could identify the elements by their atomic numbers rather than by their names if we wished. Elements are ordered in the periodic table in part by their atomic numbers.

The *mass number* of an element is the number of nucleons in an atom of that element and is therefore the sum of the number of protons and neutrons in the nucleus. Atoms of a given element that have differing mass numbers are called *isotopes* of the element. A given isotope of an element is sometimes designated by a notation that includes the element's chemical symbol, its mass number, and its atomic number. For example, the most common isotope of uranium is denoted as ${}_{92}U^{238}$, where 92 is the atomic number of the element uranium and 238 is the sum of the number of protons and neutrons in the isotope nucleus. The isotopes are also sometimes simply identified by their name or symbol and mass number, such as U-235 or Uranium-235. Other significant examples are the isotopes of hydrogen, *deuterium,* ${}_{1}H^{2}$, and *tritium*, ${}_{1}H^{3}$, which have, respectively, one and two neutrons accompanying the proton. These isotopes are sometimes written ${}_{1}D^{2}$ and ${}_{1}T^{3}$ to reflect their commonly used names. The form of water, H_2O, in which the isotope deuterium replaces hydrogen is commonly called *heavy water*, D_2O, because of the added mass of the extra neutron in each nucleus. It will be seen later that heavy water has characteristics that are advantageous in some nuclear processes.

10.3 Nuclear Reactions

Just as chemical fuels undergo chemical reactions that release energy, nuclei may also participate in energy-releasing nuclear reactions. When this happens atoms of the reacting elements are converted to atoms of other elements, the sort of *transmutation* sought by the alchemists of the past.

Mass-Energy Equivalence

Two nuclear processes are known to be capable of releasing energy on a scale large enough to influence the personal and business lives of humans. The reality of both processes has been amply demonstrated by the production and detonation of the atomic and hydrogen bombs. Both processes owe their energy release to the annihilation of matter consistent with the famous *Einstein formula*, $E = mc^2$, which asserts the convertibility of mass to energy. Because the speed of light is so large, the equation shows that the annihilation of a small amount of mass yields a large quantity of energy. These energy releases are usually measured in MEV, millions of electron-volts. The *electron-volt*, EV, is defined as the energy required for an electron to pass through a potential difference of one volt. The reactions of individual nuclei typically produce particles with energies measured in MEV. On the other hand, as indicated earlier, the most energetic of chemical reactions releases much less energy, only a few EV per molecule.

As a result of our encounter with Einstein's energy-mass relation we must adjust our philosophical position on the conservation laws of mass and energy and think instead in terms of conservation of mass-energy. Mass and energy may be thought of as different forms of the same thing. Any change in the mass of an isolated system must be accompanied by a corresponding change in system energy. This in no way influences the discussions in previous chapters, because changes in mass are entirely insignificant in chemical and other nonnuclear processes.

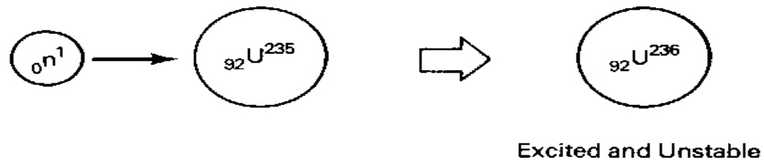
Fission and Fusion

The process known as *nuclear fusion* occurs in nature in the stars, including our own sun. Since the Second World War, scientists have been attempting to achieve the conditions for fusion in the laboratory. Because it can use heavy water from the sea as a fuel, *controlled thermonuclear fusion* offers the hope of vast quantities of power for many centuries in the future.

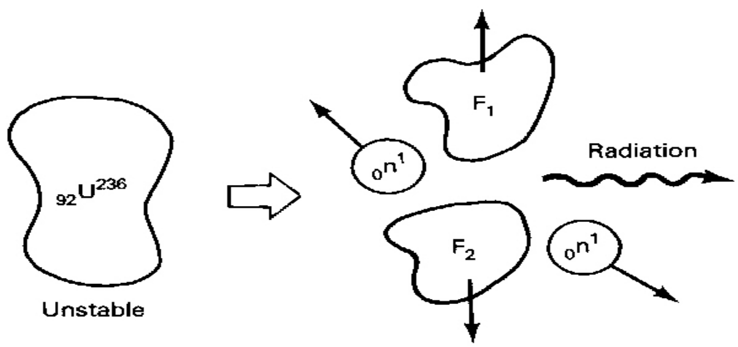
Fusion occurs when light atoms interact to form a heavier atom in reactions such as

$${}_1D^2 + {}_1D^2 \Rightarrow {}_2He^3 + {}_0n^1 + 3.2 \text{ MEV}.$$

Here, two deuterium atoms collide to form helium-3 and a neutron while releasing 3.2 MEV of energy. Other fusion reactions exist that provide comparable amounts of energy. Using precise atomic masses measured with mass spectrometers, we can determine the differences of the masses of the reactants and products in this reaction. Application of the Einstein relation to the mass loss yields the same energy release (in this case 3.2 MEV) as is obtained by energy measurements. Thus the energy yield of



(a) Capture of a Neutron to form the Unstable Isotope, U-236.



(b) Creation of Fission Fragments from U-236.

FIGURE 10.2 The two parts of the fission process.

known nuclear reactions may be determined with only mass measurements.

For over fifty years, researchers have pursued the goal of achieving controlled thermonuclear fusion on a scale suited for commercial power production. Since the reactants are two positively charged nuclei, they must have high kinetic energies to overcome their mutual repulsion. These high energies imply a gaseous state with enormously high temperature, a condition known as a *plasma*. Because solid materials cannot exist at plasma conditions and plasmas would be cooled by the presence of solids, magnetic confinement of plasmas has been one approach to achieving a thermonuclear plasma. Large experimental devices called Stellerators, Tokamaks, and mirror machines have been built to help solve the problems inherent in achieving large-scale fusion reactions and in stably confining the associated thermonuclear plasma. While progress continues, controlled thermonuclear fusion remains, and will likely continue, in the research stage for many years. It will therefore not be considered further here.

Whereas nuclear fusion annihilates mass by forming larger atoms from light atoms, *fission* is a process of breaking massive atoms into two large, more-or-less equal-sized atoms, with an accompanying mass loss and energy release. While controlled fusion remains elusive, nuclear fission has been producing electrical power on a commercial scale for decades. The remainder of this chapter therefore deals with the fundamentals and commercial use of nuclear fission.

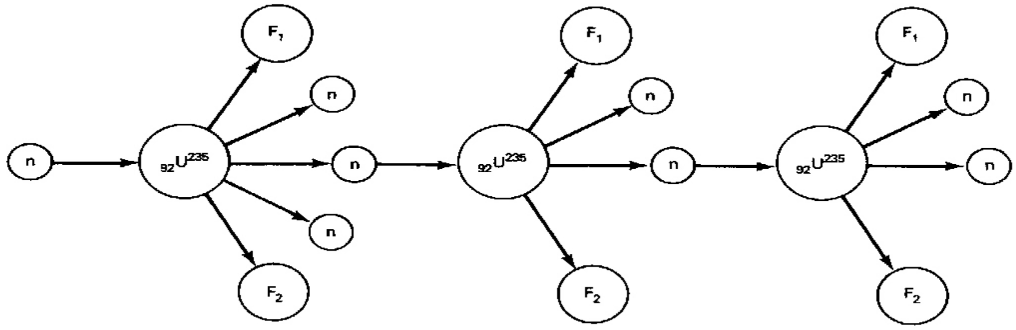


FIGURE 10.3 A U-235 self-sustaining chain reaction. After coming to rest, the fission fragments become nuclear wastes that generate heat when they decay, and the nonfissioning neutrons are absorbed in surrounding materials or escape from the reactor core.

10.4 Fundamentals of Nuclear Fission

Nuclear fission involves the breakup of certain massive elements resulting from collisions with neutrons. Uranium U-235, for instance, the isotope of uranium with 235 nucleons, forms an highly excited isotope with 236 nucleons upon capture of a neutron, as in Figure 10.2(a). These excited U-236 atoms are unstable and break up into a variety of pairs of large atoms shortly after they are created. The many such fissions occurring in a reactor may be expressed as an average reaction:

$${}_{92}U^{235} + {}_{0}n^{1} \Rightarrow {}_{92}U^{236} \Rightarrow F_1 + F_2 + 2.47\ {}_{0}n^{1} + 203 \text{ MEV}$$

where F_1 and F_2 represent *fission fragment* elements, as in Figure 10.2(b). On the average, 2.47 neutrons are released in the process, one of which must initiate another fission to sustain a *chain reaction*, as illustrated schematically in Figure 10.3. In addition alpha, beta, and gamma radiation is released. One of the many reactions that participate in the average reaction above creates xenon and strontium fission fragments and two neutrons:

$${}_{92}U^{235} + {}_{0}n^{1} \Rightarrow {}_{92}U^{236} \Rightarrow {}_{54}Xe^{139} + {}_{38}Sr^{95} + 2\ {}_{0}n^{1} + \text{energy}$$

U-235 reactions exemplified by the Xe-Sr reaction create diverse fission fragments and small integral numbers of neutrons that average to 2.47 and release energies that average to 203 MEV.

Over 80% of the 203 MEV of energy released by the average U-235 fission reaction is the kinetic energy of the fission fragments associated with their large mass and high velocity. Because of their high energy, the fission fragments become ionized and ionize nearby atoms as they tear their way through surrounding materials. The fission fragments, however, travel only a very short distance, for interactions cause them to lose much of their energy to the surrounding solid. Thus most of the fission energy appears immediately as internal energy and therefore locally high temperature of the surrounding materials. It is, therefore, necessary that the fuel be cooled continuously to counter the *fission heat generation* to avoid temperature buildup and possible melting. The heated coolant is then provides the energy input for a thermodynamic (usually steam) cycle.

As the fission fragments come to rest, their presence changes the character of the reactor materials. Non-fissionable material exists where fissionable and other atoms once resided. In time these materials may decay radioactively, forming still different substances and releasing additional heat. But, importantly, they absorb neutrons without offering the possibility of fission. Thus the fission fragments are said to *poison* the reactor.

Fission Reactor Design Considerations

While most of the fission heat generation is due to the fission fragments, reactor design focuses on the actions of neutrons. This is because neutron captures produce fissions, fissions produce more neutrons, and, as we will see later, fissile material may also be produced using neutrons. Thus neutrons are the currency of the reactor and may be used constructively or wastefully. The manner in which the neutrons in a reactor are used is called the *neutron economy*. The reactor designer pays close attention to all of the details of the neutron economy. One of these details is the distribution of the kinetic energy of the neutrons, that is, the fractions of the neutrons that lie in given energy ranges.

Almost all commercial *power reactors* (as opposed to experimental reactors) are *thermal reactors* in which fission is caused by thermal neutrons. A *thermal neutron* is a neutron that is in thermal equilibrium with the surrounding atoms. This implies that they have energies on the order of 0.02 EV. Because they move relatively slowly, thermal neutrons are much more likely to cause fission of U-235 than are higher-energy neutrons; therefore, they are the choice for most power reactor designs. However, the neutrons created by the fission reaction are *not* thermal neutrons. The neutrons created by fission have kinetic energies that range from about 1 to 10 MEV. They are called *fast neutrons* because of this high kinetic energy.

Thus, if a chain reaction is to be sustained in a thermal reactor, it is necessary for the fast neutrons to be slowed, or *thermalized,* to much lower energies so that they can cause fissions before they are absorbed in nonproductive captures in reactor material or before they escape from the reactor. They do this by colliding with certain other nuclei in the core, put there for that purpose. These nuclei are called *moderators*. A good moderator is a light element that, on collision with a neutron, is speeded up by the collision, and thereby extracts energy from the neutron without absorbing it.

The moderator concept may be understood by considering what happens when a billiard cue ball hits another ball head-on. The cue ball stops, and the second ball carries away the kinetic energy. However, with balls of differing mass, if the second ball is heavy the cue ball bounces off without losing energy, whereas light balls are propelled at high speed and extract significant energy from the cue ball. In neutron collisions with atoms, it is the lightest atoms that extract the most energy.

Ordinary water, heavy water, graphite, and beryllium are all used as moderators because they are light and are poor absorbers of neutrons. Hydrogen and deuterium are the moderators in water. The oxygen in the water is not an effective moderator but has low absorption and therefore does not interfere significantly with the moderation process.

It is evident that conservation of neutrons is a prime consideration in any reactor design. The number of neutrons per unit volume, or *neutron density*, in a reactor is an important design parameter. Neutrons diffuse about in the reactor as they are scattered and slowed by moderator atoms. Because they have no charge, they are uninfluenced by electromagnetic fields and therefore may travel further than charged particles.

Four events can influence local neutron densities as they pass through the surrounding reactor core:

1. The neutrons can be *captured* by a fissionable atom and produce a fission.
2. They can be *absorbed* non-productively by fission products, structural materials, or nonfissioning fuel.
3. They can *escape* through the walls of the reactor.
4. They can be absorbed in nuclei that *create* more fuel.

The last possibility will be considered in Sections 10.5 and 10.8.

Events 2 and 3, absorption and escape, result in nonproductive waste of neutrons in the neutron economy. The necessity that enough events of type 1, rather than types 2 and 3, occur to sustain the chain reaction suggest several important considerations for reactor design:

- The reactor should be large enough that only events of types 1, 2, and 4 take place and hence that escape of neutrons from the reactor is rare. Since reactor size is dictated primarily by the cooling requirements imposed by nuclear heat generation, this usually follows automatically from the design process. In addition, positioning moderating material such as water at the boundaries of the reactor as a *reflector*, to deflect escaping neutrons back into the reactor, may allow a more compact design, in some cases.

- Materials to be used in the reactor design are selected so that type 2 events are minimized. The gradual buildup of poisons must also be considered in designing for the change in reactor performance with time between refuelings.

- The reactor should be designed to minimize the amount of structural materials in the active fuel region, to reduce the frequency of type 2 events.

- Neutron-absorbing materials may be moved into and out of the reactor to change the average neutron density for *reactor control* purposes.

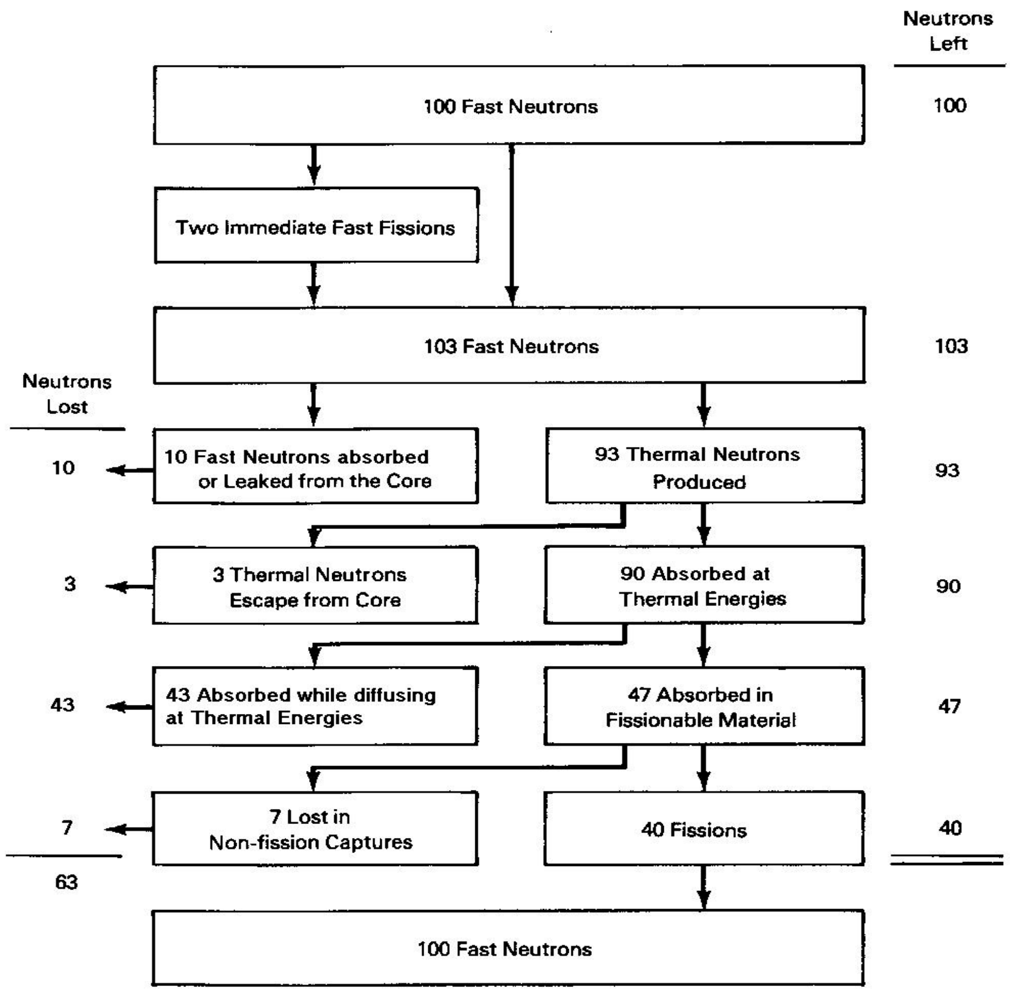


FIGURE 10.4 Neutron economy. Tracking a single generation of 100 fast neutrons.

A reactor can function over a range of neutron densities. When the neutron density and power levels are constant, the reactor is said to be *critical*. This implies that the number of neutrons producing fission at one instant is the same as at a later instant, a situation depicted in Figure 10.4. Since high neutron densities produce more fissions, each of which produces 203 MEV in the case of a U-235 thermal reactor, heat generation, and thus power output, increases with neutron density.

Let's assume that 2.5 fast neutrons are produced per fission in a critical thermal reactor and track the activities of 100 fast neutrons created in an instant. In a critical reactor, about 40 of these fast neutrons must thermalize and undergo fission to produce another 100 fast neutrons. Figure 10.4 shows that 2 of the original 100 fast neutrons produce (fast) fissions and 5 more neutrons immediately so that there are 103 fast neutrons diffusing around. Of these, about 10 escape from the core or are absorbed in non-fissile materials while slowing down. Of the remaining 93 thermal neutrons, about

3 escape from the core and about 43 are absorbed in non-U-235 materials while diffusing around at thermal energies. The remaining 47 captures by U-235 result in 7 non-fission captures and 40 fissions. These 40 fissions, in turn, produce 100 fast neutrons for the next generation in a reactor operating at critical. This sequence representing a single generation in a critical reactor is repeated over and over by an enormous number of neutrons. By slight changes in the reactor configuration during operation (for instance, by addition or removal of a small amount of poisons), the neutron economy can be adjusted to increase or decrease the number of neutrons in the next generation and thus change the reactor operating condition.

A parameter, k, called the *multiplication factor,* is defined as the ratio of the number of neutrons in one generation to the number in the preceding generation. Thus $k = 1$ for a critical reactor. If k is less than or greater than 1, the neutron density is decreasing or building and the reactor is said to be *subcritical* or *supercritical*, respectively.

A reactor is designed to have a maximum value of $k > 1$ so that it may be brought up to a desired power level and maintained there over the duration of the fuel cycle. Once the reactor approaches the desired power level, control actions are taken to adjust the value of k to 1, bringing the reactor to critical and stabilizing its operation. The control action requires the introduction of additional neutron-absorbing material or the reduction of the amount of moderator in the reactor, to reduce the rate of buildup of the neutron density to zero. As fissionable material is depleted and poisons build up, control material is gradually withdrawn from the reactor to maintain critical operation. When the reactor can no longer maintain critical at its design power level with no control material present, it must be refueled.

The presence of a nuclear heat source is the major difference between fossil fuel and nuclear plants. In that connection, there are several important factors that must be considered in the design of a nuclear power plant that are not factors in conventional power plant design and operation. First, the entire amount of fuel needed to operate a nuclear power plant for up to two years is loaded into the plant at one time. The rate at which power is generated must then be maintained by controlling the neutron chain reaction over the wide range of reactor operating and fuel depletion conditions that can arise between refueling operations. This calls for detailed planning and analysis, both before and throughout the operating cycle.

Second, because the products of fission are highly radioactive and their rate of decay cannot be controlled, the heat from radioactive decay of fission products after shutdown amounts to as much as 7% of full power output. Consequently, provision must be made in the thermal design to remove this heat under all credible operating and accident conditions. (The reactor at Three Mile Island melted over an hour after the nuclear chain reaction was terminated, because the operators misinterpreted their instruments and turned off the emergency systems that were removing the decay heat.) The nuclear decay process is not self-limiting and has no maximum temperature, as with chemical reactions. If the heat generated is not removed from the reactor core, the core will melt and be destroyed.

Third, if radioactive materials from the reactor core find their way to the environment, they can be hazardous to nearby life. Although the proper cooling of the core will ensure that these materials will remain contained inside the fuel assemblies, a defense-in-depth approach to safety must be employed to make the possibility of a release extremely remote.

Table 10.1 Nuclear Fuels

Fertile Fuels		*Fuels Fissionable by Thermal Neutrons*
		U-235
U-238	⇒	Pu-239
Th-232	⇒	U-233

10.5. Nuclear Fuels

Uranium-235 is the only material that is both fissionable by thermal neutrons and found in nature in sufficient abundance for power production. Other *fissile fuels* are uranium-233 and plutonium-239 (Table 10.1) which are created from thorium-232 and uranium-238, respectively, by absorption of neutrons. Substances from which fissionable fuels are created, called *fertile fuels*, are transmuted into fissionable fuels in a reactor by extra neutrons not needed to sustain the fission chain reaction. Fertile fuel used in this way is said to have been *converted*. The resulting fissile materials may be processed to make new fuel elements when sufficient quantities have accumulated. Some of the converted material may be consumed directly by fissions during reactor operation.

The composition of uranium ore is about 99.3% U-238 and 0.7% U-235. Because of the ore's small percentage of U-235, it is difficult to design a water-cooled, thermal reactor that uses natural uranium. Therefore the power reactors in the United States and most other parts of the world are thermal reactors that employ uranium enriched to between 2% and 5% U-235. Such reactors use ordinary (light) water for both cooling and moderation and are therefore commonly called *light-water reactors*.

Uranium enrichment is an expensive and difficult process because it involves separation of two isotopes of the same element, which rules out most chemical methods. Thus processes that rely on the small mass difference between U-235 and U-238 are usually used. The *gaseous diffusion process* involves conversion of a uranium compound processed from the ore to gaseous uranium hexafluoride, UF_6. The gaseous UF_6 flows in hundreds of stages of diffusion through porous walls that eventually produce separate UF_6 gas flows containing enriched and depleted uranium. The enriched UF_6 then is processed to UO_2 powder which is sintered into hard ceramic fuel pellets such as those shown in Figure 10.5. The pellets are sealed in long cylindrical metal tubes for use in the reactor.

Newer enrichment processes currently available or under development include: high speed centrifugal separation; which also relies on the uranium isotopic mass difference;

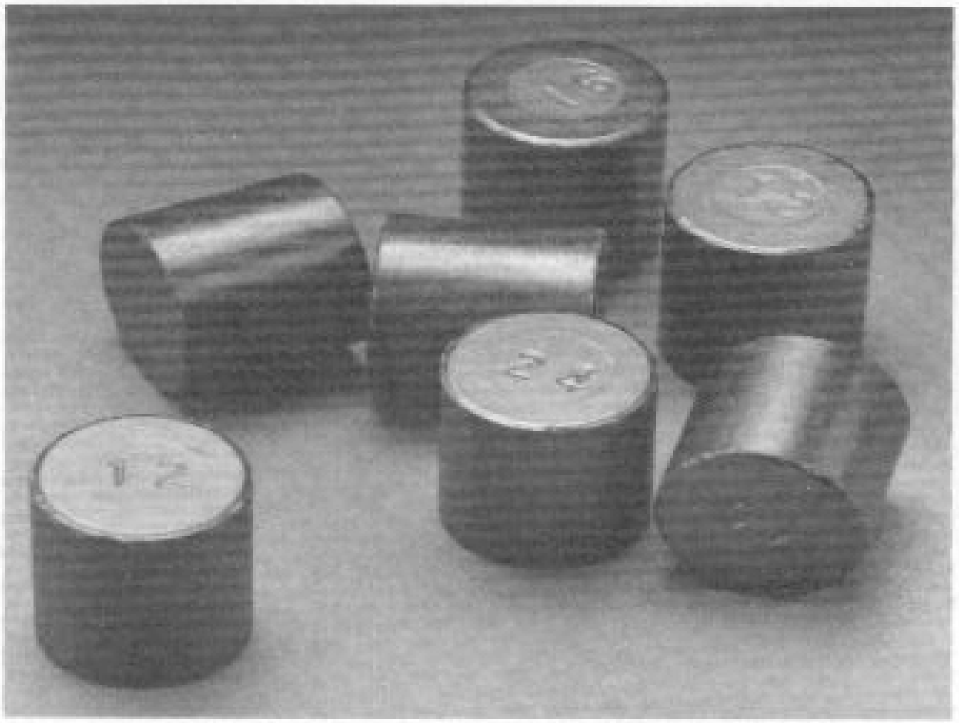


FIGURE 10.5 Nuclear fuel pellets. (Courtesy of G.E. Nuclear Energy.)

a separation process that relies on differences in chemical reactivity between the isotopes; and laser enrichment, which relies on ionization of uranium by an intense light beam with subsequent chemical or physical separation of the ions (ref. 8).

10.6 Light-Water Power Reactors

Power reactors active in the United States today are *light-water reactors*. They are designed so that the core is both moderated and cooled by highly purified water and therefore must use a fuel that fissions with thermal neutrons.

Water has many advantages in thermal reactors. From a neutron point of view, H_2O is an extremely efficient moderator. As we know from its extensive use in conventional power plants, water has excellent heat transfer characteristics, and the technologies of its use in steam power plants are well established.

Water has disadvantages as well. To maintain its excellent moderation and heat transfer capabilities, it must remain a liquid. Thus water reactors are currently limited to producing hot liquid or steam with little superheat. Moreover, boiling temperatures suited to an efficient plant require very high pressures, as in fossil fuel plants. Thus water-cooled reactor cores must be encased in pressure vessels that operate with high temperatures nearby. In addition, they must endure, for the design life of the plant, the

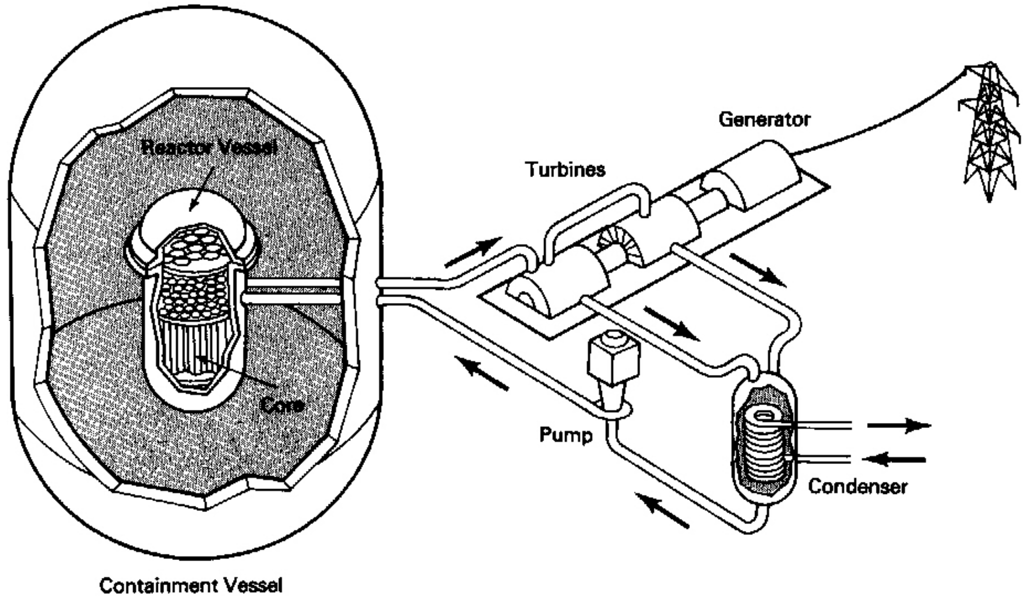


FIGURE 10.6 Boiling Water Reactor. U.S. Congress, Office of Technical Assessment. (See reference 14.)

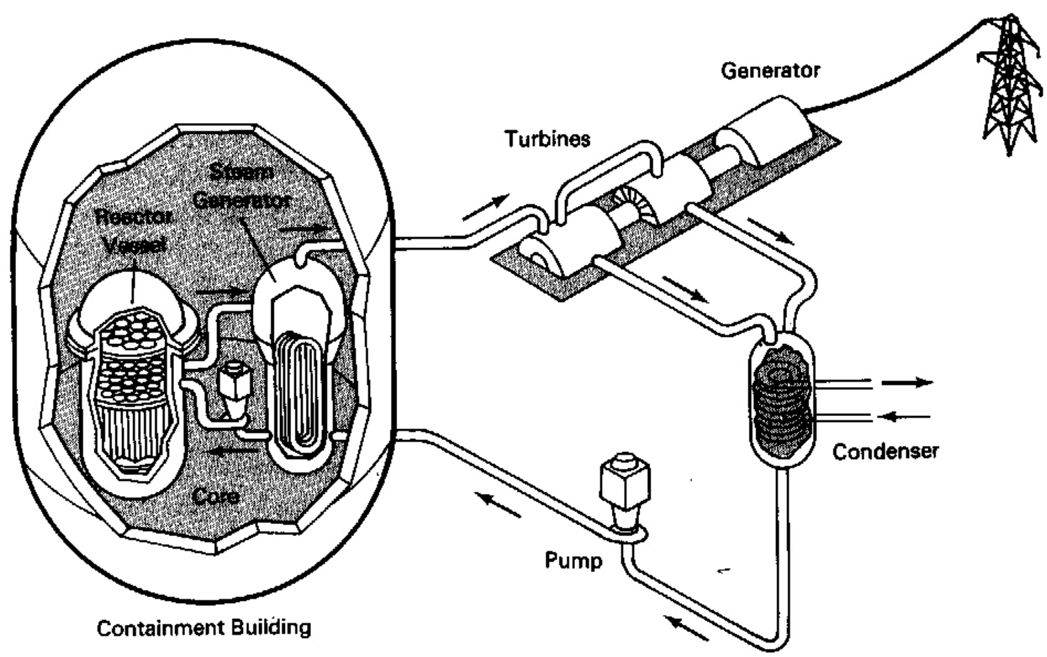


FIGURE 10.7 Pressurized water reactor. U.S. Congress, Office of Technical Assessment. (See reference 14.)

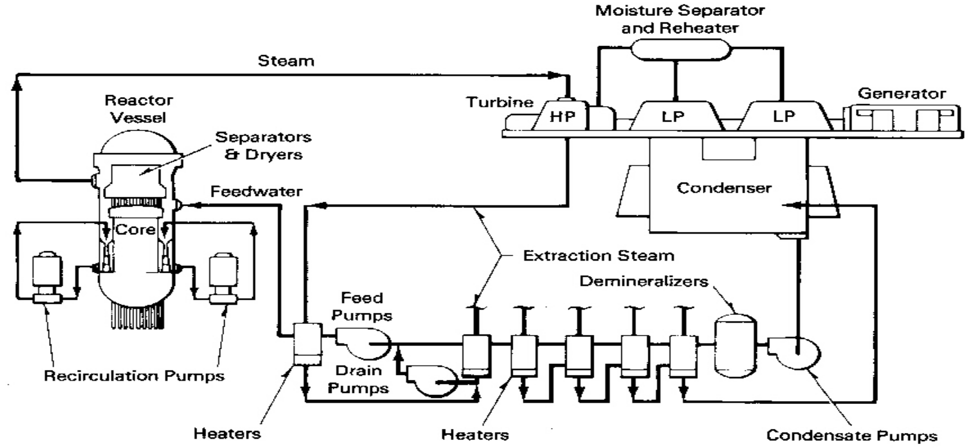


FIGURE 10.8 Flow schematic for a boiling water reactor. (Courtesy of G.E. Nuclear Energy.)

severe environment resulting from the fission reactions. Finally, and perhaps most importantly, should reactor pressure integrity be lost while the reactor is operating, the liquid water will flash to steam, losing much of its heat transfer advantages. All of these factors contribute significantly to the challenges that an engineer faces in the thermal and mechanical design of light-water reactors.

There are two major types of light water reactors (Figures 10.6 and 10.7), which are differentiated primarily by the thermodynamic conditions of the water used to cool uranium fuel elements in the reactor vessel. The *boiling water reactor* (BWR) operates at a pressure that allows boiling of the coolant water adjacent to the fuel elements. The water in the *pressurized water reactor* (PWR) is at about the same temperature as in the BWR but is at a higher pressure, so that the reactor coolant remains a liquid throughout the reactor coolant loop. In addition to their use in utility power reactors, PWRs are used in American nuclear submarines.

Boiling Water Reactors

A schematic of the layout of the General Electric BWR/6 system is shown in Figure 10.8. There the turbines, condenser, pumps, and feedwater heaters studied in Chapter 2 appear in a familiar configuration. Water boils inside the reactor core, producing slightly radioactive steam that passes directly to the steam turbines. The radioactivity in the steam, however, has a half-life of only a few seconds. The carryover of radioactivity to the turbine-feedwater system is virtually nonexistent, and experience has shown that components outside the reactor vessel (turbine, condensate pump, etc.) may be serviced essentially as in a fossil-fueled system. Some other reactor designs, such as the pressurized water reactor to be considered later, have an additional separate water loop, as seen in Figure 10.7, that isolates the turbine steam loop from the reactor coolant to provide further assurance that the turbine-feedwater system components remain free of radioactivity.

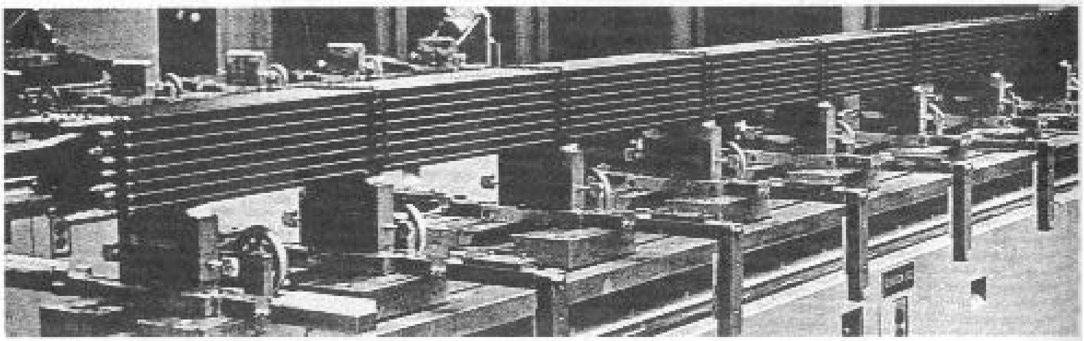


FIGURE 10.9 Boiling water reactor fuel bundle. (Courtesy of G.E. Nuclear Energy.)

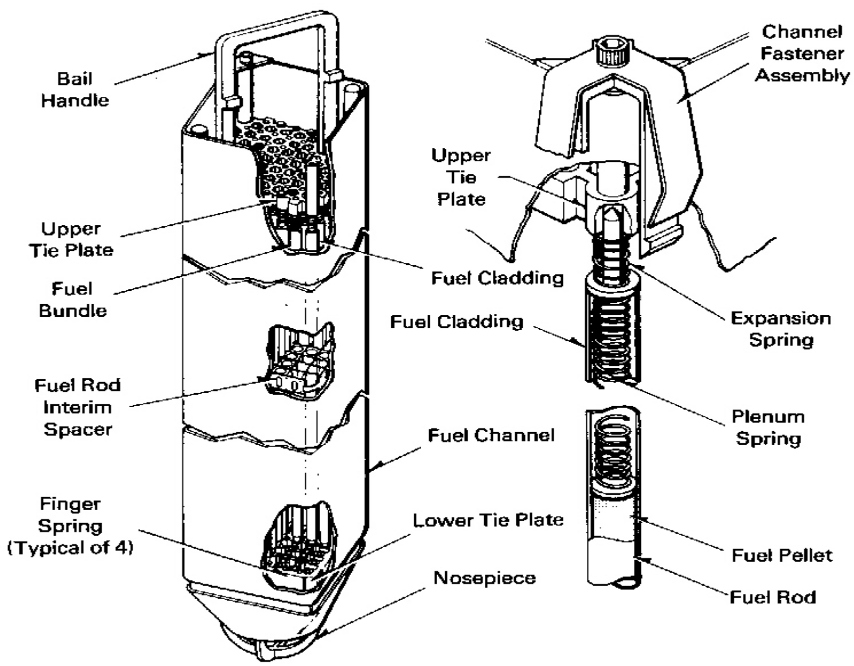


FIGURE 10.10 Boiling water reactor fuel assembly. (Courtesy of G.E. Nuclear Energy.)

Fuel Assemblies

Uranium appears in most boiling water reactors in the form of sintered cylindrical pellets of uranium dioxide (UO_2) (Figure 10.5), about 0.4 inches in diameter and about 0.4 inches long. These pellets are stacked inside of long sealed zirconium alloy (zircaloy) tubes called *fuel rods*. Fuel rods, in turn, are mounted in an eight-by-eight array in a *fuel bundle*, as seen in Figure 10.9. The fuel bundle and the *fuel channel* that surrounds the fuel rods comprise a *fuel assembly,* as shown in Figure 10.10.

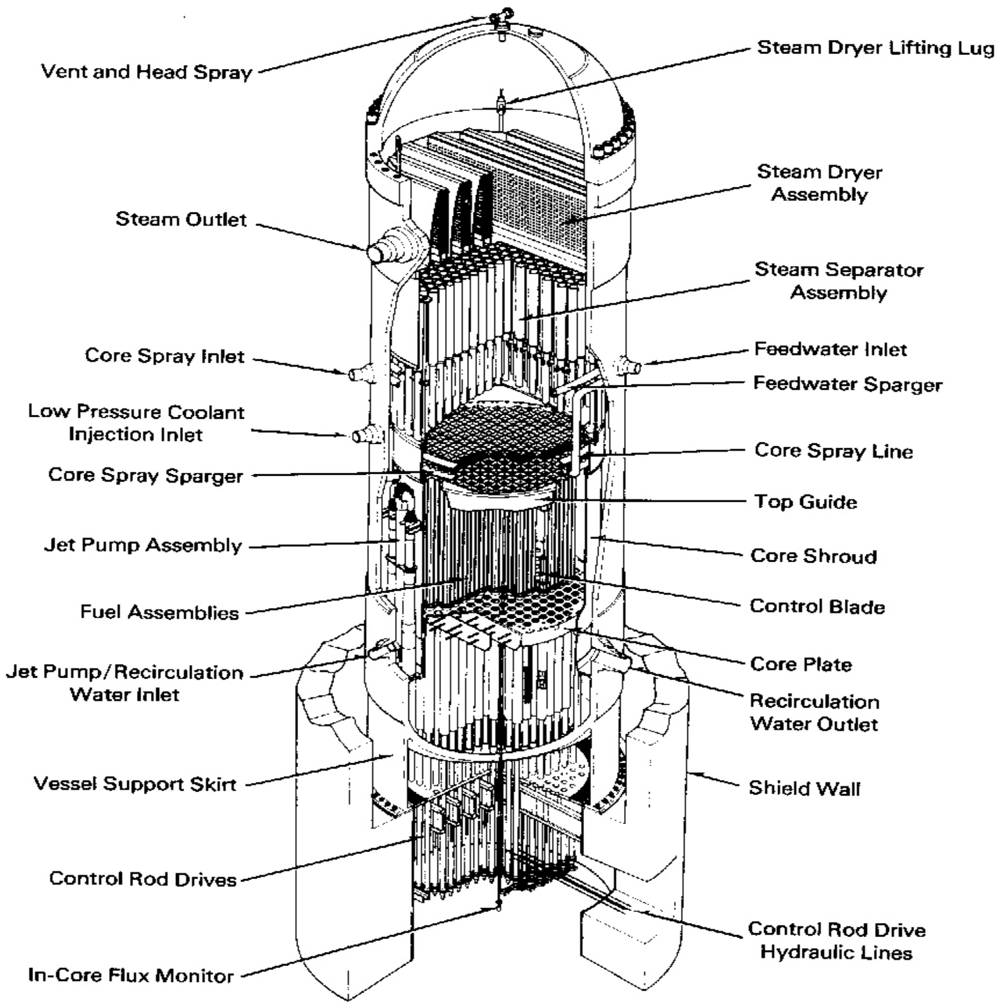


FIGURE 10.11 Cutaway of a boiling water reactor. (Courtesy of G.E. Nuclear Energy.)

The lower-tie-plate nose piece seen there, together with the fuel channel, directs the coolant water flow over the fuel rods inside of the fuel assembly.

The fuel assemblies, mounted vertically in the core, are designed to minimize operating stresses on the fuel rods. For example, Figure 10.10 shows that the fuel rods are spring loaded so they are free to expand in the axial direction in response to changes in reactor operating temperatures.

The Reactor Assembly

The core of a BWR/ 6 1220-MW_e reactor (MW_e stands for electrical generator power output, as opposed to MW_t for reactor thermal output) has 732 fuel assemblies and 177 control rod assemblies in an approximately 16-ft-diameter circular array, as shown in Figure 10.11. Cooling water receives heat from the fuel rods by forced-convection and

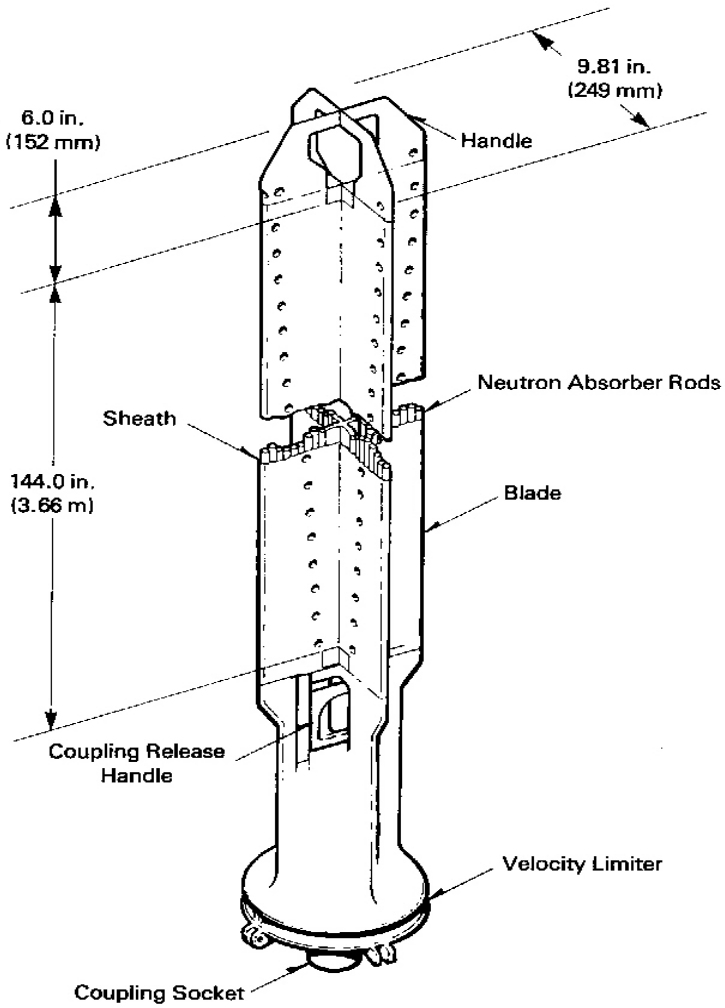


FIGURE 10.12 Cruciform control rod assembly. (Courtesy of G.E. Nuclear Energy.)

two-phase nucleate boiling as it flows upward through the fuel assemblies. As the water cools the fuel assemblies, it also thermalizes the fast neutrons that diffuse through the core.

Water that bypasses the fuel assemblies and flows upward on the outside of the fuel channels is confined by the cylindrical *core shroud,* as seen in Figure 10.11. This flow cools the channels and the neutron absorber *control rods*. The control rods are mounted in cruciform assemblies, as seen in Figure 10.12. The control rod assemblies fit vertically in a *fuel module* between 4 fuel assemblies in the core, as diagrammed in Figure 10.13. The control rod assemblies move vertically between the fuel bundles to change the effective multiplication factor to compensate for changes in reactor operating conditions due to buildup of poisons over time. The stainless-steel-clad control rods contain boron carbide (B_4C), which absorbs neutrons and hence tends to

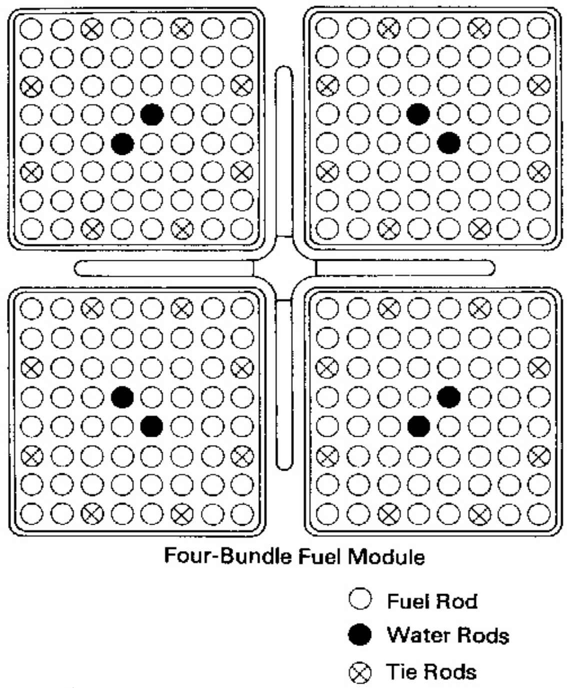


FIGURE 10.13 Cross-section of a boiling water reactor fuel module. (Courtesy of G.E. Nuclear Energy.)

terminate fission processes when in place in the core. The control rod assemblies are moved up and down through the core to change the rate of absorption of neutrons when significant changes in power level or adjustments to account for fuel burnup are required, or in the event of an emergency shutdown.

The positions of the control rods are adjusted by hydraulic drives located below the core. The rods are fully inserted in the core during a shut down making the multiplication factor less than 1. Most of the control rods are fully out of the core during critical operation. When all of the rods are out of the reactor the multiplication factor slightly exceeds 1, which allows the neutron density and power level to increase. The bottom-entry fuel rod drives in the GE BWR are an unusual feature in reactors. Their location below the reactor simplifies the refueling process which is carried out from above in most reactor designs.

The quality of the steam leaving the top of the core is approximately 11%, indicating that most of the water is still liquid and must be recirculated through the core for additional heating. The liquid water leaving the top of the core and the steam separators flows downward outside the *core shroud*, driven by the jet pumps located between the shroud and the reactor vessel wall, as shown in Figure 10.14. The jet pumps, in turn, are driven by recirculation pumps located outside the reactor vessel. The jet pumps induce the downward flow outside the shroud by momentum transfer to the slower-moving liquid.

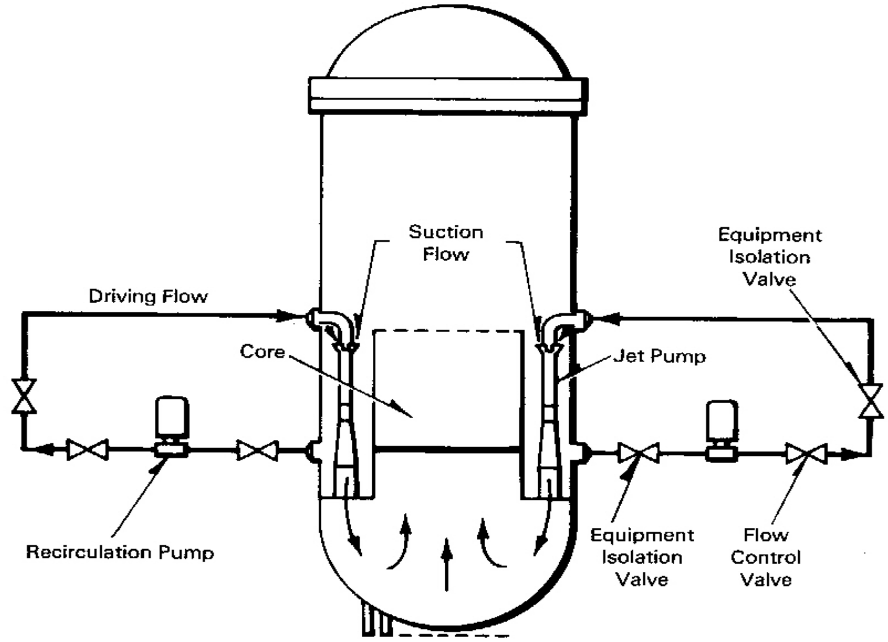


FIGURE 10.14 Boiling water reactor jet pump and recirculation pump loops. (Courtesy of G.E. Nuclear Energy.)

Vapor bubbling up through the fuel assemblies leaves the core and passes upward with the liquid. The flow is turned by stationary vanes in the *steam separator,* where the higher angular momentum of the liquid separates it from the vapor. The separated liquid flows to the outside of the reactor vessel where it is recirculated outside the core shroud.

The steam passing through the separator is further dried in the steam dryer assembly before it leaves the reactor vessel as a slightly superheated or saturated vapor. The turbines are specially designed to operate with saturated vapor at the throttle and small amounts of liquid within. Liquid is separated from the wet steam leaving the HP turbine. Steam tapped from the HP-turbine-throttle steam line is used to reheat the HP exit steam before its entry to the LP turbines. The low HP-turbine-throttle conditions of 550°F and 1040 psia lead to a plant thermal efficiency of about 32%.

In the boiling water reactor, control is primarily achieved by adjustment of the rate of recirculation through the reactor by the recirculation and jet pumps shown in Figures 10.11 and 10.14. Change in the rate of water recirculating through the core changes both the onset of boiling and the volume fraction of steam in the cooling channels, and thus the amount of moderator in the core at a given time. This allows significant adjustment of reactor power output without control rod movement. For example, increasing the recirculation and jet pump speeds sweeps bubbles away faster, increasing moderation and raising the power level until the additional boiling restores the proper void fraction for critical operation at a higher power level.

FIGURE 10.15 Oconee Nuclear Station of the Duke Power Co. (Courtesy of Babcock and Wilcox Co.)

As the coolant passes upward through the fuel assemblies, heat from the fuel rods produces vapor bubbles via *nucleate-boiling heat transfer*. Nucleate boiling is characterized by the local formation of bubbles of vapor that break away from the fuel rod surface, causing vigorous, agitated, fluid motion with resulting high heat transfer rates. This may be contrasted with *film boiling*, in which a stable vapor layer covers the tube surface, resulting in low heat transfer rates. Film boiling occurs at high surface-to-bulk fluid temperature differences. It is crucial that a *departure from nucleate boiling*, DNB, be avoided, because reduced heat transfer coefficients and cooling rates produce drastic increases in tube temperatures, leading to fuel melting or zircaloy fuel rod burnout. Thus the ratio of maximum heat flux to the critical heat flux for DNB is a major thermal design parameter for the water reactor. The reactor fuel rod heat-generation rate and flow-channel convective cooling are designed to maintain tube-to-fluid heat fluxes well below the unstable transition range between nucleate and film boiling. The maximum UO_2 fuel temperature based on the maximum design fuel rod heat-generation rate of 13.4 kW/ft is approximately 3400°F, whereas the fuel melting temperature is about 5100°F.

The boiling water reactor, like other reactors, has numerous active and passive safety systems. A thick pressure vessel, for instance, surrounds the reactor core. An even thicker concrete containment structure surrounds the pressure vessel to confine anything that may escape from it. An *emergency core cooling system* (ECCS) senses overheating of the core and supplies a flood of water to take away the heat generated by the fuel elements. These and other safety systems clearly reduce the danger of accidents but also increase the cost and complexity of plant operation.

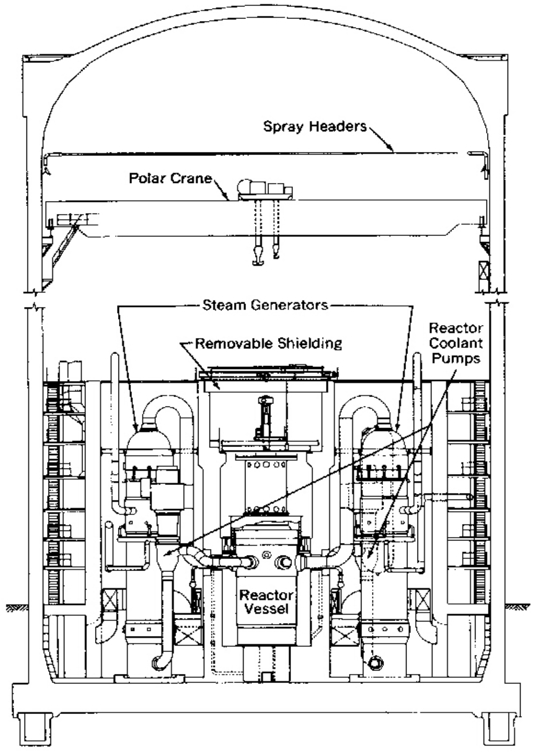


FIGURE 10.16 Sectional view of a reactor building at Oconee Nuclear Station. (Courtesy of Babock and Wilcox Co.)

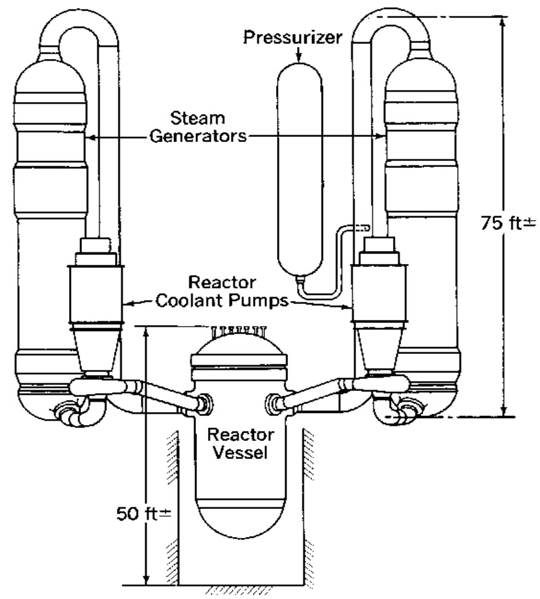


FIGURE 10.17 Reactor, pumps, pressurizer, and steam generators in a pressurized water reactor. (Courtesy of Babcock and Wilcox Co.)

Pressurized Water Reactors

The *pressurized water reactor*, PWR, is currently the predominant reactor type in the world. It is a light-water reactor that uses slightly enriched U-235 as fuel. Figure 10.15 shows the three reactor containment buildings of the Oconee PWR Nuclear Station.

A major difference between BWRs and PWRs is that the pressure of the PWR coolant is above the saturation pressure (it is subcooled liquid) through the entire cooling loop so that there is no possibility of bulk boiling in the core. As shown in Figure 10.7, separate steam generators receive heat from the reactor liquid cooling loop, thus preventing radioactive material from entering the turbine power loop. Another difference is that control rods are at the top of the PWR and can drop by gravity into the reactor when necessary. Figure 10.16 shows a sectional view of the reactor building of the Oconee plant. The stairs and landings give some idea of the size of the equipment within the containment. Figure 10.17 gives a less cluttered view of the major components. Table 10.2 shows that, for PWRs, the turbine loop is at a lower pressure than the liquid in the reactor loop and therefore produces an outflow of steam to the turbine throttle with about 50 Fahrenheit degrees of superheat.

TABLE 10.2 (Design Parameters—Nuclear Steam Supply Systems)

	Oconee Unit 1	1300-MW Unit*
Reactor Hydraulic and Thermal Design		
Rated heat output (core), Mw	2,568	3,760
Rated heat output (core), million Btu/hr	8,765	12,833
Design overpower, %	114	112
System pressure (nominal), psi	2,185	2,250
Power distribution		
Maximum/average power ratio, radial × local	1.78	1.55
Maximum/average power ratio, axial	1.70	1.67
Overall power ratio	3.03	2.65
Power generated in fuel and cladding, %	97.3	97.3
DNB ratio at rated conditions	2.0	1.81
Minimum DNB ratio at design overpower	1.55	1.40
Coolant flow		
Total flow rate, million lb/hr	131.3	158.2
Effective flow rate for heat transfer, million lb/hr	124.2	148.5
Effective flow area for heat transfer, ft^2	49.19	56.6
Average velocity along fuel rods, ft/sec	15.73	16.85
Coolant temperature, °F		
Nominal inlet	554	568.6
Average rise in vessel	50	56.9
Average in vessel	579	597
Average film coefficient, Btu/ft^2, hr, °F	5,000	5,000
Average film temperature difference, °F	31	60
Heat transfer at 100% power		
Total heat transfer surface area, ft^2	49,734	63,991
Reactor Hydraulic and Thermal Design (continued)		
Average heat flux, Btu/ft^2, hr	171,470	195,100
Maximum heat flux, Btu/ft^2, hr	534,440	517,640
Average thermal output, kW/ft	5,656	5,670
Maximum thermal output, kW/ft	17.63	15.04
Maximum clad surface temperature at nominal pressure, °F	654	657
Fuel central temperature, °F		
Maximum at 100% power at hot spot	4,250	4,040
Maximum at design overpower	4,650	4,400
Maximum thermal output, kW/ft at design overpower	20.1	16.85
Core Mechanical Design		
Fuel assemblies		
Number	177	205
Rod pitch, in.	0.568	0.503
Overall dimension, in. (side of square)	8.536	8.632
Total weight, lb	274,350	313,855
Number of spacer grids per assembly	8	8
Fuel assembly pitch spacing, in.	8.587	8.587
Fuel rods		
Number	36,816	54,120
Outside diameter, in.	0.430	0.379
Diametral gap, in.	0.007	0.008
Clad thickness, in. (zircaloy-4)	0.0265	0.0235
Active fuel length, in.	144	143

(continued on next page)

TABLE 10.2 (continued)

	Oconee Unit 1	1300-MW Unit*
Core Mechanical Design (continued)		
Fuel pellets (UO_2 sintered)		
Density, % of theoretical		
First core	93.5	94.0
Subsequent cores	92.5	94.0
Diameter, in.	0.370	0.324
Length, in.	0.7	0.375
Control rod assemblies†		
Neutron absorber	5% Cd-15% In-80% Ag	Ag-In-Cd or boron carbide
Length of poison section, in.	134	134
Cladding material	304SS	304SS
Clad thickness, in.	0.021	0.019
Number of assemblies	61	64
Number of control rods per assembly	16	24
Axial-power-shaping rod assemblies		
Neutron absorber	5% Cd-15% In-80% Ag	Ag-In-Cd or boron carbide
Length of poison section, in.	36	36
Cladding material (poison section)	304SS	304SS
Clad thickness, in.	0.021	0.019
Number of assemblies	8	8
Number of control rods per assembly	16	24
Orifice rod assemblies		
Rod material	304SS	304SS
Number of orifice rods per assembly	16	24
Core structure		
Core barrel ID/OD, in.	141/145	156/161
Thermal shield, ID/OD, in.	147/151	
Nuclear Design		
Structural characteristics		
Fuel weight (as UO_2), lb	207,486	233,850
Nuclear Design (continued)		
Clad weight (active zone), lb	42,200	48,175
Core diameter, in. (equivalent)	128.9	138.7
Core height, in. (active fuel)	144	143
Reflector thickness and composition		
Top (water plus steel), in.	12	12
Bottom (water plus steel), in.	12	12
Side (water plus steel), in.	18	21.5
Metal/Water (unit cell—volume basis)	0.82	0.83
Fuel rods per fuel assembly	208	264
Performance characteristics		
Loading technique	3 region	modified checkboard
Fuel discharge burnup, Mwd/ton of uranium		
First cycle average	9,600	16,864
Succeeding cycle average	9,700	11,243
Fuel enrichments, wt % U^{235}		
Core average, first cycle	2.10	2.80
First reload	2.98	3.28
Core average, equilibrium cycle	3.06	3.18
Control characteristics		
Effective multiplication (beginning of life)		
Cold, no power, clean	1.248	1.333
Hot, no power, clean	1.198	1.286
Hot, rated power, Xe and Sm equilibrium	1.132	1.192
Control rod worth ($\Delta k/k$), %	10.9	8.1‡

TABLE 10.2 (continued)

	Oconee Unit 1	1300-MW Unit*
Nuclear Design (continued)		
Boron concentrations		
To shut reactor down (k_{eff} = 0.99) with all rods inserted (clean), cold/hot ppm	864/587	1178/910
Boron worth (hot), % $(\Delta k/k)$/ppm	1/85	1/105
Boron worth (cold), % $(\Delta k/k)$/ppm	1/64	1/79
Kinetic characteristics (range during life cycle)		
Moderator temperature coefficient, $(\Delta k/k)$/F	$+0.5\times10^{-4}$ to -3.0×10^{-4}	$+0.1\times10^{-4}$ to -3.0×10^{-4}
Moderator pressure coefficient, $(\Delta k/k)$/psi	-5.0×10^{-7} to $+3.0\times10^{-6}$	-0.5×10^{-7} to $+3.0\times10^{-6}$
Moderator void coefficient, $(\Delta k/k)$% void	$+4.0\times10^{-4}$ to -1.6×10^{-3}	$+3.0\times10^{-4}$ to -3.0×10^{-3}
Doppler coefficient, $(\Delta k/k)$/F	-1.1×10^{-5} to -1.7×10^{-5}	-1.1×10^{-5} to -1.7×10^{-5}
Reactor Coolant System		
System heat output, MW	2,584	3,780
System heat output, million Btu/hr	8,819	12,901
Operating pressure, psi	2,185	2,250
Reactor inlet temperature, °F	554	568.6
Reactor outlet temperature, °F	604	625.5
Number of loops	2	2
Design pressure, psi	2,500	2,500
Design temperature, °F	650	670
Hydrostatic test pressure (cold), psi	3,125	3,125
Coolant volume, including pressurizer, ft^3	11,478	13,200
Total reactor flow, gpm	352,000	432,800

	Oconee Unit 1	1300-MW Unit*
Reactor Coolant System Code Requirements		
Reactor vessel and closure head	ASME III	ASME III
Steam generator and pressurizer	ASME III	ASME III
Reactor coolant piping	ANSI B31.7	ASME III
Reactor coolant pump casing	ASME III§	ASME III
Reactor Vessel		
Base material	Low-alloy steel	
Cladding material	SS	SS and Inconel
Design pressure, psi	2,500	2,500
Design temperature, °F	650	670
Operating pressure, psi	2,185	2,250
Inside diameter of shell, in.	171	182
Straight shell thickness, in.	8 7/16	9 1/8
Minimum clad thickness, in.	1/8	1/8
Outside diameter across nozzles, ft-in.	20-9	23-4
Overall height of vessel and closure head (over control rod drive and instrument nozzles), ft-in	40-8 3/4	42-2
Steam Generators		
Number of units	2	2
Type	Once-through	
Materials		
Tubes	Inconel	
Shell	Carbon-steel	Low-alloy steel
Hemispherical heads	Low-alloy steel, SS clad	
Tubesheets	Low-alloy steel, Inconel clad on reactorcoolant side‖	

TABLE 10.2 (continued)

	Oconee Unit 1	1300-MW Unit*
Steam Generators (continued)		
Tube side design pressure, psi	2,500	2,500
Tube side design temperature, °F	650	670
Tube side design flow, million lb/hr	65.66	79.1
Shell side design pressure, psi	1,050	1,235
Shell side design temperature, °F	600	631
Operating pressure, tube side, nominal, psi	2,185	2,250
Operating pressure, shell side, nominal, psi	910	1,060
Superheat at outlet at rated load, °F	35	35
Hydrostatic test pressure (tube side cold), psi	3,125	3,125
Shell minimum thickness, in.	4 3/16	5 3/4
Maximum outside diameter of straight shell, ft-in.	12-7 1/4	12-6 1/2
Overall height (including supports), ft-in.	73-2 1/2	77-6 1/4
Pressurizer		
Material, shell and heads	Carbon-steel, SS clad	Low-alloy steel, SS clad
Design pressure, psi	2,500	2,500
Design temperature, °F	670	670
Steam volume, ft^3	700	1,050
Water volume, ft^3	800	1,200
Electric heater capacity, kW	1,638	1,745
Shell minimum thickness, in.	6 3/16	5 15/16
Shell outside diameter, ft-in.	8-0 3/8	9-11 7/8
Overall height, ft-in.	44-11 3/4	41-4 1/2

	Oconee Unit 1	1300-MW Unit*
Reactor Coolant Pumps		
Number of units	4	4
Type	Vertical, single-stage	
Design pressure, psi	2,500	2,500
Design temperature, °F	650	670
Operating pressure, nominal, psi	2,185	2,250
Suction temperature, °F	554	568.6
Design capacity, gpm	88,000	108,200
Total developed head, ft	350	370
Hydrostatic test pressure (cold), psi	4,100	3,340
Motor type (single speed)	a-c induction	Squirrel-cage induction
Motor rating (nameplate), hp	9,000	12,500
Reactor Coolant Piping		
Material	Carbon steel, SS clad	
Hot leg (ID), in.	36	38
Cold leg (ID), in.	28	28
Engineered Safeguards		
Safety injection system		
Number of high-pressure injection pumps	3	3
Number of low-pressure injection pumps	2**	2
Reactor building coolers		
Number of units	3	3
Rated capacity, each, million Btu/hr	80	140
Core flooding system		
Number of tanks	2	2
Total volume, each ft^3	1,410	1,800
Reactor building spray		
Number of pumps	2	2

* Approximate net electric power.

† The 1300-Mw unit has also 116 burnable poison rod assemblies, having 24 rods per assembly, each rod containing 126-in.-length of $Al_2O_3 \cdot B$ contained in zircaloy-4 cladding tubes 0.032-in. thick.

‡ Burnable-poison rods are worth an additional 5.1%.

§Not code stamped.

‖1300-MW-unit lower tubesheet is Inconel-clad on both sides.

** Plus one installed spare pump.

Source: Courtesy of Babcock and Wilcox Co.

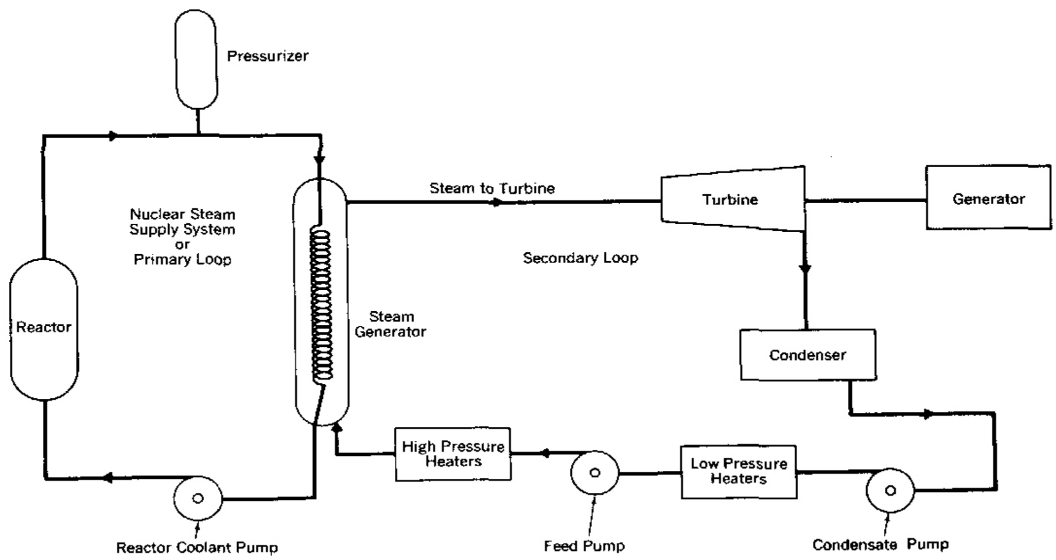


FIGURE 10.18 Diagram of a pressurized water reactor system. (Courtesy of Babcock and Wilcox Co.)

We have seen that, in contrast to the BWR, which circulates the reactor coolant through the turbine, the PWR uses two loops: a primary loop that cools the reactor core, and a secondary loop, heated by the primary loop, that provides steam for the turbines, as diagrammed in Figure 10.18. The primary loop is contained completely within the reactor containment building (Figure 10.16) and is designed so that the water that cools the core is completely isolated from the environment. The secondary loop executes a steam turbine cycle similar to those of conventionally fueled power plants, with the exceptions that the steam is generated by heat transported from the reactor and that the maximum turbine inlet temperature and superheat are limited by the maximum temperature in the reactor core.

The Babcock & Wilcox steam generator is a counterflow shell-and-tube heat exchanger, as seen in Figure 10.19. The primary water enters the top of the unit and flows downward through thousands of small-diameter tubes to provide a large heat transfer surface. Feedwater is piped into the bottom of the shell side of the steam generator and is first heated, then boiled by heat from the hot tubes containing the primary water flow. As the steam rises, it encounters the hotter portions of the primary tubing, reaching 50 Fahrenheit degrees of superheat at the top of the steam generator.

An important thermal design parameter of the steam generator is its size. A larger unit increases the heat transfer area from the primary loop, but it also increases the capital cost of the plant, both in the cost of manufacturing the generator itself and in the larger size required of the primary containment. This must be balanced with the cost savings, lower heat exchanger effectiveness and consequent lower thermal efficiency achieved, if the heat transfer area is made smaller.

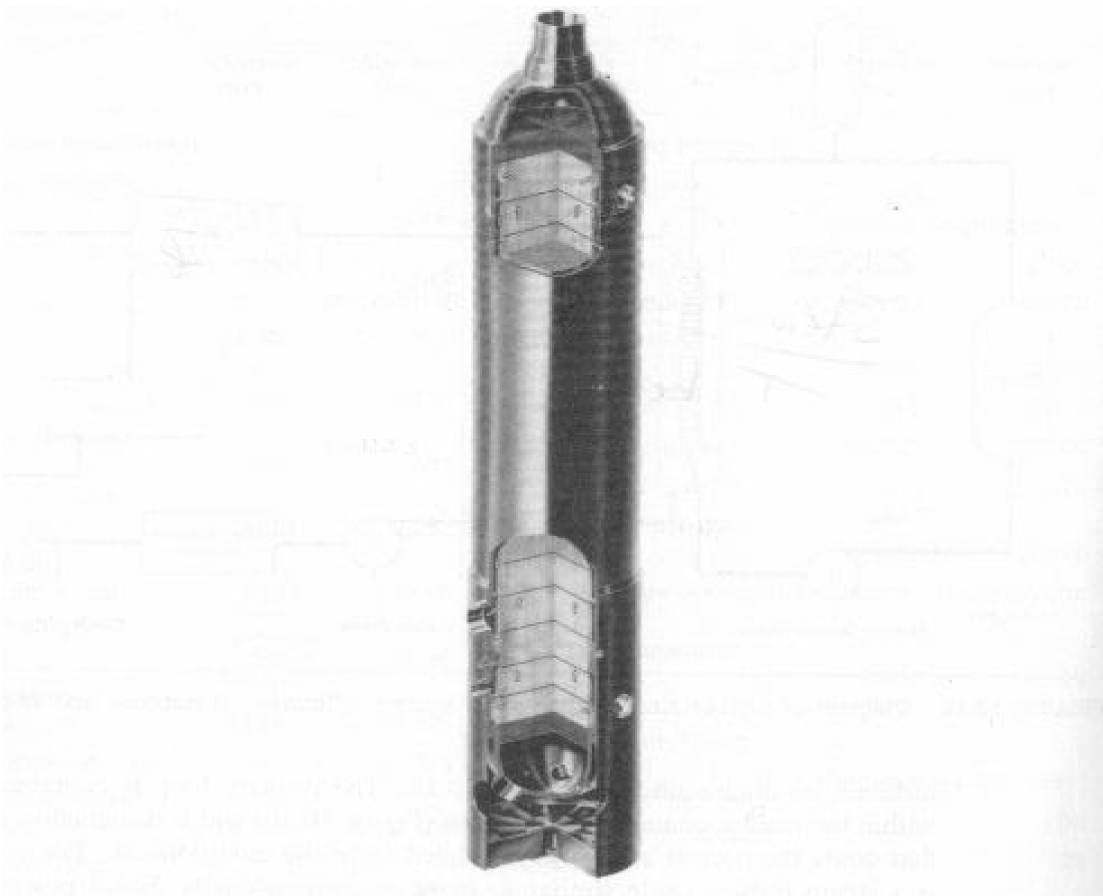


FIGURE 10.19 Pressurized water reactor steam generator. (Courtesy of Babcock and Wilcox Co.)

The pressurizer (Figure 10.17) controls the pressure in the primary loop and serves as a surge tank to accommodate the reactor coolant water's expansion and contraction with temperature changes. It is a large tank designed to contain saturated water and steam, each occupying about one-half of the tank volume. As shown in Figures 10.17 and 10.18, the pressurizer is connected to the primary loop at the outlet of the reactor through a single pipe. The temperature of the pressurizer contents are controlled with internal electric heaters and water sprays. The resulting saturation temperature establishes the operating pressure of the reactor.

An important part of the PWR thermal design is sizing the pressurizer tank to permit the primary system to respond to all possible transients without bursting the coolant pipes. Recalling that water is an almost incompressible fluid that expands when heated, if the reactor power rises and heats the primary water to a higher temperature, the expanding water will flow into the pressurizer, compressing the steam bubble. Pressure sensors will detect the increased pressure and open spray valves at the top of the pressurizer to condense some of the steam, thus restoring a lower operating pressure. Conversely, if the primary loop temperature declines, heaters in the bottom

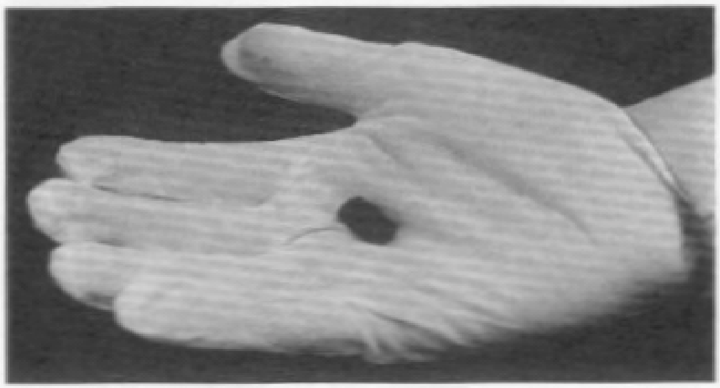


FIGURE 10.20 Pressurized water reactor fuel pellet. (Courtesy of Babcock and Wilcox Co.)

of the pressurizer will turn on, creating more steam to fill the volume formerly occupied by the contracting water and thus increasing the reactor temperature and pressure.

Although there is no bulk boiling in a PWR core, heat transfer to the coolant is by *subcooled nucleate boiling*. In subcooled nucleate boiling, steam bubbles are formed on the surface of the cladding. As the bubbles expand they become detached from the tubes and immediately collapse as they are swept into the coolant channel. The lateral motion in the coolant achieved by this action is an extremely effective mechanism to transport the energy from the cladding into the coolant stream. With subcooled-nucleate-boiling heat transfer, the cladding surface temperature will stay within 10°F of the water's saturation temperature while providing the very high rates of heat transfer needed to remove the fission energy from the fuel rods.

An extremely important design limitation is the *critical heat flux* at which steam bubbles form and grow so fast that they coalesce to form a vapor film over the clad. At this point the heat transfer undergoes a departure from nucleate boiling to *film boiling*, and the clad no longer touches liquid water. Since metal to vapor heat transfer is much more inefficient than nucleate boiling, the fuel rod temperature rises in order to transfer the accumulating heat that is created by fissions. Unfortunately, film boiling heat transfer coefficients are so low that the high surface temperatures needed to achieve a steady state are sufficient to melt the clad and severely damage the fuel rods. For this reason the Nuclear Regulatory Commission has established safety factors for the ratio between the critical heat flux and the maximum heat flux expected in a reactor under its most severe overpower transient conditions. In its application for an operating license, every nuclear power plant must be able to demonstrate through engineering analysis that it will maintain the required safety factor under all credible overpower and undercooling transients.

The Babcock and Wilcox PWR uses U-235 enriched to about 3% in the form of UO_2 fuel pellets (Figure 10.20) encased in a zircaloy-clad tube similar to the fuel rods described for the BWR. Table 10.2 shows that a 1300-MW_e plant has 205 fuel assemblies with 54,120 fuel rods. Figure 10.21 shows fuel assemblies for an electric

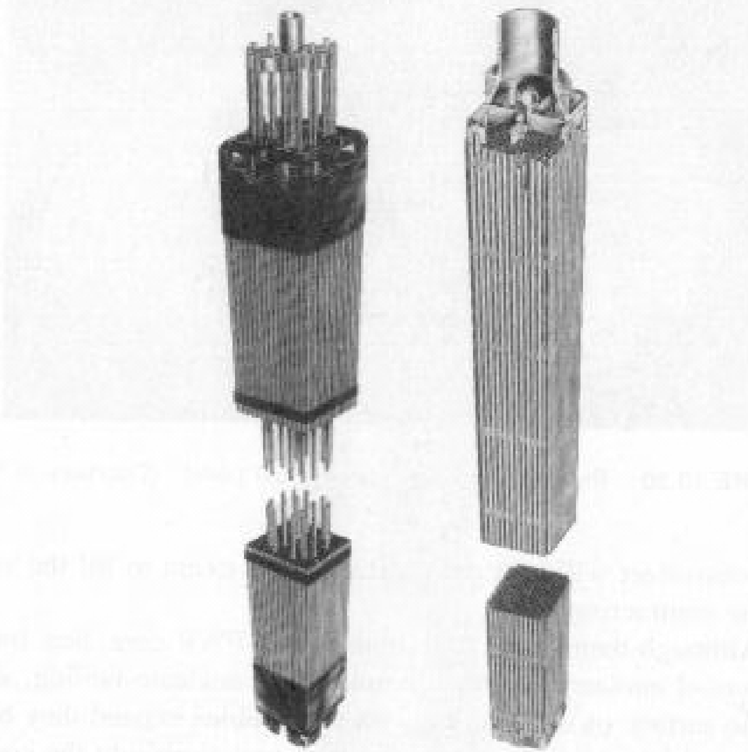


FIGURE 10.21 Fuel assemblies of pressurized water reactors. (Courtesy of Babcock and Wilcox Co.)

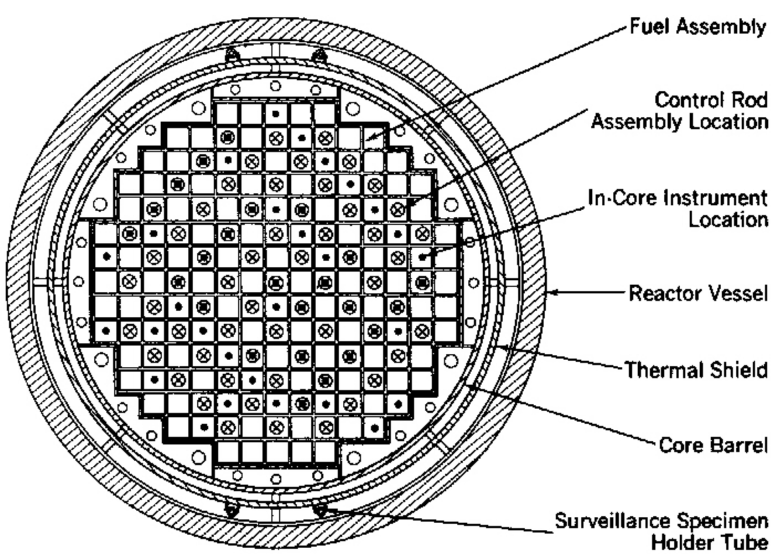


FIGURE 10.22 Cross-section of a pressurized water reactor vessel and internals. (Courtesy of Babcock and Wilcox Co.)

utility PWR and for the Nuclear Ship Savannah. Each fuel assembly for electric power generation is fitted for the inclusion of sixteen control rods, as seen attached to an activating spider in the left fuel assembly photo. Positioning of the fuel assemblies in the core and those assemblies containing control rods are shown in Figure 10.22. There are sixty-nine control rod assemblies in Oconee Unit 1, of which sixty-one are for control of power level; the remaining eight contain poisons in the lower part of the rods for shaping of the reactor power distribution. The control rod neutron-absorbing material, silver-indium-cadmium, is encased in stainless steel.

10.7 The CANDU Reactor

The CANadian Deuterium Uranium, CANDU, reactor is a reactor of unique design that utilizes natural uranium as a fuel and heavy water as a moderator and coolant. These reactors produce a substantial saving due to the absence of fuel enrichment costs, but a large chemical plant is required to supply the quantities of heavy water required. The Pickering station near Toronto, an Ontario Hydro plant shown in Figure 10.23, uses eight CANDU reactors to generate 4800 MW_e and has been generating power since 1971.

One of the important and unique features of the CANDU reactor is that, whereas light-water reactors are shut down for refueling annually, CANDU reactors are refueled daily. The pressurized heavy-water-cooled fuel bundles are horizon-tally oriented in individual fuel channels inside the unpressurized "*calandria,*" as diagrammed in Figure 10.24. Each bundle may be individually accessed, rearranged in

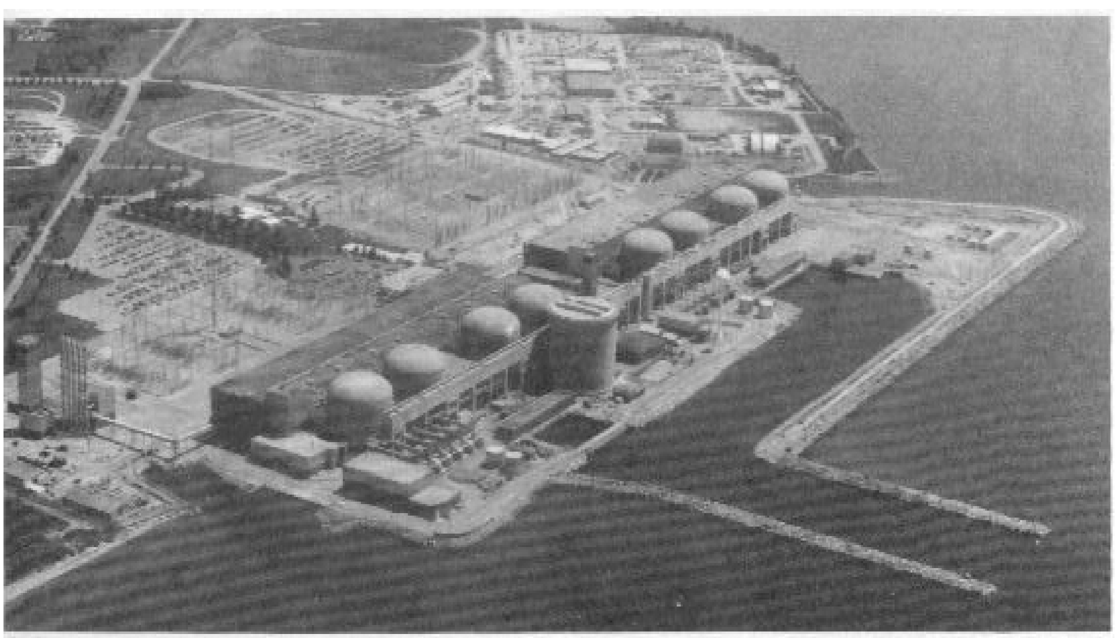


FIGURE 10.23 Pickering Nuclear Generating Station has eight CANDU units. (Courtesy of Ontario Hydro, Toronto, Canada.)

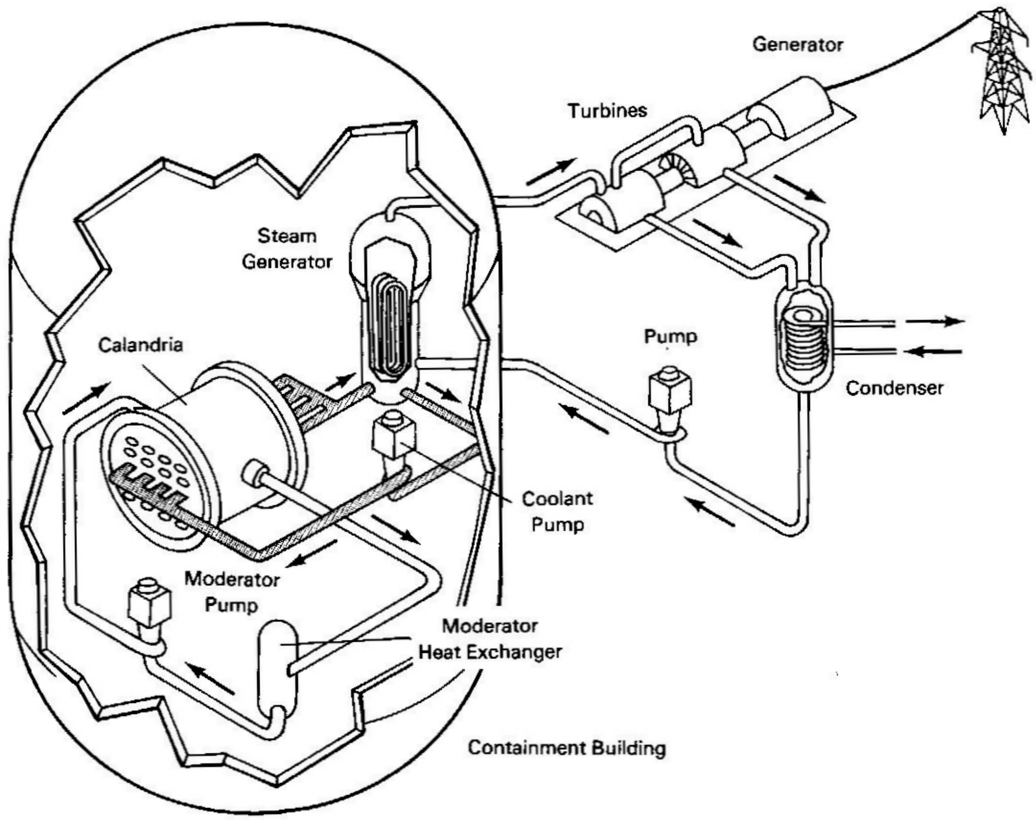


FIGURE 10.24 Schematic of a CANDU nuclear power plant. (Courtesy of the U.S. Congress Office of Technical Assessment.)

the calandria, or replaced using special fuel handling equipment while the reactor is operating. Heavy water at atmospheric pressure in the calandria surrounds the fuel channels and moderates the reactor. Thus, in an emergency, the reactor can be shut down by draining the calandria to remove the moderator, thereby depriving the fuel of thermal neutrons. The pressurized heavy-water loop draws hot coolant from the fuel channels through headers to supply heat to steam generators as in a PWR. A light-water loop passing through the steam generator in turn supplies steam to the turbine-feedwater loop. A turbine hall at the Pickering Station is shown in Figure 10.25.

The largest cylindrical structure seen in Figure 10.23 is a vacuum building that connects with each of the reactor buildings. In the event of an emergency, escaping steam and radioactive materials would be drawn by vacuum into the structure. A cold-water spray would condense the steam to limit any pressure buildup.

The thermal efficiency of a CANDU reactor plant is only about 29%, but the CANDU reactor uses a larger fraction of U-235 in uranium ore than other reactors and also makes better use of the U-238 to Pu-239 conversion process to extend fuel burnup.

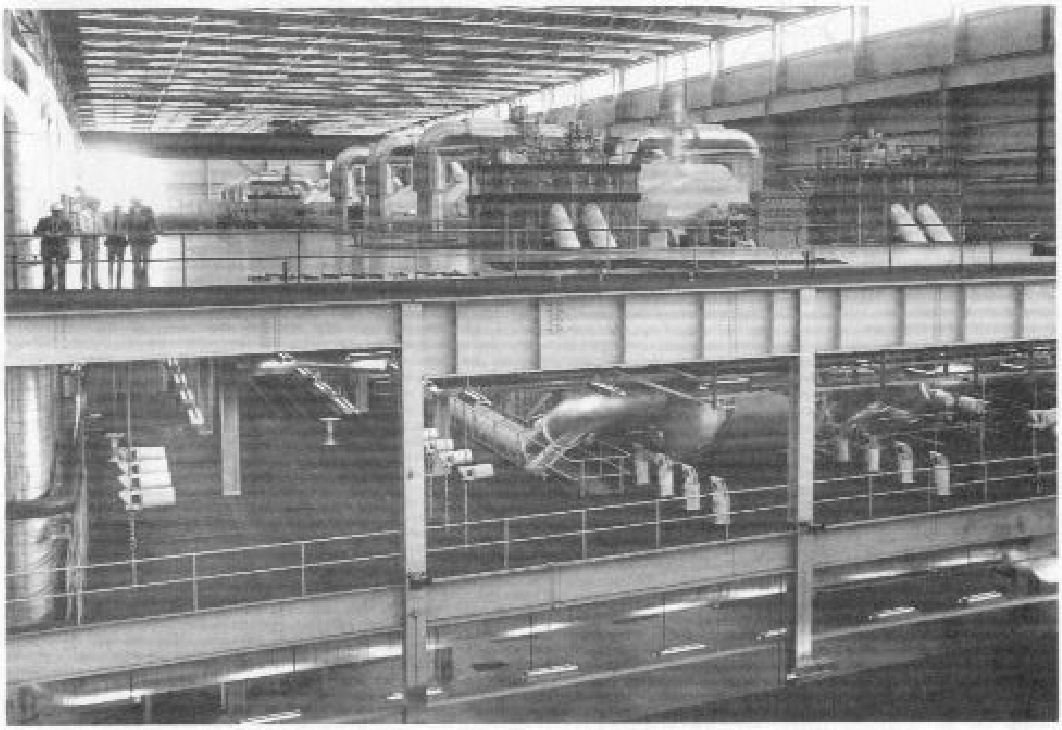


FIGURE 10.25 A turbine hall of the Pickering CANDU nuclear generating station. (Courtesy of Ontario Hydro, Toronto, Canada.)

Moreover, statistics show that, among large reactors, CANDU reactors have outstanding reliability records, with annual capacity factors (the ratio of annual electrical energy output to maximum possible annual output) as high as 96% and cumulative capacity factors as high as 88% (ref. 14).

10.8 Fast Reactors

Reactors may be designed to fission with fast neutrons, but these fast reactors must be more compact than thermal reactors so that the fast neutrons may produce fissions quickly before they are absorbed or moderated by surrounding materials. They are designed with structural materials that are poor absorbers and moderators of neutrons, such as stainless steel. The core of a fast reactor must contain a fissionable fuel of about 20% enrichment to compensate for the lowered probability of fissioning with high-energy neutrons.

Because of their high fuel density, fast reactors have a high power density that poses a difficult cooling problem. One solution is the use of a liquid metal as coolant. Liquid metals such as sodium and potassium have excellent heat transfer characteristics and do not interfere significantly with neutron functions.

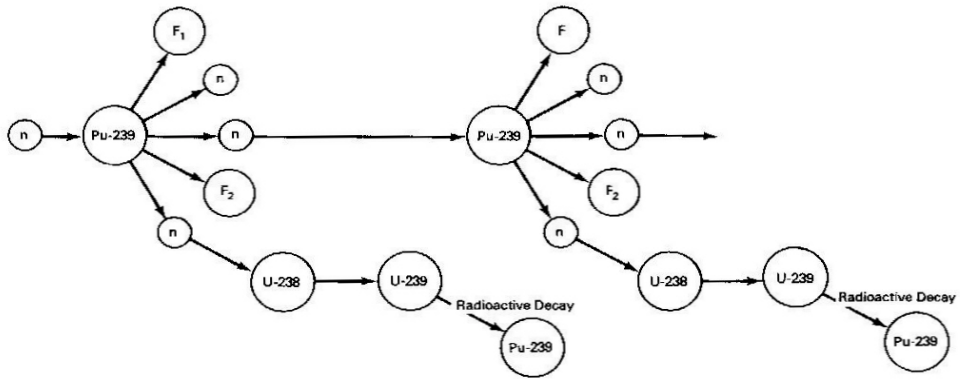


FIGURE 10.26 Two steps in the chain reaction in a plutonium reactor. Here, a fissionable Pu-239 atom is created for each Pu-239 atom consumed.

The choice of fuel used for thermal and fast reactors depends on the fuel's fission probability and the net number of neutrons produced per neutron absorbed. Most effective for thermal reactors is U-235, whereas Pu-239 is most suitable for fast reactors. In fissioning with fast neutrons, Pu-239 emits almost 20% more neutrons than does U-235. These additional neutrons are extremely important for making breeding practical, as will be discussed shortly. Because Pu-239 must be created from fertile U-238, a plutonium reactor can use fuel processed from fuel produced in a uranium thermal reactor or in another plutonium reactor. This would occur in the core of a plutonium fast reactor and in a *blanket* of U-238 surrounding the core, where additional plutonium is created using neutrons that escape from the core.

Figure 10.26 diagrams two steps in a chain reaction in a fast reactor where a plutonium atom is created for each one consumed. Note that each fission must produce a minimum of two neutrons for this reaction to continue. As a practical matter, more than two neutrons are required for complete replacement because of non-productive neutron captures.

A reactor that transmutes a fertile fuel to a fissionable fuel is called a *converter*. The *conversion efficiency* is the ratio of the number of new fissionable atoms produced to the number of atoms consumed in the fission. Thus the conversion efficiency of the reaction shown in Figure 10.26 is 1, because a new fissionable atom is created for each atom consumed. In this case there is no *net* fissionable fuel consumption.

Fast Breeder Reactors

The cover of a 1971 U. S. Atomic Energy Commission booklet (ref. 7), (see Figure 10.27) shows a notebook page that poses and answers the following question: Johnny had 3 truckloads of plutonium. He used 3 of them to light New York for 1 year. How much plutonium did Johnny have left? Answer: 4 truckloads. This neatly emphasizes the point that if unproductive neutrons in the process shown in Figure 10.26 were to

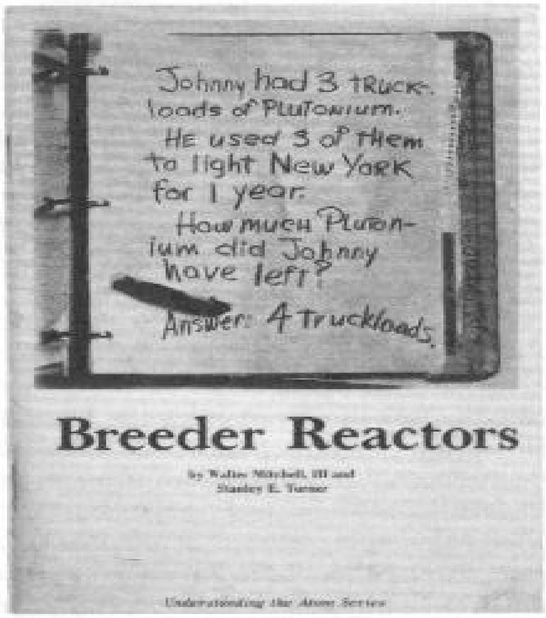


FIGURE 10.27 Why breeder reactors?

transmute a U-238 atom to Pu-239 occasionally, the chain reaction would produce more fissionable fuel than it consumes. The reactor would have a conversion efficiency greater than 1. A reactor that does this is called a *breeder reactor*. Thus, the breeder reactor provides a means of increasing the world supply of fissionable fuel as it generates power. While it may appear that the breeder provides something for nothing, it actually only takes advantage of the possibility of conversion of the fertile fuels through more efficient use of fission neutrons.

The breeder reactor is of great importance because it would allow the use of the vast store of U-238 in uranium ore that remains as a by-product of the U-235 enrichment process to provide fuel for current LWRs. This supply of U-238 has the potential to provide fuel for many years without further uranium mining. Fission of U-235 is currently the only natural large-scale source of neutrons. The continued use of low-conversion-efficiency reactors could preclude the eventual use of much of the energy resource of the U-238 in uranium ore.

One possibility for the design of a breeder reactor is a *liquid-metal fast breeder reactor*, LMFBR, which has the characteristics described briefly in the preceding section. The development of such a reactor involves careful design of its neutron economy and the development of a system of *fuel reprocessing* and *nuclear waste storage*. These topics will be considered in the next section.

In 1971, President Nixon set the development of a breeder reactor as a national goal and established a program for the development of an LMFBR pilot plant in Tennessee to be known as the Clinch River Breeder Reactor, CRBR. The vocal opposition of segments of the American public to all forms of nuclear power, coupled with well-known nuclear accidents, bureaucratic and construction delays, and concern for diversion of nuclear fuels to foreign military or terrorist uses, brought the development of new nuclear plants in the United States, including the CRBR, to a standstill. While some other nations, notably Japan, France, and Russia, continue to support extensive nuclear establishments and research, the path to the development of breeder reactors has been slow. The French, in particular, have developed experience with their 233-MW Phénix reactor, in operation since 1974, and an advanced 1200-MW LMFBR called the Super-Phénix, which went into operation in 1986. In 1994, the status of Super-Phénix was changed from production to a research facility; and in 1998 the French government directed that it be shut down, well before its planned 2015 shutdown (refs. 24 to 26). In Japan, delayed reporting of a minor sodium leak during commisioning of the Monjo breeder in 1995 produced a national turmoil that had serious repercussions for the Japanese breeder program. (Ref. 27).

10.9 Innovation in Reactor Design

During the hiatus in new domestic nuclear plant orders in the United States since the late 1970s, the industry has been working to incorporate new ideas and recommendations, based on utility operating experience, into the design of plants that will be more capable of winning public, government, and industry acceptance.

Popular opinion notwithstanding, the safety record of nuclear power plants in the United States has been a good one. No one has been killed in a nuclear accident in a U.S. power plant. Few other industries can approach that record. Nevertheless, serious and expensive accidents have taken place at home and abroad; and the threat of catastrophic accidents, while extremely remote, remains.

Designs are being considered by industry, the U.S. Department of Energy (DOE), the Electric Power Research Institute (EPRI), and the Nuclear Regulatory Commission (NRC) that would provide built-in systems and safeguards that are passive, rather than active, that must be depended on to perform when a malfunction occurs (ref. 15). Providing for natural conduction/convection cooling of the core in case of an accident, rather than depending on pumps to force water through the core, is one of such features under consideration. The General Electric Company, with an international team, is developing a 600 MW_e simplified / small boiling water reactor (SBWR) that incorporates this approach (ref. 28).

Nuclear plant incidents sometimes result from failure of an operator to interpret instrument data and act thereon or from instrument failure. Redundant instrumentation is provided to avoid such problems, but operators do not always analyze the readings correctly or make the right decisions. If the plant is designed to minimize the necessity of such actions, the likelihood of accidents can be significantly reduced. Thus, much effort focuses on reducing the complexity of reactor control. Designs have been produced that appear to accomplish this to a significant degree. The NRC must, of course, conduct hearings and evaluate the adequacy of any design before a plant is built.

One of the problems of the power industry is the lack of *standardization.* Every American power plant is different from every other because of the number of reactor suppliers and power plant engineering firms and the diverse requirements of the different utilities that buy them. One of the obvious reasons for the reliability of the CANDU reactors is that they are planned and constructed by Ontario Hydro and Atomic Energy of Canada, largely for Ontario Hydro. American suppliers are also striving for standardization, but resolving the differences between differing requirements and standards is a formidable problem. An advanced boiling water reactor (ABWR) program with an NRC certification (ref. 29) addresses these issues by providing a standardized design that places pumps within the reactor envelope, eliminates much external piping, and incorporates other passive safety features. Reference 23 indicates that this system benefits from a new NRC licensing process in which safety issues are resolved, with full public participation, before construction begins. ABWR planning calls for a forty-eight-month construction schedule once a site has been approved.

The long-term trend has been for utilities to seek larger and larger plants to take advantage of the economies of scale. When the annual rate of growth of the electrical power industry dropped drastically in the 1970s to 2–3% due to conservation and other factors, many of the utility requirements for large reactors became less critical or disappeared. Interest has appeared in smaller reactors that can be constructed economically and on a timely basis. One hope is that better quality control could be exercised, and better economics result if small reactors were built, entirely or in modular fashion, in a factory and shipped to and installed at the power plant site, rather than erected there. Some of theses issues are addressed in the ABWR and SBWR programs (ref. 23).

10.10. Nuclear Reprocessing and Waste Disposal

Spent fuel is fuel that has resided in a reactor for a year or more, has been depleted of much of its fissionable material, and includes a buildup of radioactive fission products. When a reactor is shut down for refueling, spent fuel assemblies are removed and partially spent assemblies may be repositioned in the core to obtain further fuel utilization. Ideally, the spent fuel would be reprocessed to reclaim the unused and newly created fissionable materials for use in new fuel rods, and the high-level waste would be isolated to minimize the volume of highly toxic wastes. Even without breeder reactors, current spent fuel inventories can provide ample Pu-239 to support reprocessing. In the United States, by law, fuel assemblies are being stored indefinitely in reactor-fuel storage pools in nuclear power plants until legal decisions are made on whether reprocessing will take place and on the final disposition of nuclear waste. Unfortunately, these decisions are among the most politically difficult ones of our times.

10.11 Concluding Remarks

Nuclear power engineering has become so hopelessly intertwined with politics that any speculation on the future of nuclear power must simultaneously consider both technical and political realities. Even the continuation and expansion of nuclear power generation in the United States is in question.

A major concern for many years has been the availability of fissionable resources for an expanding nuclear power generation industry. It has been widely recognized that U-235, as the sole natural neutron resource, places limits on extensive nuclear development. For that reason, countries with nuclear generation capability have planned on developing breeder reactors to greatly extend that neutron resource. It is possible that the continuing long-term use of reactors that do not provide for efficient conversion of the fertile fuels may result in the waste of this enormous energy resource.

While the American nuclear power industry languishes, foreign development of nuclear power continues in some areas. The eventual recovery of the American nuclear power industry may depend on the success of international nuclear development. The absence of CO_2 production by nuclear plants, and its global warming implications, gives an added incentive for international perpetuation and development of the industry.

Expansion of the nuclear power industry and development of a breeder reactor nuclear power economy inevitably entails the development of nuclear processing and hazardous waste disposal facilities and reactor fuel recycling. These activities arouse concerns about the possible diversion of nuclear materials for the production of weapons, (refs. 13 and 17–19). The tradeoffs among nuclear power, coal, and other energy conversion alternatives are also becoming more prominent as concerns over global warming and atmospheric pollution intensify. Well-thought-out and consistent government policies and regulation, as well as international cooperation, would give welcome direction to the power generation industry. Reference 16 observes that success in nuclear programs in other countries seems to correlate with limiting public intervention, and it questions whether the democratic institutions of the United States are consistent with the growth of the American nuclear power industry.

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EXERCISES

10.1 Consider the collision of a particle of mass m and velocity v with a stationary particle of mass M. Write energy and momentum equations for the collision. Derive an equation for the ratio of the final to the initial kinetic energy of the original moving particle in terms of the masses of both particles. Use the result to show why light atoms are used as moderators.

10.2 Evaluate the assertion of the title of reference 12 that 100 grams of uranium equal 290 tons of coal. Assume coal to be represented by pure carbon. Could you make the title of the book more accurate? If so how?

10.3 Estimate the core thermal power and thermal efficiency of a 1220-MW_e boiling water reactor that has 46,376 fuel rods in a 150-in.-high core with a maximum fuel rod linear energy density of 13.4 kW / ft and a fuel rod peak-to-average power release of 2.2. If there are 748 fuel assemblies, what is the average number of fuel rods per assembly? Estimate the number of thermally inactive rods in a reactor with an eight-by-eight fuel assembly array.

10.4 Assuming the fuel temperature to be 295K, calculate the energy of a thermal neutron using $3kT/2$ where k is the Boltzmann constant.

10.5 Study the literature and then discuss the details of the nuclear processes by which neutrons convert U-238 to fissionable fuel. Include nuclear reaction equations.

10.6 Study the literature, then discuss the details of the nuclear processes by which neutrons convert thorium to a fissionable fuel. Include nuclear reaction equations.

10.7 Sketch and label a PWR steam generator flow diagram and a T-s diagram for the two flows through the PWR steam generator, using the data from Table 10.2.

10.8 Determine the ratio of the reactor-loop flow rate to the steam flow rate from the data for the PWR given in Table 10.2.

10.9 Sketch a T-h diagram and determine the pinchpoint temperature difference for a PWR steam generator that has, respectively, 572.5°F and 630°F reactor inlet and outlet temperatures and steam generator inlet and outlet temperatures of 473°F and 603°F.

10.10 Discuss the changes in the neutron economy diagram (Figure 10.4) due to (a) insertion of control rods, (b) withdrawal of control rods, (c) increase and (d) decrease in recirculation flow rate in the boiling water reactor.

10.11 List from BWRs, PWRs, CANDUs, and LMFBRs those power reactors that are
(a) Thermal reactors
(b) Fast reactors
(c) Heavy-water reactors
(d) Light-water reactors
(e) Natural uranium reactors
(f) Enriched uranium reactors
(g) Plutonium reactors

10.12 Identify those power reactors that use moderators and those that do not.

10.13 Which power reactor (a) uses light water as coolant and has separate reactor coolant and steam loops, (b) uses heavy water and natural uranium, and (c) uses plutonium as fuel and liquid metal as coolant?

10.14 Based on data in Table 10.2 for the Oconee Unit 1 PWR, estimate the turbine-loop thermodynamic conditions (temperature and pressure) and power delivered by a simple Rankine-cycle steam turbine with 85% efficiency.

10.15 Compare the reactor temperature rise shown in Table 10.2 for the Oconee Unit 1 with an analysis based on heat transfer data given in the table. Discuss your result.

10.16 Compare the reactor temperature rise shown in Table 10.2 for the 1300-MW_e PWR with an analysis based on heat transfer data given in the table. Discuss your result.

10.17 Based on the data given in Table 10.2, determine the heat transfer rating for the Oconee Unit 1 PWR steam generator, and evaluate the average reactor-loop heat transfer loss rate and fraction. Discuss your result.

10.18 Based on data in Table 10.2, estimate the turbine-loop thermodynamic conditions (temperature and pressure) and power delivered by a simple Rankine-cycle steam turbine with 85% efficiency for the 1300-MW PWR.

10.19 Based on the data given in Table 10.2, determine the heat transfer rating for the 1300-MW_e PWR steam generator, and evaluate the average reactor-loop heat transfer loss rate and fraction. Discuss your result.

10.20 Develop a thermal design for a 2000-MW_e pressurized water reactor core with 0.4 in. diameter, 15-ft-long fuel rods having an average linear power output of 16 kW/ft. Assume an average film coefficient of 4500 Btu/hr-ft^2-R. Prepare a report.