Nuclear Power

Nuclear Fission Reactors

The discovery that several neutrons are emitted in the fission process led to speculation concerning the possibility of using these neutrons to initiate other fissions, thereby producing a *chain reaction*. On December 2, 1942, less than four years after Hahn and Strassmann's discovery of fission, a group led by Enrico Fermi produced the first self-sustaining chain reaction in a nuclear reactor that they had constructed at the University of Chicago.[24]

To sustain a chain reaction in a fission reactor, one of the neutrons (on the average) emitted in the fission of ^{235}U must be captured by another ^{235}U nucleus and cause it to fission. The *reproduction factor k* of a reactor is defined as the average number of neutrons from each fission that cause a subsequent fission. In the case of ^{235}U the maximum possible value of k is about 2.4, but it is normally less than this for two important reasons: (1) some of the neutrons may escape from the region containing fissionable nuclei, and (2) some of the neutrons may be captured by nonfissioning nuclei in the reactor. If k is exactly 1, the reaction will be self-sustaining. If it is less than 1, the reaction will die out. If k is significantly greater than 1, the reaction rate will increase rapidly and "run away." In the design of nuclear bombs, such a runaway reaction is necessary. In power reactors, the value of k must be kept very nearly equal to 1 (see Figure 11-50). If k is exactly equal to 1, the reactor is said to be *critical*; for $k < 1$, it is described as being *subcritical* and for $k > 1$ as *supercritical*.

Since the neutrons emitted in fission mostly have energies of the order of 1 MeV or higher (see Figure 11-51), whereas the cross section for neutron capture leading to fission in ^{235}U is largest at small energies as illustrated in Figure 11-52, the chain reaction can be sustained only if the neutrons are slowed down before they escape from the reactor. At high energies (1 to 2 MeV), neutrons lose energy rapidly by inelastic scattering from ^{238}U, the principal constituent of natural uranium. (Natural uranium contains 99.28 percent ^{238}U and only 0.72 percent fissile ^{235}U.) Once the neutron energy is below the excitation energies of the nuclei in the reactor (about 1 MeV), the main process of energy loss is by elastic scattering, in which a neutron collides with a nucleus at rest and, by conservation of momentum, transfers some of its kinetic energy to the nucleus. Such energy transfers are efficient only if the masses of the two bodies are comparable. A neutron will not transfer much energy in an elastic collision with a heavy ^{238}U nucleus. Such a collision is analogous to one between a marble and a billiard ball. The marble will be deflected by the much more massive billiard ball with essentially no change in its kinetic energy. Therefore, a *moderator* consisting of

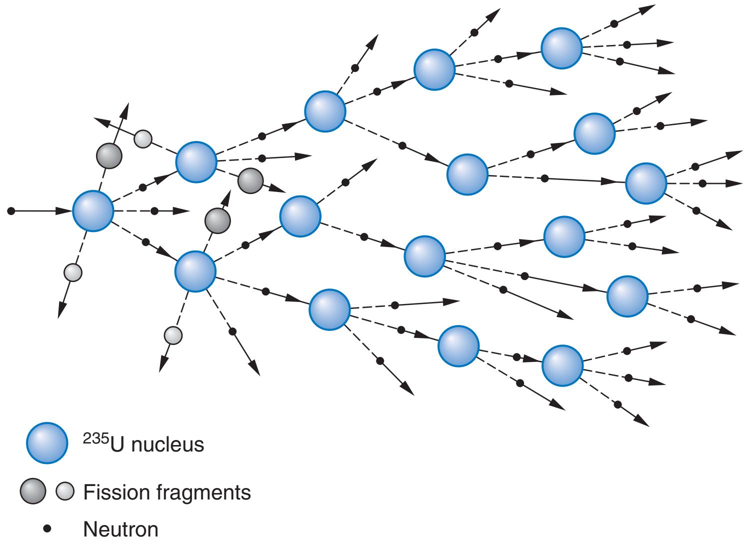


FIGURE 11-50 Schematic representation of a fission chain reaction in ^{235}U. The fission fragments are shown only for the first three fissions. The average number of neutrons produced is 2.4 per fission. In this example $k = 1.6$. Notice that, while there are 42 neutrons in the diagram, the judicious placement of absorbers to absorb as few as two of those causing fission would be sufficient to make $k = 1$ and control the reaction.

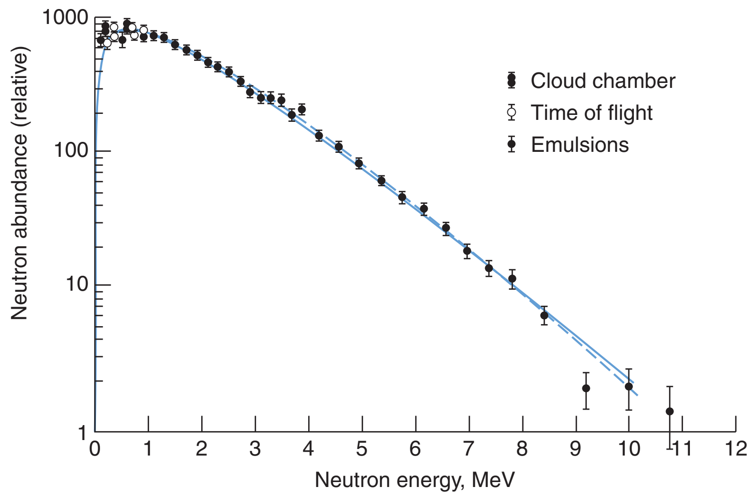


FIGURE 11-51 The energy spectrum of the neutrons emitted in the thermal-neutron-induced fission of ^{235}U. *[Data from R. B. Leachman, Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, vol. 2, United Nations, New York, 1956.]*

material such as water or graphite that contains light nuclei is placed around the fissile material in the core of the reactor to slow down the neutrons with relatively few collisions so as to minimize the number lost from the reactor. The neutrons are slowed down by elastic collisions with the nuclei of the moderator until they are in thermal equilibrium with the moderator, at which time they have approximately a Maxwell-Boltzmann energy distribution with average energy of $(3/2)kT$. Table 11-5 lists the approximate number of collisions needed to reduce 1 MeV neutrons to thermal energy for a few nuclei.

Reactors using ordinary water as a moderator cannot easily achieve $k = 1$ using natural uranium as a fuel for a combination of reasons. First, although the average number of neutrons emitted per fission is 2.4 (equal to the maximum value of k), we have noted that some of these are lost by escaping from the reactor or being absorbed in nonfission reactions. Recalling from Section 11-7 that the total cross section is the

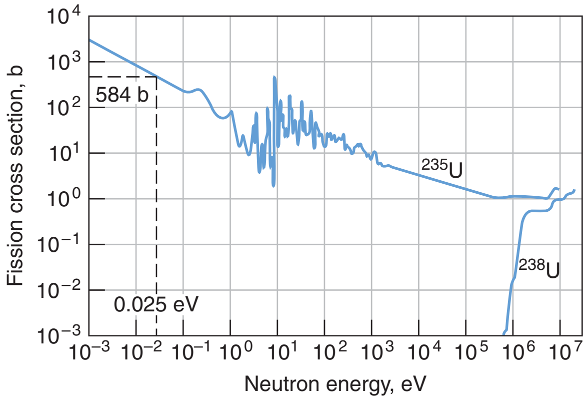


FIGURE 11-52 Neutron-induced fission cross sections for ^{235}U and ^{238}U. The region below 0.01 eV for ^{235}U shows the $1/v$ dependence, as does the cross section for the (n, γ) reaction and for the same reason. The radiative absorption reaction competes with fission and has a cross section of 97 b at 0.025 eV. The numerous resonances between 1 eV and 100 eV are associated with excited states of the $^{236}U^*$ nucleus.

Table 11-5 Properties of selected nuclei as moderators

Nucleus	$\sigma(n, \gamma)$ barns	Number of collisions to thermalize
^{1}H	0.333	18
^{2}H	0.51×10^{-3}	25
^{4}He	0	43
^{12}C	3.5×10^{-3}	110
^{238}U	2.75	2200

sum of the partial cross sections and recognizing that the ratio of the cross section for a particular reaction (Equation 11-62) and the total cross section is the relative probability of that reaction occurring, we see that then the relative probability that a thermal neutron will cause a fission reaction is given by $\sigma_f/(\sigma_f + \sigma_a)$, where σ_f is the partial cross section for fission and σ_a is the partial cross section for all other kinds of absorption of thermal neutrons. The latter are mainly (n, γ) reactions. Thus, we can write k as

$$k = 2.4\frac{\sigma_f}{\sigma_f + \sigma_a} \qquad \textbf{11-68}$$

The values of σ_f and σ_a for natural uranium are computed from the isotopic abundances above and the cross sections for each isotope. The fission cross section for ^{235}U is 584 b for thermal neutrons, while that for ^{238}U is zero. The cross sections for the (n, γ) reactions are 97 b for ^{235}U and 2.75 b for ^{238}U (see Table 11-5 and Figure 11-52). The values of σ_f and σ_a are then given by

$$\sigma_f = \frac{0.72}{100}\sigma_f(^{235}U) + \frac{98.28}{100}\sigma_f(^{238}U) = 4.20 + 0 = 4.20\text{ b}$$

and

$$\sigma_a = \frac{0.72}{100}\sigma_a(^{235}U) + \frac{98.28}{100}\sigma_a(^{238}U) = 0.70 + 2.73 = 3.43\text{ b}$$

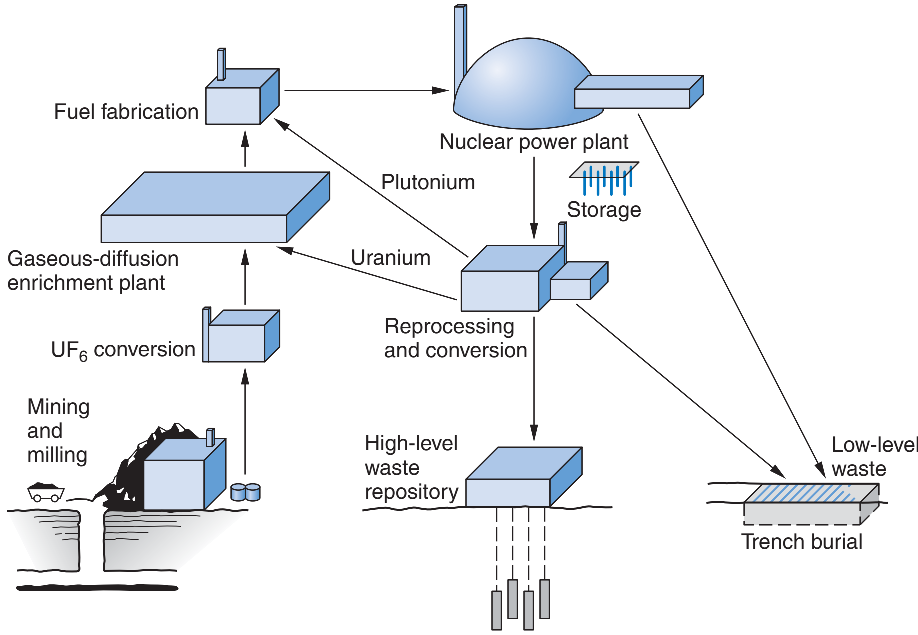


FIGURE 11-53 Schematic diagram of the *nuclear fuel cycle* for uranium-fueled light-water reactors. The UF_6 conversion plant converts solid U_3O_8, called *yellowcake* because of its color, into gaseous UF_6 for the enrichment facility. $^{235}UF_6$ is separated from $^{238}UF_6$ based on the fact that both molecules have the same average kinetic energy, $(3/2)kT$, and hence different diffusion rates due to their slightly different masses. The complete cycle includes the reprocessing facility and the repository for highly radioactive fission products.

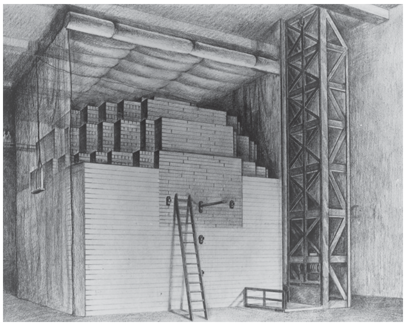
Table 11-6 Thermal-neutron fission cross sections for selected nuclei

Nuclide	Cross section (barns)	Critical energy for $A+1$ (MeV)
^{229}Th	30	8.3
^{230}Th	$<10^{-3}$	8.3
^{230}Pa	1500	7.6
^{233}Pa	$<10^{-1}$	7.1
^{233}U	531	6.5
^{234}U	$<5\times10^{-3}$	6.5
^{235}U	584	6.2
^{238}U	2.7×10^{-6}	5.9
^{236}Np	3000	5.9
^{238}Np	17	6.0
^{239}Pu	742	6.0
^{240}Pu	$<8\times10^{-2}$	6.3
^{241}Am	3.2	6.5
^{244}Am	2200	6.0
^{244}Cm	1	6.3
^{245}Cm	2000	5.9

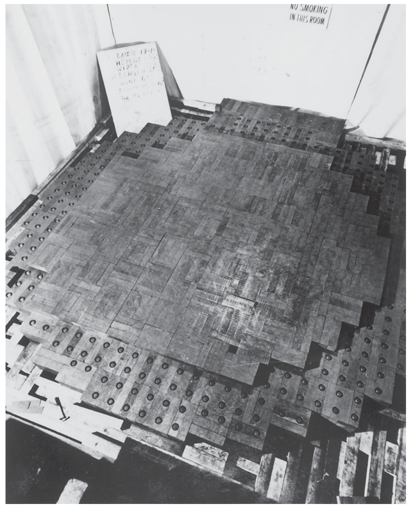
Therefore, the largest possible value of k that we can expect from natural uranium used as a reactor fuel is, from Equation 11-68,

$$k = 2.4\frac{\sigma_f}{\sigma_f + \sigma_a} = 2.4\frac{4.20}{4.20 + 3.43} = 1.32$$

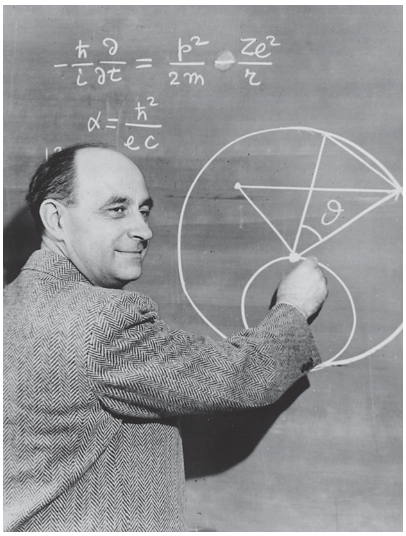
This value is already quite close to 1.0, so if loss of neutrons by leakage from the reactor or by absorption in the moderator is significant, the value of k may be easily less than 1.0. Because of the relatively large neutron capture cross section for the hydrogen nucleus (see Table 11-5), reactors using ordinary water as a moderator and natural uranium as a fuel have difficulty reaching $k = 1$. By enriching the uranium fuel in ^{235}U, that is, by increasing the ^{235}U content from 0.7 percent to, for example, 3 percent, the value of k computed from Equation 11-68 becomes 1.82, sufficient to make $k = 1$ attainable in operation. (Figure 11-53 shows where the enrichment process fits into the uranium fuel cycle.) Natural uranium can be used if heavy water (D_2O) is used instead of ordinary (light) water (H_2O) as the moderator. This is possible because the cross section of deuterium for the (n, γ) reaction for thermal neutrons is much smaller than that of ^{1}H. Although heavy water is expensive, most Canadian reactors use it for a moderator to avoid the cost of constructing uranium enrichment facilities and to help mitigate the problem of spent-fuel storage. Table 11-6 lists



(*a*)



(*b*)



(*c*)

(*a*) A sketch of the world's first nuclear reactor, the CP-1 (for Chicago Pile number 1). Projecting from the near face next to the top of the ladder is one of the cadmium-plated rods used to control the chain reaction by absorbing neutrons. The cubical balloon surrounding the reactor, open on the near side, was to contain neutron-activated radioactive air. News of the reactor's successful test was transmitted by A. H. Compton, one of those present, to President Roosevelt's advisor (and Harvard University president) J. B. Conant in a phone call thus: "The Italian navigator [i.e., Fermi] has landed in the New World," said Compton. "How were the natives?" asked Conant. "Very friendly," was Compton's reply. (*b*) The only photograph of CP-1 known to exist, taken during addition of the 19th layer of graphite. Alternate layers of graphite, containing uranium metal and/or uranium oxide, were separated by layers of solid-graphite blocks. Layer 18, almost covered, contained uranium oxide. (*c*) Enrico Fermi, leader of the group of scientists who succeeded in initiating the first man-made nuclear chain reaction, on December 2, 1942. [*(a) and (b) American Institute of Physics, Emilio Segrè Visual Archives; courtesy of Argonne National Laboratory, University of Chicago. (c) Courtesy of Argonne National Laboratory.*]

the cross sections for fission by thermal neutrons and the critical energies for several nuclei.

EXAMPLE 11-21 Thermal-Neutron Fission of ^{239}Pu and ^{233}Pa Determine the excitation energies of ^{239}Pu and ^{233}Pa when each absorbs a thermal neutron. Compare the results with the critical energies for fission and comment on the observed fission cross sections in Table 11-6.

SOLUTION

The excitation energy for ^{239}Pu is given by

$$E = [(M(^{239}\text{Pu}) + m_n) - M(^{240}\text{Pu})]c^2$$

$$= \left[(239.052157 + 1.008665)\,\text{u} - 240.053808\,\text{u} \right] c^2$$
$$= (0.007014\,\text{u})c^2 \times 931.5\,\text{MeV}/\text{u} \cdot c^2$$
$$= 6.53\,\text{MeV}$$

This is larger than the critical energy for fission in Table 11-6 by 0.53 MeV, so we expect ^{239}Pu to have a significant thermal-neutron-fission cross section, as it does.

A corresponding calculation for ^{233}Pa yields an excitation energy of 5.22 MeV, well below the critical energy for fission of 7.1 MeV given in Table 11-6. The observed low cross section of ^{233}Pa for fission by thermal neutrons is in agreement with that result.

Figure 11-54 shows some of the features of a pressurized-water reactor (PWR) commonly used in the United States to generate electricity. Of 131 reactors constructed since the development of commercial nuclear power began, 104 are currently (mid-2011) operating or operable. Of those, 70 are PWRs. Fission in the core heats the water in the primary loop, which is closed, to a high temperature. This water, which also serves as the moderator, is under high pressure to prevent it from boiling. The hot water is pumped to a heat exchanger, where it heats the water in the secondary loop and converts it to steam, which is then used to drive the turbines that produce electrical power. Note that the isolation of the water in the secondary loop from that in the primary loop prevents its contamination by exposure to the particles emitted by the radioactive nuclei in the reactor core. The remaining operational commercial nuclear power reactors in the United States are boiling-water reactors (BWR). These units do not have the secondary loop steam generator (see Figure 11-54). Instead, steam generated

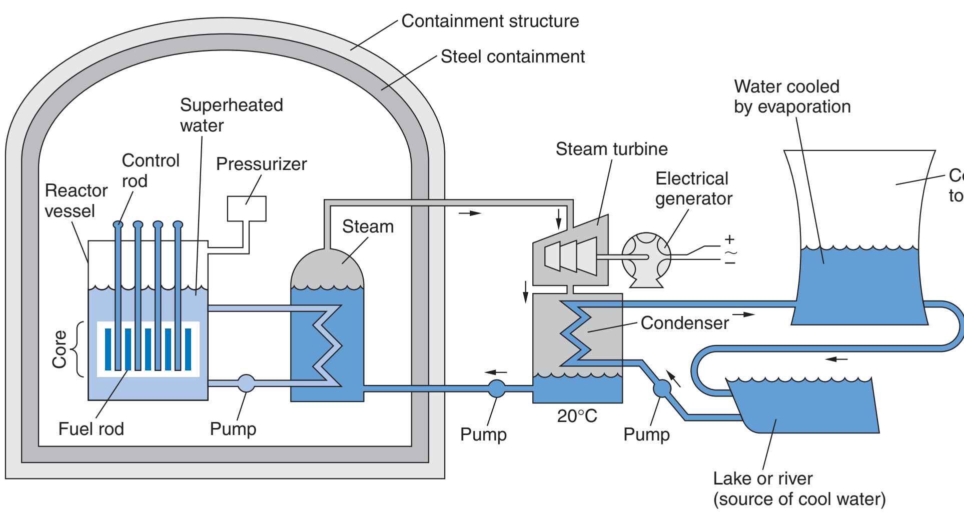
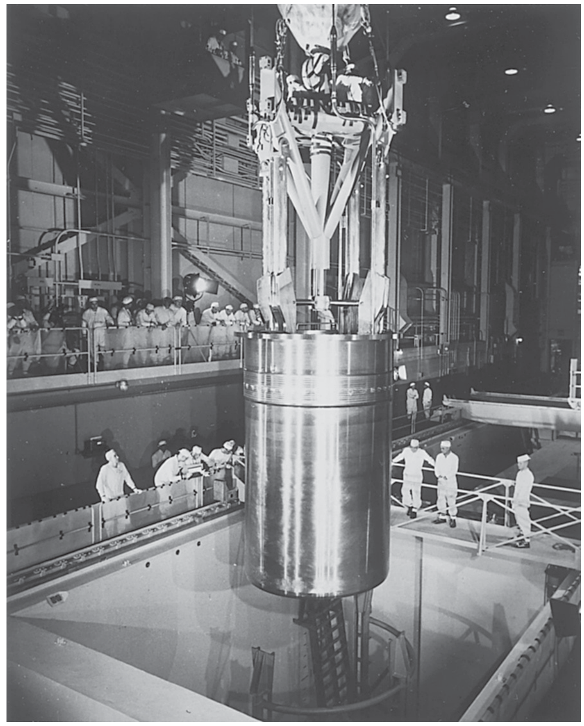


FIGURE 11-54 Simplified drawing of a pressurized-water reactor (PWR). The water in contact with the reactor core serves as both the moderator and the heat-transfer material. It is isolated from the water used to produce the steam that drives the turbines. Many features, such as the backup cooling mechanisms, are not shown here. A second type of power reactor, not shown here, is the boiling-water reactor (BWR). In this system steam produced from boiling water in the core is circulated directly to the turbine without using the isolation loop.

by boiling water surrounding the core is piped directly to the turbine. Inventories of nuclear power plants throughout the world are maintained by a number of organizations, for example, the European Nuclear Society, that are accessible on the Internet.

The ability to control the reproduction factor k is important if a power reactor is to operate with any degree of safety. There are both natural negative-feedback mechanisms and mechanical methods of control. If k is greater than 1 and the reaction rate increases, the temperature of the reactor increases. If water is used as a moderator, its density decreases with increasing temperature, and it becomes a less effective moderator. A second important method to control k is the use of control rods made of a material such as cadmium, which has a very large neutron capture cross section. When a reactor is started, the control rods are inserted, so that k is less than 1. As they are gradually withdrawn from the reactor, the neutron capture decreases and k increases to 1. If k becomes greater than 1, the rods are again inserted to the extent necessary to restore k to 1.

Control of the reaction rate of a nuclear reactor with mechanical control rods is possible only because some of the neutrons emitted in the fission process are delayed and because the time needed for a neutron to slow from 1 or 2 MeV to thermal energy and to diffuse through the fuel is of the order of a millisecond. (The number of neutrons available for fission is determined by a time constant consisting of the thermalization time, about a microsecond, and the diffusion time, about a millisecond.) If all the neutrons emitted in fission were prompt neutrons, that is, emitted immediately in the fission process, mechanical control would not be possible because statistical fluctuations in the number of prompt neutrons would cause the reactor to run out of control before the rods could be inserted. However, about 0.65 percent of the neutrons emitted are delayed by an average time of about 14 seconds, and it is these neutrons that make control of the reactor possible. These neutrons are emitted not in the fission process itself but in the decay of the fission fragments. A typical decay is



The pressure vessel containing the fuel core composed of 14 tons of natural uranium and 165 pounds of highly enriched uranium is shown being lowered into the world's first full-scale nuclear power plant at Shippingport, Pennsylvania, on October 5, 1957. The reactor was built by Westinghouse for the Duquesne Light Company. Small in comparison with reactors currently operating, the 90 MWe unit was shut down in 1982 and subsequently became the first commercial nuclear plant to be dismantled. The pressure vessel, now packed with concrete, is in storage at the federal nuclear waste facility in Hanford, Washington. *[American Institute of Physics, Emilio Segrè Visual Archives; courtesy of Westinghouse.]*

$$
\begin{array}{l}
{}^{87}\mathrm{Br} \rightarrow {}^{87}\mathrm{Kr}^{*} + \beta^{-} + \overline{\nu_e} \\
\qquad\quad\ \downarrow \\
\qquad\quad {}^{87}\mathrm{Kr}^{*} \rightarrow {}^{86}\mathrm{Kr} + n
\end{array}
$$

In the decay of ${}^{87}\mathrm{Br}$, which has a half-life of 56 s, the excitation energy of the ${}^{87}\mathrm{Kr}^{*}$ nucleus happens to exceed the neutron separation energy and a neutron is emitted. This neutron is thus delayed by 56 seconds on the average. The effect of the delayed neutrons can be seen in the following examples.

EXAMPLE 11-22 Fission Rate Doubling Time If the average time between fission generations (the time it takes for a neutron emitted in one fission to cause another) is 1 ms = 0.001 s and the reproduction factor is 1.001, how long will it take for the reaction rate to double?

SOLUTION

1. Since the initial reaction rate $R(0)$ times k is the reaction rate one generation later, then the reaction rate after N generation $R(N)$ is given by

$$R(N) = R(0)k^N$$

2. For the reaction rate to double, $R(N) = 2R(0)$:

$$2R(0) = R(0)k^N \quad \text{or} \quad 2 = (1.001)k^N$$

3. Solving this for the number of generations N:

$$N \ln(1.001) = \ln 2$$

$$N = \frac{\ln 2}{\ln(1.001)} = 693 \approx 700$$

Remarks: *Thus, it takes about 700 generations for the reaction rate to double. The time for 700 generations is 700(0.001 s) = 0.70 s. This is not enough time for response by the mechanical control system that inserts the control rods.*

EXAMPLE 12-8 Average Generation Time Assuming that 0.65 percent of the neutrons emitted are delayed by 14 s, find the average generation time and the doubling time if $k = 1.001$.

SOLUTION

Since 99.35 percent of the generation times are 0.001 s and 0.65 percent are 14 s, the average generation time is

$$t_{av} = 0.9935(0.001\text{ s}) + 0.0065(14\text{ s}) = 0.092\text{ s}$$

Note that these few delayed neutrons increase the generation time by nearly 100-fold. The time for 700 generations is

$$700(0.092\text{ s}) = 64.4\text{ s}$$

This is plenty of time for the mechanical insertion of control rods.

Breeder Reactors

Because the small fraction of ^{235}U in natural uranium limits the economically recoverable supply of uranium and because of the limited capacity of enrichment facilities, reactors based on the fission of ^{235}U cannot be expected to meet long-term energy needs. A possible alternative is the *breeder reactor*, a type of reactor that has the potential for producing more fissile fuel than it consumes. When the relatively plentiful but essentially nonfissile ^{238}U captures a neutron, the result is an (n, γ) reaction producing ^{239}U, which β^- decays (with a half-life of 23.5 minutes) to ^{239}Np, which in turn decays by β^- emission (with a half-life of 2.35 days) to the fissile nuclide ^{239}Pu as illustrated by Equation 11-64.

$$
\begin{array}{l}
{}^{238}\mathrm{U} + n \longrightarrow {}^{239}\mathrm{U}^* \longrightarrow {}^{239}\mathrm{U} + \gamma \\
\qquad\qquad\qquad\qquad \downarrow \\
\qquad\qquad\qquad\qquad {}^{239}\mathrm{U} \longrightarrow {}^{239}\mathrm{Np} + \beta^- + \bar{\nu}_e \\
\qquad\qquad\qquad\qquad\qquad\qquad \downarrow \\
\qquad\qquad\qquad\qquad\qquad\qquad {}^{239}\mathrm{Np} \longrightarrow {}^{239}\mathrm{Pu} + \beta^- + \bar{\nu}_e
\end{array} \qquad \textbf{11-69}
$$

Since ^{239}Pu fissions with fast neutrons in addition to slow neutrons, a *fast breeder reactor* needs no moderator. A further advantage is that the average yield of neutrons per absorbed neutron (that is, the maximum value of k) for ^{239}Pu is 2.7 for a neutron energy of 1 MeV. Thus, a reactor initially fueled with a mixture of ^{238}U and ^{239}Pu needs only one of the 2.7 neutrons to sustain the chain reaction and will breed as much fuel as it uses or more if one or more of the neutrons emitted in the fission of ^{239}Pu is captured by ^{238}U. Practical studies indicate that a typical fast breeder reactor can be expected to double its fuel supply in 7 to 10 years.

There are three major safety problems with fast breeder reactors that have limited their commercial use. The fraction of delayed neutrons is only 0.3 percent for the fission of ^{239}Pu, so the time between generations is much less than for ^{235}U-fueled fission reactors. Mechanical control is therefore much more difficult. Also, since the operating temperature of a breeder reactor is relatively high and a moderator is neither necessary nor desired, a heat-transfer material such as liquid sodium metal is used rather than water (which is the moderator as well as the heat-transfer material in an ordinary ^{235}U-fueled reactor). If the temperature of the reactor increases, the resulting decrease in the density of the heat-transfer material leads to positive feedback since it will absorb fewer neutrons than before, making more available for fission, which further increases the temperature, and so on. As it happens, there is another temperature-dependent process at work, particularly in fast breeder reactors, that helps mitigate this problem. As the temperature increases, the resonances for the (n, γ) reaction broadens due to the Doppler effect, increasing the number of neutrons absorbed and decreasing the number available for producing fission reactions. This is a very important intrinsic safety feature. The third problem concerns the loss of the reactor coolant. While this is also a serious problem for the water-moderated, thermal-neutron reactors discussed earlier, as we describe in the section "Safety Issues of Fission Reactors" that follows, it is particularly so for fast breeder reactors since the coolant is not serving as a moderator and its loss does not shut down the fission reactions. Because of these and other safety considerations and new, somewhat lower projections of future electricity demand, commercial breeder reactors are not yet in use in the United States. There are, however, several in operation in France, Great Britain, and Russia.

There is also the potential for constructing a breeder reactor utilizing thermal-neutron fissions. It is based on ^{232}Th, the only isotope of thorium that occurs naturally, and employs ^{233}U as the fissile nuclide. There is somewhat more thorium than uranium in Earth's crust, so the energy ultimately available is essentially doubled if both ^{238}U and ^{232}Th are used in breeder reactors. ^{233}U does not occur in nature. It is produced as follows:

$$
\begin{array}{l}
{}^{232}\mathrm{Th} + n \longrightarrow {}^{233}\mathrm{Th}^* \longrightarrow {}^{233}\mathrm{Th} + \gamma \\
\qquad\qquad\qquad\qquad\quad \downarrow \\
\qquad\qquad\qquad\qquad {}^{233}\mathrm{Th} \longrightarrow {}^{233}\mathrm{Pa} + \beta^- + \bar{\nu}_e \\
\qquad\qquad\qquad\qquad\qquad\qquad \downarrow \\
\qquad\qquad\qquad\qquad\qquad\qquad {}^{233}\mathrm{Pa} \longrightarrow {}^{233}\mathrm{U} + \beta^- + \bar{\nu}_e
\end{array} \qquad \textbf{11-70}
$$

The half-life of ^{233}Th is 22.3 min and that of ^{233}Pa is 27.0 days. The cross section of ^{233}U for fission by thermal neutrons is only slightly smaller than that of ^{235}U, but it produces more neutrons per fission, 2.5 compared with 2.4. The difference may not seem significant, but it is sufficient to allow ^{233}U to sustain a chain reaction and provide a neutron to initiate the breeding reaction of Equation 11-70. Neither ^{235}U nor ^{239}Pu produces enough neutrons on thermal-neutron-induced fission to breed replacement or excess fissile nuclei. Thus, ^{233}U is a better thermal reactor fuel than ^{235}U and is much better than ^{239}Pu.

Safety Issues of Fission Reactors

There has been heated debate concerning the safety of fission reactors, beginning with the Windscale accident in Great Britain in 1957 but particularly since the disastrous accident at Chernobyl in Ukraine (then part of the U.S.S.R.) in 1986, the less serious accident at the Three Mile Island generating station in the United States in 1979, and the devastating damage to the Japanese Fukushima generating station caused by a tsunami produced by an earthquake in 2011. A common concern expressed is that a reactor might explode as a uranium bomb, although that is virtually impossible. Even in light-water reactors, the enriched uranium contains only 1 to 4 percent ^{235}U, whereas a uranium bomb typically requires uranium enriched to 90 percent or more ^{235}U. Another, more realistic concern is *meltdown*, the melting of the fuel core because of the heat produced by the radioactive decay of the fission fragments that occurs even after the reactor is shut down. If the cooling system fails, even though the loss of the coolant/moderator in light-water reactors halts fission, it is possible that the core would melt and, in a worst-case scenario, melt its way through the floor of the containment building into the ground. Meltdown did not occur at Chernobyl; however, about 40 percent of the core melted at Three Mile Island and considerable meltdown apparently occurred at Futushima, although the extent was not known at this writing (summer 2011). A partial meltdown also occurred at the Enrico Fermi reactor, an LMFBR, near Detroit in 1966. The Enrico Fermi reactor, the only commercial LMFBR ever built in the United States to date, was not restarted and was subsequently decommissioned. At neither Three Mile Island nor Enrico Fermi was the reactor containment breached.

A potentially more serious problem is that radioactive material may be released into the atmosphere, as did occur at Chernobyl and Futushima. The reactor at Chernobyl was a graphite-moderated reactor designed to produce plutonium for weapons as well as electrical power. At the time of the accident, it was running at low power but with the separate cooling system partially disabled. The heat developed by the continuing fission was sufficient to ignite the graphite, which in turn ignited the uranium fuel itself. There are no comparable dual-purpose reactors outside Ukraine and Russia that are operated in this way. A similar accident with water-cooled and moderated reactors is probably not possible. Furthermore, a common safety feature, not used at Chernobyl, is a containment building with walls of concrete and steel at least 1 m thick. The extent of containment damage at Futushima is not yet fully known (see Figure 11-54).

With any type of fission reactor, there is the problem of storage of the long-lived radioactive waste products produced. Despite the fact that elaborate storage methods are used, their long-term efficacy is always open to question. Nuclear fuel reprocessing raises a number of additional safety questions. Among these are the proper means for safe, long-term (thousands of years) storage of high-level wastes from the process, the potential for clandestine diversion of reprocessed fuel, particularly ^{239}Pu, to weapons use, and the release of airborne radioisotopes from reprocessing facilities. Widely accepted solutions for many of these questions are not yet in hand.