NUCLEAR REACTORS

1. Introduction

The nuclear reactor is a device in which a controlled chain reaction takes place involving neutrons and a heavy element eg, uranium. Neutrons are typically absorbed in nuclei of uranium-235 [15117-96-1], ^{235}U, or plutonium-239 [15117-48-3], ^{239}Pu. These nuclei split, releasing two fission fragment nuclei and several fast neutrons. Some of these neutrons cause fission in other uranium nuclei in a sequence of events called neutron multiplication. The fission fragments are stopped within the nuclear fuel, where their kinetic energy becomes thermal energy. The thermal energy is removed by a cooling agent and converted into electrical energy in a turbine-generator system. Many of the fission fragments are radioactive, releasing radiation and decay heat. Some of the radioactive materials have useful purposes; others form nuclear waste.

Nuclear reactors as a source of heat energy and radiation were the outgrowth of World War II defense applications. Research and development was pursued on several fronts in the Manhattan Project. Success of a graphite and uranium pile built and tested at the University of Chicago in 1942 prompted construction of production reactors at Hanford, Washington, to accumulate plutonium for the atomic bomb. A second approach to obtaining weapons material involved uranium isotope separation methods. Research was successful on two techniques, the electromagnetic process at the University of California and gaseous diffusion at Columbia University. Oak Ridge, Tennessee, became the enriched-uranium production center, utilizing both methods. At the same time, knowledge was gained at Los Alamos, New Mexico, about conditions for controlled chain reactions in uranium and plutonium assemblies. The Manhattan Project culminated in the use of nuclear weapons in Japan to end World War II. The first public account of the nuclear project was the book by Smyth (1). The most comprehensive reference is the book by Rhodes (2).

After the war, the U.S. Atomic Energy Commission (AEC) was formed, as the predecessor of the Nuclear Regulatory Commission (NRC) and the Department of Energy (DOE). The AEC led U.S. research and development programs on nuclear naval vessels and central station power plants, in cooperation with industry. Excellent accounts of the period 1947–1961, during which the designs of newer reactors came into being, are found in Hewlett and Duncan (3) and Hewlett and Holl (4).

A variety of nuclear reactor designs is possible using different combinations of components and process features for different purposes. Two versions of the light water reactors were favored: the pressurized water reactor (PWR) and the boiling water reactor (BWR). Each requires enrichment of uranium in ^{235}U. To assure safety, careful control of coolant conditions is required.

The minimum ingredients of a reactor, where the basic reactions are nuclear rather than chemical, are neutrons and a fuel eg, uranium, the atoms of which can undergo fission. Products of the fission process, in order of importance, are (1) heat energy, originally in the form of kinetic energy of particles; (2) neutrons, originally of high energy, which can be slowed to lower energies; (3) radionuclides, originally as fission fragments and collectively called fission products; (4) beta and gamma radiation, released in the fission process and by decay of fission products, which contributes both to heat and hazard; and (5) neutrinos, which play no role, because of the ease with which they penetrate matter. The distribution of energy among these products of fission is as follows, for a total of 200 MeV (5).

Particle	Energy, MeV
fission fragments	166
neutrons	5
prompt γ-rays	7
fission product γ-rays	7
beta particles	5
neutrinos	10

2. Design

The chronology of the development of nuclear reactors can be divided into several principal periods: pre-1939, before fission was discovered (6); 1939−1945, the time of World War II; 1945−1963, the era of research, development, and demonstration; 1963−mid-1990s, during which reactors have been deployed in large numbers throughout the world; and extending into the twenty-first century, a time when advanced power reactors are expected to be built. Design of nuclear reactors has been based on a combination of theory, measurement of basic and derived parameters, and experiments with complete systems. Accounts of the development of reactors are discussed in a number of books (7–9).

A number of books have been written on reactor physics or reactor analysis (10–15). Each of them describes the following fundamentals of reactor operation. The nuclear chain reaction can be modeled mathematically by considering the probable fates of a typical fast neutron released in the system. This neutron may make one or more collisions, which result in scattering or absorption, either in fuel or nonfuel materials. If the neutron is absorbed in fuel and fission occurs, new neutrons are produced. A neutron may also escape from the core in free flight, a process called leakage. The state of the reactor can be defined by the multiplication factor, k, the net number of neutrons produced in one cycle. If k is exactly 1, the reactor is said to be critical; if $k < 1$, it is subcritical; if $k > 1$, it is supercritical. The neutron population and the reactor power depend on the difference between k and 1, ie, $\delta k = k - 1$. A closely related quantity is the reactivity, $\rho = \delta k/k$. If the reactivity is negative, the number of neutrons declines with time; if $\rho = 0$, the number remains constant; if ρ is positive, there is a growth in population.

The mathematical model originally used for steady-state behavior of a reactor was diffusion theory, a simplification of transport theory that in turn is an adaptation of Boltzmann's kinetic theory of gases. By solving a differential equation, the flux distribution in space and time was found or the conditions on materials and geometry that give a steady-state system were determined.

A key parameter in determining the possibility of a self-sustained chain reaction is the value of k for an infinite medium, k_∞. In the four-factor formula,

$$k_\infty = \epsilon p f \eta$$

the succession of events in the neutron cycle are shown, where ε represents the fast fission factor; p the resonance escape probability; f the thermal utilization; and η the reproduction factor. A finite assembly k-effective, k_e, is defined as

$$k_e = k_\infty L$$

where L is a nonleakage probability, which depends on neutron slowing and diffusion properties in reactor materials and on the size and shape of the reactor. Critical experiments in which fuel is gradually accumulated determine the critical mass of uranium or the number of fuel assemblies.

The analysis of steady-state and transient reactor behavior requires the calculation of reaction rates of neutrons with various materials. If the number density of neutrons at a point is n and their characteristic speed is v, a flux $\phi = nv$ can be defined. With an effective area of a nucleus as a cross-section σ, and a target atom number density N, a macroscopic cross-section $\Sigma = N\sigma$ can be defined, and the reaction rate per unit volume is $R = \phi\Sigma$. This relation may be applied to the processes of neutron scattering, absorption, and fission in balance equations leading to predictions of k_e, or to the determination of flux distributions. The consumption of nuclear fuels is governed by time-dependent differential equations analogous to those of Bateman for radioactive decay chains. The rate of change in number of atoms N owing to absorption is as follows:

$$dN/dt = -\phi\Sigma_a$$

Greater detail in the treatment of neutron interaction with matter is required in modern reactor design. The neutron energy distribution is divided into groups governed by coupled space-dependent differential equations.

The simplest model of time-dependent behavior of a neutron population in a reactor consists of the point kinetics differential equations, where the space dependence of neutrons is disregarded. The safety of reactors is greatly enhanced inherently by the existence of delayed neutrons, which come from radioactive decay rather than fission. The differential equations for the neutron population, n, and delayed neutron emitters, c_i, are

$$\frac{dn}{dt} = (\rho - \beta)\frac{n}{\Lambda} + \sum \lambda_i c_i$$

$$\frac{dc_i}{dt} = \beta_i \frac{n}{\Lambda} - \lambda_i c_i$$

where the reactivity $\rho = \delta k_e/k_e$ and the neutron cycle time is Λ. Decay constants are λ_i, and delayed fractions are β_i, where $i = 1,2,\ldots,6$, and β is the sum of β_i over i.

A fraction β of only 0.0065 of the neutrons having an average delay of 13 s greatly slows down the transient response to a change in k_e. In this simple reactor kinetics model, neglecting temperature effects, a positive reactivity gives, for long times after application, an exponential increase in neutron number with time, ie, $n(t)$ is proportional to $\exp(t/T)$, with T the asymptotic period.

More generally, the neutron number density and the reactor power distribution are both time- and space dependent. Also, there is a complex relation between reactor power, heat removal, and reactivity.

Operation of a reactor in steady state or under transient conditions is governed by the mode of heat transfer, which varies with the coolant type and behavior within fuel assemblies (16–21). Qualitative understanding of the different regimes using water cooling can be gained by examining heat flux, q'', as a function of the difference in temperature between a heated surface and the saturation temperature of water (Fig. 1).

In region A of Fig. 1, transfer is by convection between heated fuel surfaces and contacting water. In region B, evaporation of water into bubbles occurs at points. The bubbles detach and rise owing to buoyancy, carrying vapor and the heat of vaporization with them. As the temperature difference increases into region C, a transition occurs because the vapor that begins to coat the surface is a poorer conductor of heat than was the liquid in the previous region. This region is unstable because of intermittent wetting. Finally, film boiling, or post-dryout heat transfer (region D), takes place, where sufficient wall temperatures have developed to cause the heat flux again to increase.

The mathematical formulation of forced convection heat transfer from fuel rods is well described in the literature. Notable are the Dittus–Boelter correlation (18) for pressurized water reactors (PWRs) and gases, and the Jens–Lottes correlation (19) for boiling water reactors (BWRs) in nucleate boiling.

Designs are sought that maximize inherent safety by achieving a net negative reactivity feedback, where any tendency for the power of the reactor to increase and raise temperatures results in a counteracting effect. Several mechanisms are available. For a reactor in which neutron multiplication depends strongly on moderation, thermal expansion of the fluid or the creation of steam bubbles results in reduced neutron thermalization and greater neutron leakage. Both effects provide negative reactivity feedback. For a reactor having considerable uranium-238 content, a fuel temperature rise changes the rate of interaction between neutrons and the fuel in the resonance region. This is the Doppler effect. Such effects are quantified by measurements of various coefficients of reactivity, eg, temperature, power, Doppler, or void. For safety, the net coefficient must be sufficiently negative at all times.

Account must be taken in design and operation of the requirements for the production and consumption of xenon-135 [14995-12-1], ^{135}Xe, the daughter of iodine-135 [14834-68-5], ^{135}I. Xenon-135 has an enormous thermal neutron cross-section, 2.7×10^{-18} cm^2 (2.7×10^6 barns). Its reactivity effect is constant when a reactor is operating steadily, but if the reactor shuts down and the neutron flux is reduced, xenon-135 builds up and may prevent immediate restart of the reactor.

Several of the reactor physics parameters are both measurable and calculable from more fundamental properties eg, the energy-dependent neutron cross-sections and atom number densities. An extensive database, Evaluated Nuclear Data Files (ENDF), has been maintained over several decades. There is an interplay between theory and experiment to guide design of a reactor, as in other engineering systems.

The results of design studies, calculations, and experiments for the reactor types selected for investigation in the post-World War II period were collected in the multivolume proceedings of several international conferences at Geneva, the first of which was in 1955 (22). The U.S. Atomic Energy Commission (AEC) sponsored the publication of individual books in 1958 describing the status of several reactor types (23–27). The book by Nero (28) gives descriptions of the various reactors. Several textbooks provide information on reactor design, construction, and operation (29–34).

3. Power Generation

The primary use of a nuclear reactor is as a compact alternative heat source, wherever that is needed. Because a reactor is a complex system, it is economical as a power source only if it produces fairly large blocks of power, especially for steady electric power in the 500−1500-MWe range. If, however, expense is not a factor, a reactor could be used to provide electricity in small units to remote locations in which it is difficult to supply ordinary fuels. Electrical energy from reactors can be used to provide mechanical energy for propulsion of ocean vessels, including submarines, aircraft carriers, cargo ships, and icebreakers. A reactor can provide heat only, for a variety of applications, including process steam production, district heating, desalination of water, and direct propulsion of space vehicles.

The neutrons in a research reactor can be used for many types of scientific studies, including basic physics, radiological effects, fundamental biology, analysis of trace elements, material damage, and treatment of disease. Neutrons can also be dedicated to the production of nuclear weapons materials eg, plutonium-239 from uranium-238 and tritium, ^{3}H, from lithium-6. Alternatively, neutrons can be used to produce radioisotopes for medical diagnosis and treatment, for gamma irradiation sources, or for heat energy sources in space.

As of December 31, 2006, there were 104 U.S. commercial nuclear generating units that are fully licensed by the U.S. Nuclear Regulatory Commission (NCR) to operate in the United States. Of these 104 reactors, 69 are categorized as pressurized water reactors (PWRs) totaling 67,291 net megawatts (electric) and 35 units are boiling water reactors (BWR) totaling 34,223 net megawatts (electric), giving a total of 101,514 MWe. In 32 countries of the world there were 439 reactors in operation, with total power 372,711 MWe.

Although the United States has the most nuclear capacity of all nations, no new commercial reactor has come on line since May 1996. The current Administration has been supportive of nuclear expansion, emphasizing its importance in maintaining a diverse energy supply. As of 2007, however, no U.S. nuclear company had yet received a new construction permit.

The last reactor to come on line in the United States, in May 1996, was Watts Bar in Tennessee, owned and operated by the Tennessee Valley Authority.

Nevertheless, U.S. commercial nuclear capacity has increased in recent years through a combination of license extensions and uprating (upgrading) of existing reactors (35).

Table 1 gives information on historical and projected operable nuclear capacities by region for 2001–2025.

4. Classification

Nuclear reactors can be classified in a variety of ways: by purpose or use, key components, method of heat extraction, role in application, neutron energy, power level or neutron flux, arrangement of materials, stage of development, and manufacturer. Reactors are used for heat power, electrical power, propulsion, training, neutron production for basic research or materials testing, and for radioisotope production, including weapons materials. The U.S. commercial power plants are called light water reactors because ordinary water is used for neutron slowing and heat removal. Canadian reactors use deuterium oxide, ie, heavy water, and some European reactors use graphite.

Most nuclear reactors use a heat exchanger to transfer heat from a primary coolant loop through the reactor core to a secondary loop that supplies steam to a turbine. The pressurized water reactor is the most common example. The boiling water reactor, however, generates steam in the core.

The role of the reactor may be either as a converter, which produces some plutonium by neutron absorption in uranium-238 but depends on uranium-235 for most of the fission, or as a breeder, which contains a large amount of plutonium and produces more fissile material than it consumes. Breeding is also possible using uranium-233 produced by neutron absorption in thorium-232.

The characteristic neutron energy, ie, that at which most of the fission occurs, may be thermal, fast, or intermediate. In thermal reactors, those that have a moderator, the typical energy is on the order of a fraction of an electron volt (eV). Fast reactors, in which little neutron slowing occurs, operate with the neutrons in the megaelectronvolt (MeV) range. Intermediate reactors are those that operate on epithermal neutrons, those above thermal, but considerably slowed from fission energy. The reactor power level or neutron flux is another classification system. For example, a low power reactor, high flux reactor, or 1200-MW version all aid in identifying a given system.

Another category is that of arrangement. Historically, studies were made of homogeneous reactors, where the fuel was solid metal, or in solution or slurry. All modern reactors are heterogeneous, with the fuel and moderator in distinctly separate zones. The term heterogeneous is also applied to an alternative arrangement of materials in a fast breeder reactor. Also, in the fast breeder there are two arrangements of the coolant: the loop, in which fluid circulates through the core and heat exchanger; and the pot, a large reservoir in which the core sits.

Reactors may be experimental, test, prototype, demonstration, or commercial. In the United States, the four principal companies that have designed and built most of the power reactors are Westinghouse Corporation, General Electric Company, Babcock and Wilcox Company, and Combustion Engineering. There are several important foreign manufacturers as well, eg, Framatome in France; Atomic Energy of Canada, Ltd. in Canada; Mitsubishi and Toshiba in Japan; Siemens AG in Germany; and UKAEA in the United Kingdom.

Herein reactors are described in their most prominent application, that of electric power production. Five distinctly different reactors, ie, pressurized water reactors, boiling water reactors, heavy water reactors, graphite reactors, and fast breeder reactors are emphasized.

5. Reactor Components

Several components are required in the practical application of nuclear reactors (30–34). The first and most vital component of a nuclear reactor is the fuel, which is usually uranium slightly enriched in uranium-235 [15117-96-1] to 3–5%, in contrast to natural uranium, which has 0.72% ^{235}U. Less commonly, reactors are fueled with plutonium produced by neutron absorption in uranium-238 [24678-82-8]. Even more rare are reactors fueled with uranium-233 [13968-55-3], produced by neutron absorption in thorium-232. The chemical form of the reactor fuel typically is uranium dioxide, UO_2, but uranium metal and other compounds have been used, including nitrides, carbides, and molten salts.

The second important component is the cooling agent or reactor coolant, which extracts the heat of fission for some useful purpose and prevents melting of the reactor materials. The most common coolant is ordinary water at high temperature and pressure to limit the extent of boiling. Other coolants that have been used are liquid sodium, sodium–potassium alloy, helium, air, and carbon dioxide. Surface cooling by air is limited to unreflected test reactors or experimental reactors operated at very low power.

The third component is the moderator, a substance containing light elements eg, hydrogen, deuterium, or carbon. Because low ($\sim$ 0.025 eV) energy neutrons are much more effective in causing fission than high ($\sim$ 2 MeV) energy neutrons, such a medium is desirable to slow neutrons by causing multiple collisions. The lighter the nuclear target, the greater is the energy loss per collision. Thermal reactors are those having a moderator that brings neutrons down to energies comparable to the thermal agitation of atoms. Fast reactors do not have a moderator, and because there is limited slowing, neutrons remain at $\sim$1 MeV.

The fourth component is the set of control rods, which serve to adjust the power level and, when needed, to shut down the reactor. These are also viewed as safety rods. Control rods are composed of strong neutron absorbers, eg, boron, cadmium, silver, indium, or hafnium, or an alloy of two or more metals.

The fifth component is the structure, a material selected for weak absorption for neutrons, and having adequate strength and resistance to corrosion. In thermal reactors, uranium oxide pellets are held and supported by metal tubes, called the cladding. The cladding is composed of zirconium, in the form of an alloy called Zircaloy. Some early reactors used aluminum; fast reactors use stainless steel. Additional hardware is required to hold the bundles of fuel rods within a fuel assembly and to support the assemblies that are inserted and removed from the reactor core. Stainless steel is commonly used for such hardware.

If the reactor is operated at high temperature and pressure, a thick-walled steel reactor vessel is needed.

The sixth component of the system is the shield, which protects materials and workers from radiation, especially neutrons and gamma rays. Concrete is commonly used, augmented by iron and lead for gamma rays and water for fast neutrons.

6. Graphite Reactors

The first nuclear reactor made was composed of graphite, the only moderator available at that time, 1942, for use with natural uranium. Reactors for the production of plutonium during World War II and for power in the United Kingdom also utilized carbon in the form of graphite. A modern helium-cooled graphite reactor has been tested. A distinct advantage of having carbon as the moderator is that it provides the ability to use lower enriched or natural uranium as fuel, avoiding the necessity of expensive and power-absorbing isotope separation facilities.

6.1. The First Reactor. When word about the discovery of fission in Germany reached the United States, researchers found that (1) the principal uranium isotope involved was uranium-235; (2) slow neutrons were very effective in causing fission; (3) several fast neutrons were released; and (4) a large energy release occurred. The possibility of an atom bomb of enormous destructive power was visualized.

At about the same time, the artificial isotope plutonium-239 [15117-48-3] was discovered and was recognized as also being fissionable. This led to the conjecture that a controlled chain reaction might be achieved and that neutrons could be used to produce enough plutonium for a weapon. Experiments were conducted during the Metallurgical Project, centered at the University of Chicago, and led by Enrico Fermi. Subcritical assemblies of uranium and graphite were built to learn about neutron multiplication. In these exponential piles, the neutron number density decreased exponentially from a neutron source along the length of a column of materials. There was excellent agreement between theory and experiment.

A larger assembly, ie, one that might be self-sustaining, or critical, was built. Of special importance was the need for graphite of sufficiently high purity, because traces of boron would absorb neutrons and prevent multiplication. The pile was constructed of 37 layers of graphite blocks where chunks of uranium oxide and uranium metal alternated with the layers of graphite only. To control the reaction and provide safety in case of accident, a set of neutron-absorbing rods and an emergency cadmium solution were provided. On December 2, 1942 the reactor was brought to critical and the power allowed to increase to a few hundred watts (36,37). Safety aspects of that experiment are highlighted in a 1988 article (38).

7. The Hanford Production Reactors

On the basis of the success of the first reactor experiment, construction of several plutonium production reactors began at Hanford, Washington. These reactors used graphite as moderator, but because high power levels were involved in producing the required amounts of plutonium, water cooling of the fuel was provided. Graphite blocks were pierced by holes lined with aluminum tubes, into which aluminum-canned uranium metal cylinders known as slugs were placed. After a period of operation, the slugs could be pushed along and discharged.

Chemical processing or reprocessing (39) of the fuel to extract the plutonium and uranium left a residue of radioactive waste, which was stored in underground tanks. By 1945, the reactors had produced enough plutonium for two nuclear weapons. One was tested at Alamogordo, New Mexico, in July 1945; the other was dropped at Nagasaki in August 1945.

A second approach to the production of weapons material was the uranium electromagnetic separation process, based on research at the University of California and production facilities at Oak Ridge, Tennessee. In a two-stage process, uranium of $> 90\%$ U-235 was obtained for use at Hiroshima.

7.1. Magnox and AGR Reactors. The greatest use of graphite has been in the United Kingdom (40). In the period 1956–1960 the first eight 50-MWe reactors were built and put into operation at Calder Hall and Chapelcross. These used natural uranium in the form of metal rods, clad with a magnesium alloy, Magnox, and using carbon dioxide as coolant. The reactor cores were very large and contained as many as 10,000 fuel channels. Many of these reactors were rather inefficient in converting heat to electricity, but were especially reliable. Later, the United Kingdom and France built advanced gas-cooled reactors (AGR), which were much more compact. These latter used slightly ($\sim 2\%$ ^{235}U) enriched uranium as fuel.

7.2. Sodium Graphite Reactor. A reactor cooled by liquid sodium and moderated by graphite can take advantage of excellent heat-transfer features and low neutron absorption, permitting use of low enrichment uranium (24). El Wakil (41) provides information on liquid metals as coolants. The sodium reactor experiment (SRE) and the Hallam, Nebraska nuclear power facility (HNPF), both designed by Atomics International (AI), were the only U.S. examples of this reactor type. These were part of the Atomic Energy Commission's power reactor demonstration program. The 75-MWe HNPF reactor used liquid sodium as coolant. Its fuel consisted of slugs of uranium–molybdenum alloy in Zircaloy-2 tubes, as 18-rod clusters. A few tests of uranium carbide fuel were made. The Hallam reactor operated from 1962 to 1964, when the thin stainless steel cans for the graphite blocks developed leaks that admitted sodium. In that brief period, many technical problems were solved, the reactor concept was tested successfully, and the ability to manage large volumes of sodium was demonstrated (42).

7.3. High Temperature Gas-Cooled Reactors. The high temperature gas-cooled reactor (HTGR) uses graphite as moderator, but has an unusual type of fuel FL (43). As produced by General Atomics of San Diego, California, the highly enriched (93%) fuel consists of coated spherical particles of diameter ~ 1 mm. As shown in Fig. 2, the kernel is a sphere of uranium dioxide, uranium carbide, or mixtures of these with silicon or aluminum. Kernels are prepared by a series of chemical processes and heat treated. Several thin coatings are applied, consisting of pyrolytic carbon or silicon carbide, or a combination of the two. These layers prevent fission products from escaping from the kernel, even when the temperature is as high as 1000°C. Solid fuel rods are fabricated from the coated particles and a carbon binder, and inserted into holes in large hexagonal graphite blocks. The prisms also have holes for passage of coolant. Stacks of blocks form the large core, measuring several meters in each direction. The core is located within a large prestressed concrete reactor vessel.

The coolant for the HTGR is helium. The helium is not corrosive; has good heat properties, with a specific heat that is much greater than that of CO_2; does not condense and can operate at any temperature; has a negligible neutron absorption cross section; and can be used in a direct cycle, driving a gas turbine with high efficiency.

The highest power of a reactor of the HTGR type was 330 MWe in Fort St. Vrain, Colorado. The reactor, started in 1979, had many technical problems, including helium leaks, and did not perform up to expectations. It was shut down in 1989.

7.4. Chernobyl. The best-known graphite-moderated reactor is the infamous Chernobyl-4, in Ukraine. It suffered a devastating accident in 1986 that spread radioactivity over a wide area of Europe.

The 950-MWe RBMK Chernobyl-4 reactor had graphite for the moderator and slightly enriched (2%) uranium as oxide canned in a zirconium–niobium alloy for fuel. The bundles of fuel rods were inside 8.6-cm diameter pressure tubes in which light water was brought to boiling. The core was cylindrical, 7 m high and 12 m wide, pierced by 1661 vertical fuel channels and 222 control and safety channels. An overhead refueling machine allowed fuel insertion and removal during operation.

Chernobyl-4 was completely destroyed in a violent explosion in 1986. The roof of the reactor building blew off and the graphite caught on fire and released radioactive fuel into the atmosphere. A number of workers were killed and the public was exposed to radiation. The accident was caused by a combination of reactor design features and operational errors: performance of an experiment that bypassed the safety equipment, without evaluation of hazardous consequences; removal of all safety rods during the course of the experiment in order to raise the reactor power; inadequate speed of control by the neutron-absorbing rods; and an inherently unsafe design, having a positive temperature coefficient of reactivity. It is generally believed that the consequences of the Chernobyl accident would have been far less serious if there had been a containment of the type used in Western reactors, rather than simple confinement by a conventional building (44).

Other water-cooled graphite reactors are still in operation in the former USSR. Some changes improving the prospects of safe operation have been made.

7.5. The Hanford N Reactor. The Hanford N reactor was built in 1964 for purposes of plutonium production during the Cold War. It used graphite as moderator, pierced by $>$ 1000 Zircaloy-2 tubes. These pressure tubes contained slightly enriched uranium fuel cooled by high temperature light water. The reactor also provided 800 MWe to the Washington Public Power Supply System. This reactor was shut down in 1992 because of age and concern for safety. The similarity to the Chernobyl-type reactors played a role in the decision.

8. Pressurized Water Reactors

The development of the pressurized light water reactor (PWR) involved studies of many types of reactors, including a gas-cooled power reactor, a fast breeder reactor, an aircraft propulsion reactor, and a high flux reactor for radiation testing. This last, called the Materials Testing Reactor (MTR), was authorized by the AEC in 1948 to test structural materials and fuels under high radiation conditions. It was built and put into operation in 1952 at the National Reactor Testing Station (NRTS) (now the Idaho National Laboratory) in Idaho, through the cooperation of Argonne National Laboratory in Illinois and Oak Ridge National Laboratory in Tennessee. Fuel plates were sandwiches of aluminum and uranium–aluminum metal alloy. Water served as moderator and coolant. The reactor had a beryllium reflector to enhance thermal neutron flux.

Another reactor that was approved for development was a land-based prototype submarine propulsion reactor. Westinghouse Electric Corp. designed this pressurized water reactor, using data collected by Argonne. Built at NRTS, the reactor used enriched uranium, the metal fuel in the form of plates. A similar reactor was installed in the submarine *Nautilus*.

The experience and capability of the Westinghouse Bettis Laboratory were then applied to designing and constructing the first full-scale commercial power reactor, the 60-MWe Shippingport, Pennsylvania reactor of Duquesne Light Company. The core of the Shippingport reactor (27,28) was composed of two types of fuel. There were 32 "seed" assemblies of highly enriched (90%) uranium alloyed with zirconium and clad with Zircaloy-2 (a 98.3% Zr alloy having 1.45% Sn and 0.05% Ni), in the form of plates 3.175 mm (1/8 in.) thick. There were 113 blanket assemblies, each of 120 fuel rods composed of natural uranium as pellets in Zircaloy-2 tubes. The seed region was in the shape of a square ring, with blanket fuel both inside and outside the ring. Neither type of fuel could sustain a chain reaction by itself, the seed because of excessive neutron leakage, the blanket because of the low uranium-235 content. Together these formed a critical system, and about the same amount of power was produced by each type of fuel. The use of Zircaloy tubes filled with uranium dioxide pellets has become standard for the industry.

Modern PWRs have power levels far > 60 MWe, are considerably larger, and are much more complex. Figure 3 shows a schematic diagram of the PWR vessel, heat exchanger, turbine generator, and other equipment. Also shown is a schematic of the containment, a large concrete and steel structure capable of withstanding a significant excess pressure from accidental release of hot steam from the reactor vessel.

The key feature of the pressurized water reactor is that the reactor vessel is maintained above the saturation pressure for water and thus the coolant-moderator does not boil. At a vessel pressure of 15.5 MPa (2250 psia), high water temperatures averaging $> 300°C$ can be achieved, leading to acceptable thermal efficiencies of ~ 0.33.

About one-half of the world's nuclear power plants are from Westinghouse Electric Corporation or its licensees. One Westinghouse PWR design is the four-loop Model 412 (45). To maintain the pressure on the coolant-moderator in the reactor vessel at the required 15.5 MPa (2250 psia), an auxiliary device called a pressurizer is used. For the Model 412, the pressurizer is a container with a length of 16.1 m and a diameter of 2.3 m that is connected directly to the reactor vessel. The pressurizer has electrical heaters that can heat the water to raise the pressure, and a spray head that injects cold water to lower the pressure.

The steel reactor vessel must be quite thick, eg, 203 mm (8 in.), to withstand the pressure. Figure 4 shows a cutaway view of the pressure vessel, showing the relationship of fuel, control rods, and coolant passages; Table 2 gives the corresponding data of the Westinghouse Model 412; the fuel assembly, consisting of an array of 264 fuel rods and 25 vacant spaces is shown in Fig. 5. Typically uranium of three different enrichments is initially loaded into the reactor. The configuration shown in Fig. 6, where the highest enrichment is placed on the outside and a checkerboard arrangement of two other enrichments is put in the center, favors uniform power and burn-up.

The Model 412 PWR uses several control mechanisms. The first is the control cluster, consisting of a set of 25 hafnium metal rods connected by a spider and inserted in the vacant spaces of 53 of the fuel assemblies (see Fig. 6). The clusters can be moved up and down, or released to shut down the reactor quickly. The rods are also used to (1) provide positive reactivity for the startup of the reactor from cold conditions, (2) make adjustments in power that fit the load demand on the system, (3) help shape the core power distribution to assure favorable fuel consumption and avoid hot spots on fuel cladding, and (4) compensate for the production and consumption of the strongly neutron-absorbing fission product xenon-135. Other PWRs use an alloy of cadmium, indium, and silver, all strong neutron absorbers, as control material.

The second control mechanism is the soluble reactor poison boric acid [10043-35-3], H_3BO_3. Natural boron contains 20% boron-10 [14798-12-0], ^{10}B, which has a thermal neutron cross section of $\sim 4.0 \times 10^{-25}$ m^2 (4000 barns). As fuel is consumed and fission products build up during a year or so of operation, the concentration of boron is adjusted by dilution. Starting from an initial value of $\sim$ 2000 ppm, the boron concentration goes to near zero at the end of the cycle.

The third control is by use of a fixed burnable poison. This consists of rods containing a mixture of aluminum oxide and boron carbide, included in the initial fuel loading using the vacant spaces in some of the fuel assemblies that do not have control clusters. The burnable poison is consumed during operation, causing a reactivity increase that helps counteract the drop owing to fuel consumption. It also reduces the need for excessive initial soluble boron. Other reactors use gadolinium as burnable poison, sometimes mixed with the fuel.

In the startup of a reactor, it is necessary to have a source of neutrons other than those from fission. Otherwise, it might be possible for the critical condition to be reached without any visual or audible signal. Two types of sources are used to supply neutrons. The first, applicable when fuel is fresh, is californium-252 [13981-17-4], ^{252}Cf, which undergoes fission spontaneously, emitting on average three neutrons, and has a half-life of 2.6 years. The second, which is effective during operation, is a capsule of antimony and beryllium. Antimony-123 [14119-16-5], ^{123}Sb, is continually made radioactive by neutron absorption. The product antimony-124 [14683-10-4], ^{124}Sb, is radioactive, has half-life 60 days, and its gamma rays cause beryllium-9, ^{9}Be, to emit neutrons.

An engineered safety system is provided to protect against hazard from a loss-of-coolant accident (LOCA). If a coolant pipe should break, causing a drop in pressure in the vessel, an emergency core cooling system (ECCS) begins supplying auxiliary water from storage tanks to continue cooling the core. A water spray system in the containment helps condense steam, and cooling fans go into operation.

A PWR can operate steadily for periods of 1–2 years without refueling. Uranium-235 is consumed through neutron irradiation; uranium-238 is converted into plutonium-239 and higher mass isotopes. The usual measure of fuel burnup is the specific thermal energy release. A typical figure for PWR fuel is 33,000 MWd/t. Spent fuel contains a variety of radionuclides (5):

Isotope	Percent, %
^{235}U	0.81
^{236}U	0.51
^{238}U	94.3
^{239}Pu	0.52
^{240}Pu	0.21
^{241}Pu	0.10
^{242}Pu	0.05
fission products	3.5

The original fuel contained 3.3 wt% uranium-235 and 96.7 wt% uranium-238.

9. Boiling Water Reactors

Water is also used as moderator-coolant in the BWR. The principal distinguishing feature from the PWR is that in the BWR steam is produced in the core and delivered directly to a steam turbine for the generation of electricity, eliminating the need for a heat exchanger. Benefits are that the system is simpler than a PWR and the capital equipment cost is lower. The flow diagram of Fig. 7 shows a direct cycle BWR. A dual-cycle BWR is one where some heated water from the core goes through a heat exchanger.

Initial studies for the BWR were made at Argonne National Laboratory in the early 1950s (46). The first experiments used electrical heating of metal plates and tubes immersed in a water bath. Stable boiling, a high velocity of steam, and a very short time necessary for steam bubbles to form were observed. These results encouraged planning for a boiling water experiment involving fuel.

A series of tests were performed at the AECs National Reactor Testing Station in Idaho, starting in 1953. The reactor was situated outdoors, and was operated remotely. The core of the first version had fuel assemblies of aluminum and enriched uranium plates of the Materials Testing Reactor (MTR) type, installed in a water tank. One of the five control rods could be ejected downward and out of the core by spring action upon interruption of a magnet current. The other four rods could be dropped into the core to terminate an excursion in power. Among the findings of this, the BORAX program, were that the reactor had a high degree of inherent safety; operation was more stable than had been expected, although it was possible to induce oscillations in power; high heat-transfer rates from fuel surfaces to steam were possible; the reactor operated well at either atmospheric or elevated pressures; and there was little radioactivity in the steam, although two radionuclides were formed by neutron absorption in oxygen, that is, oxygen-19 and nitrogen-16 having half-lives of 29 and 7 s, respectively. An excellent description of the BORAX equipment, experimental results, and boiling water reactors in general is available (25).

The next facility in the evolution of the BWR was the experimental boiling water reactor (EBWR), built at Argonne National Laboratory (46). This reactor went critical in 1956 and eventually reached a power of 100 MWt. Operating at a power twice that of BORAX, it was more stable. Another BWR having still higher pressure, the Vallecitos boiling water reactor (VBWR), was operated in California by General Electric Company. A succession of central station power reactors appeared: 200-MWe Dresden (Illinois) in 1960, 63-MWe Humboldt Bay (California) in 1963, and 610-MWe Oyster Creek (New Jersey) in 1969.

Figure 8 shows a cutaway of the reactor vessel of the General Electric Company's model BWR/6 (47). Table 3 lists numerical data about this reactor.

In the operation of the BWR, water is caused to flow past the fuel rods by a system consisting of two recirculation pumps (only one is shown in Fig. 7), and a number of internal jet pumps. The jet pump has no moving parts, but consists of a high pressure (driving) stream that converges as it is injected along the axis of a low pressure (suction) stream. The two streams merge and are allowed to expand in a diffuser section of the pump. In simplest terms, the high pressure stream drags along the low pressure one. The operating efficiency of a jet pump is the ratio of energy gain of suction flow to energy loss of driving flow.

As the water rises in the reactor core, the steam void volume fraction increases. To remove water droplets from steam, two devices are installed in the upper part of the BWR reactor vessel. The first is an array of steam separators, which consist of tubes having vanes inside giving the steam–water mixture a spinning motion. This allows centrifugal force to carry the water to the walls, where it flows back down. The second is the steam dryer, a bank of vanes through which the stream passes. Any remaining water condenses on the vanes, flows into a trough, and goes down toward the core.

The reactor is equipped with a set of cross-shaped control rods. These are inserted into the core from the bottom. The position can be controlled automatically or manually for both start-up and power adjustments during operation. The reactor is started from cold conditions by moving the control rods to change the reactivity and by varying the flow of water in the recirculation loops. The role of the rods in relation to the negative steam void coefficient of reactivity is as follows. If the flow is increased, steam is swept out more rapidly, reducing the void fraction and thus giving a positive reactivity, which causes a power increase. This creates more steam and a negative reactivity. The reactor stabilizes at a higher power level.

For safety in case of accident, an unusual containment system is employed. Early models of BWRs used large dry containments, eg, those used for many PWRs. Modern versions eg, General Electric's BWR/6 Mark III, instead make use of a pressure-suppression system (Fig. 9). This is designed to accommodate a loss-of-coolant accident (LOCA), eg, a pipe break, which releases steam and water from the reactor vessel and tends to build up pressure in the containment drywell. The pressure is relieved by allowing steam, water, and air to flow through a vent into a large pool of water, where the steam condenses. Above the pool is the wetwell, a large volume that receives air from the drywell. The water in the pool also serves as a reservoir for makeup cooling of the core after a LOCA. Two features not shown in the simplified diagram are vacuum breakers to allow flow from the wetwell to the drywell if spray cooling is used to reduce drywell pressure, and a quencher that conducts steam from the reactor vessel directly into the suppression pool.

10. Heavy Water Reactors

A heavy water reactor (HWR) uses deuterium oxide, D_2O, also called heavy water, as moderator. There has been relatively little experience using commercial heavy water moderated power reactors in the United States. Early experimental reactors were operated at Argonne National Laboratory in the 1950s. Additionally, a prototype power reactor was built in the 1960s, sponsored by the Carolinas–Virginia Power Associates. It provided valuable experience about reactor design and operation for the organizations involved. Several heavy water production reactors were operated at the Savannah River Plant. The most successful application of heavy water reactors has been in Canada.

10.1. Savannah River Production Reactors. For the production of weapons plutonium and tritium, the U.S. Government operated heavy water reactors for $\sim$ 35 years at the Savannah River Plant (SRP) in South Carolina. Design, construction, and operation of the five reactors and related equipment were under the leadership of E. I. du Pont de Nemours & Co., Inc. Heavy water produced on-site was used as both moderator and coolant. Reactor designs were based in part on the early Argonne reactors. The moderator was contained in large (5 m) diameter stainless steel reactor vessels. Primary coolant passed through 600 vertical tubes in a hexagonal array, and then through a heat exchanger. The secondary loop was ordinary river water. The maximum power achieved in any reactor was 2915 MWt.

Many redundant safety features were provided at the SRP. These included a moderator dump tank, gadolinium nitrate solution as emergency absorber, continuously running diesel generators, and a 95×10^6 L (25×10^6 gal) elevated water tank for each reactor, for assurance of cooling.

Over the years, a variety of fuel types were employed. Originally, natural uranium slugs canned in aluminum were the source of plutonium, while lithium–aluminum alloy target rods provided control and a source of tritium. Later, to permit increased production of tritium, reactivity was recovered by the use of enriched uranium fuel, ranging from 5 to 93%.

Fuel assemblies became much more sophisticated, eventually consisting of concentric tubes made from an outer sheath, three fuel tubes, and an inner lithium-target tube, thus having four coolant channels. Locally developed extrusion techniques were used.

Among other isotopes produced at SRP were uranium-233 for breeder research, cobalt-60 [10198-40-0] for irradiators, plutonium-238 for spacecraft such as *Voyager* and lunar research power supplies, and californium-252 as a fast neutron source. The accomplishments of Du Pont at SRP are well chronicled (48).

When the Du Pont contract ended in 1989, Westinghouse took over. Plans for upgrading the existing reactors and constructing a new production reactor have been abandoned.

10.2. The CANDU Reactors. The Canadian deuterium uranium (CANDU) reactors are unique among power reactors in several respects. Heavy water is used as moderator; natural uranium having ^{235}U isotopic content of 0.72 wt% is used as fuel, rather than the typical 2–4 wt% ^{235}U for light water reactors; the heavy water coolant flows through pressure tubes passing through the moderator tank; and continuous refueling is performed.

There are several hundred pressure tubes, each containing bundles of 28 fuel rods, 50 cm long. The coolant is at a pressure of around 10 MPa (1450 psia) and the D_2O is at 310°C. Headers on each side of the vessel collect and return coolant from all the tubes. The 4-mm wall-thickness zirconium–4.5% niobium alloy pressure tubes are surrounded by heavy water moderator at much lower temperature and pressure. The reactor vessel, called a calandria, is a large cylinder 8.5 m in diameter and 6 m long, oriented horizontally. Because heavy water is expensive, costing around $300/kg, negligible leakage is mandatory. Refueling is done without shutting the reactor down, reducing outage times. Refueling machines are located at each end of the reactor vessel. A steam generator having light water in the secondary side supplies steam to the turbine. Figure 10 shows a cutaway view of the reactor building and its contents.

CANDU has a unique negative pressure containment that functions if an accident, eg, a cooling-water pipe break should occur in the reactor building, resulting in a release of steam, hot water, and radioactive material. The increased pressure is relieved into a vacuum building, maintained at nearly zero pressure. The building is 50 m in diameter and in height, having 1-meter thick walls and roof. Inside is a large emergency storage tank that provides a water spray to quench the hot vapor and wash out radioactivity.

The Canadian nuclear power program was developed by Atomic Energy of Canada, Ltd. (AECL) of Ottawa. The CANDU reactors provide a large fraction of the electricity of that region. Fifteen heavy water reactors are available, capable of producing 15,164 MW of electrical power. However, some of the reactors are idle and some are to be refurbished. Most of these are operated in the province of Ontario by Ontario Hydro. Pickering, Bruce, and Darlington are multiple reactor stations having eight, eight, and four reactors, respectively. The AECL has supplied heavy water reactors to several countries, including Argentina, China, South Korea, and Romania. The use of spent fuel from light water reactors as fuel for the CANDU is being explored.

Early research and development is described in a symposium proceedings (49). The status of the CANDU program as of 1975 is available (50) and a brief history may be found in a more recent publication of the American Nuclear Society (36).

A variant of the HWR is the Fugen reactor developed by Japan. This reactor is heavy water moderated, but light water cooled. It is fueled by mixed uranium–plutonium oxides.

11. Fast-Breeder Reactors

Breeding of nuclear fuel was recognized as having a potentially important impact on the availability of energy resources as soon as plutonium was discovered. The most likely nuclear reaction involved the absorption of a neutron in uranium-238 to form plutonium-239, a fissile nuclide with a half-life of ~24,000 years. Fission in plutonium-239 by fast neutrons gives rise to about three fast neutrons per absorption. Thus in a reactor using plutonium as fuel the chain reaction can be maintained and enough neutrons are left over to produce more fuel than is burned. At the same time, by consuming uranium-238 instead of merely burning uranium-235 as in converter reactors, the amount of uranium ore needed to produce a given energy is reduced by a factor as large as 50, thus extending the practical life of the uranium resource for thousands of years.

Full advantage of the neutron production by plutonium requires a fast reactor, in which neutrons remain at high energy. Cooling is provided by a liquid metal, eg, molten sodium or NaK, an alloy of sodium and potassium. The need for pressurization is avoided, but special care is required to prevent leaks that might result in a fire. A commonly used terminology is liquid-metal fast-breeder reactor (LMFBR).

An important parameter of any breeder is the breeding ratio (BR) defined as the ratio of the fissile atoms produced to the fissile atoms consumed and given by the simple relation

$$\mathrm{BR} = \eta - 1 - \ell$$

where η is the number of neutrons per absorption and ℓ is the number of neutrons lost by leakage and nonfuel absorption. Values of BR $\sim$ 1.2 are regarded as excellent. A related quantity is the breeding gain (BG) where

$$\mathrm{BG} = \mathrm{BR} - 1$$

In the evaluation of these parameters, the chain of plutonium isotopes produced and consumed must be taken into account. Successive neutron captures create plutonium-239, -240, -241, and -242. Isotopes having odd mass number are fissile, the others are not.

An extensive theoretical, experimental, and computational knowledge base for fast breeders has been developed. A compact review of key design concepts, analytic methods, and data are available (51). Two types of cooling systems for fast breeders have been employed. The first is the loop, in which the liquid metal is circulated by pump through the reactor vessel and an external heat exchanger. The other is the pot or pool, in which the heat exchanger and pump are in a tank with the reactor core. There are advantages and disadvantages of each arrangement.

There are two ways to locate fissile and fertile materials to achieve breeding. One is the homogeneous arrangement, in which all fissile fuel such as Pu is located in a core and the fertile material such as natural or depleted U is outside in a breeding blanket. The other is the heterogeneous arrangement, having concentric rings of fertile materials within a larger core. The second technique has improved breeding gain and safety.

Most fast reactors that use Na or NaK as coolant utilize an intermediate heat exchanger (IHX) that transfers heat from the radioactive core coolant to a nonradioactive liquid-metal coolant loop, which has the reactor's steam generator. This helps minimize the spread of contamination in the event of a leak or fire.

The first experimental breeder reactor (EBR-I), which was the first reactor to generate electricity on a practical basis, went into operation in 1951 at the National Reactor Testing Station in Idaho. After the first reactor was damaged by a power excursion, EBR-II was put into operation in 1961 (52). It operated very well over a period of 30 years.

As a part of the power demonstration program of the AEC in the 1950s, the Enrico Fermi fast breeder reactor (Fermi-1) was built near Detroit by a consortium of companies led by Detroit Edison. Fermi-1 used enriched uranium as fuel and sodium as coolant, and produced 61 MWe. It suffered a partial fuel melting accident in 1966 as the result of a blockage of core coolant flow by a metal plate. The reactor was repaired but shut down permanently in November 1972 because of lack of funding. Valuable experience was gained from its operation, however (53).

The United States continued fast-breeder reactor research and development with the building of the fast flux test facility (FFTF) at Hanford and the SEFOR reactor in Arkansas (54). The next planned step was to build a prototype power reactor, the Clinch River fast-breeder plant (CRFBP), which was to be located near Oak Ridge, Tennessee.

Prospects in the United States for deploying breeders on a large scale were bright when it was believed that rich uranium ore would be quickly exhausted as use of nuclear power expanded. The expected demand for uranium was not realized, however. Moreover, the utilization of breeders requires reprocessing. In 1979, a ban was placed on reprocessing in the United States. A dampening effect on development of that part of the fuel cycle for breeder reactors resulted. The CRFBP was canceled and France and Japan became leaders in breeder development.

One of the most advanced versions of a LMFBR was the French SuperPhénix, located at Creys-Malville (55). Partners in development were Electricité de France and firms of Italy, Germany, Belgium, the Netherlands, and the United Kingdom. It was a pool-type system using sodium coolant and a small core surrounded by a breeding blanket. The reactor was shut down permanently in 1998 as a political decision. A few of its pertinent features are listed in Table 4 (56). SuperPhénix was originally expected to become a research and demonstration facility having an emphasis on burning plutonium and possibly other actinides.

The MONJU fast-breeder reactor is located on the northern coast of Japan. It is the result of a long-term research and development program led by Hitachi, Ltd. and integrates the work of several companies. Japan is totally dependent on foreign sources of uranium and seeks to make effective use of resources through a plutonium recycle. For earthquake protection and ease of maintenance, the loop-type arrangement was chosen. Specially designed mechanical snubbers were used to support piping. The MONJU core is very compact. It is 93 cm high, 180 cm in diameter, and has only 2340-L volume. It has four concentric regions using a triangular (hexagonal) arrangement (Fig. 11). The inner core has 108 assemblies and 19 control rods, the outer core has 90 assemblies, the blanket has 172 assemblies, and the shield has 324 assemblies. This reactor uses mixed uranium and plutonium oxides for fuel and uranium metal for blanket. The electrical output is 280 MW.

For additional information about nuclear fuel, see the article by Daniel B. Bullen titled Nuclear Fuel Reserves.

12. Other Reactors

12.1. The Natural Reactor. Some 2 billion years ago, uranium had a much higher ($\sim$ 3%) fraction of ^{235}U than that of modern times (0.7%). There is a difference in half-lives of the two principal uranium isotopes, ^{235}U having a half-life of 7.08×10^8 years and ^{238}U 4.43×10^9 years. A natural reactor existed, long before the dinosaurs were extinct and before humans appeared on the earth, in the African state of Gabon, near Oklo. Conditions were favorable for a neutron chain reaction involving only uranium and water. Evidence that this process continued intermittently over thousands of years is provided by concentration measurements of fission products and plutonium isotopes. Useful information about retention or migration of radioactive wastes can be gleaned from studies of this natural reactor and its products (6).

12.2. Homogeneous Aqueous Reactors. As a part of the research on neutron multiplication at Los Alamos in the 1940s, a small low power reactor was built using a solution of uranium salt. Uranyl nitrate [36478-76-9], $UO_2(NO_4)_2$, dissolved in ordinary water, resulted in a homogeneous reactor, having uniformly distributed fuel. This "water boiler" reactor was spherical. The ^{235}U mass was quite low, $\sim$ 1 kg.

The Los Alamos water boiler served as a prototype for the first university research and training reactor, started in September 1953 at North Carolina State College. That cylindrical reactor core used uranyl sulfate [1314-64-3], UO_2SO_4, and cooling water tubes wound inside the stainless steel container. A thick graphite reflector surrounded the core.

The homogeneous aqueous reactor was studied extensively at Oak Ridge in the 1950s. The objective was to develop a circulating-fuel power reactor that would be easy to refuel, have no temperature limitations on materials, and allow continuous extraction of fission products. The homogeneous reactor experiment-1 (HRE-1) had a core having a 45.72-cm (18 in.) diameter stainless steel spherical vessel containing highly enriched uranium as uranyl sulfate in light water. Because radiation in the water caused dissociation, a flame recombiner was used. The remaining fission gases, xenon, and krypton, were held for decay. Operation at powers up to 1.6 MW in the period 1952–1954 was quite satisfactory (26). The large negative temperature coefficient of reactivity provided a high degree of stability. Maintenance of the highly radioactive system was performed using long-handled tools and temporary shielding.

The homogeneous reactor experiment-2 (HRE-2) was tested as a power breeder in the late 1950s. The core contained highly enriched uranyl sulfate in heavy water and the reflector contained a slurry of thorium oxide [1314-20-1], ThO_2, in D_2O. The reactor thus produced fissile uranium-233 by absorption of neutrons in thorium-232 [7440-29-1], the essentially stable single isotope of thorium. Local deposits of uranium caused reactivity excursions and intense sources of heat that melted holes in the container (7), and the project was terminated.

12.3. Aircraft Reactors. As early as World War II, the U.S. Army Air Force considered the use of a nuclear reactor for the propulsion of aircraft (57–59). In 1946, the program Nuclear Energy for the Propulsion of Aircraft (NEPA) was set up at Oak Ridge, under Fairchild Engine and Airplane Corporation. Basic theoretical and experimental studies were carried out. The emphasis was on materials. A high temperature reactor was built and operated successfully. It had beryllium oxide [1304-56-9], BeO, moderator and nickel tubes, through which ran a molten salt fuel consisting of fluorides of Na, Be, and U.

In 1950, a new program, Aircraft Nuclear Propulsion (ANP), was begun by General Electric Company, in Cincinnati, Ohio, and the National Reactor Testing Station in Idaho. A reactor was built to test the heating of air for a turbojet. The first heat-transfer reactor experiment (HTRE-1) consisted of an aluminum water-filled tank through which many tubes passed. Metal fuel elements were placed in the tubes and air pumped through to a ground-based turbojet engine. Operation for 150 h led to energy of 5000 MWh. In a later version, beryllium oxide moderator was used to achieve higher temperature operation.

Another program under Pratt & Whitney in Hartford, Connecticut, was designed to test the indirect cycle. It involved basic studies leading to the design of a liquid-metal cooled reactor, having a heat exchanger to air for a turbojet. The advantage claimed was high temperature operation without contamination of the air.

The launching of the Russian satellite *Sputnik* in 1957 galvanized the United States, and there were recommendations for expansion of the aircraft program. As late as 1959, the Air Force was optimistic about the use of nuclear reactors for aircraft (59). However, the projects were canceled in 1961. The technology of jet aircraft had advanced, reducing the need for a long range, but slow nuclear bomber. There was the unsolved problem of adequately shielding the crew of the vehicle without excessive weight, and concerns about radioactive contamination in case of an airplane accident (59). Subsequently, the programs were reoriented to materials studies related to advanced high temperature reactors for space (58).

12.4. Naval Reactors. The possibility of using nuclear energy to propel ocean vessels was discussed as soon as fission and the chain reaction were discovered. After the end of World War II serious consideration was given to nuclear-propelled submarines. These would not depend on air from the surface, would be extremely silent, could travel at high speed, and could remain submerged for long distances because of the large energy yield from nuclear fuel. The translation of such concepts into a working propulsion system was effected quickly. Then-Captain Hyman G. Rickover was selected in 1946 as one of the U.S. naval officers to study nuclear technology at Oak Ridge. He took charge of the group, collected all information that might be relevant to the goal of a nuclear submarine, and promoted a program of engineering development (60). There were two approaches toward the achievement of a nuclear submarine. One was a water-moderated and -cooled pressurized reactor; the other was a liquid-metal cooled intermediate neutron energy reactor. A land-based prototype submarine power plant called Mark I was built and tested at the National Reactor Testing Station. Argonne National Laboratory provided scientific data and Bettis Laboratory of Westinghouse Electric Corp. supplied engineering expertise.

A great deal of technical information was needed on the behavior of materials under severe conditions of temperature and radiation. Stainless steel was selected for structures, but for the fuel cladding zirconium, a rare metal, was chosen because of its very low thermal neutron absorption cross-section and its resistance to corrosion by hot water. Methods were developed for extraction of zirconium from ore, of removal of the strong accompanying absorber hafnium, and of fabrication into desired shapes. Eventually, hafnium was found useful for control rods in place of an alloy of silver, cadmium, and indium. Special seals were needed to prevent leakage of water and radioactivity. Extensive radiation-shielding studies assured safety of the crew in the cramped quarters of a submarine. Voluminous technology handbooks have been written on these and other findings (61–63).

Following a successful test of the Mark I, construction of the first nuclear submarine *N.S. Nautilus* was begun in 1953, *Nautilus* made a trip of $>$ 114,824 km (62,000 nautical mi) submerged, from the United States to the British Isles in 1955, and served as a model for the fleet of $>$ 100 submarines of the U.S. Navy (64). Several nuclear-powered aircraft carriers and guided missile cruisers were also built. The carrier *Enterprise*, launched in 1961, is propelled by eight PWRs. The ship, having a length of 342 m, carries 75 aircraft and $>$ 5000 personnel (65).

12.5. Maritime Reactors. Nuclear power has had limited use for propulsion of merchant ships, largely because of economic reasons, although public reaction has also played a role (66). The construction of a nuclear-powered merchant ship was proposed in the 1950s to demonstrate the U.S. interests in peace. The AEC and the Maritime Administration sponsored the design and construction of the *N.S. Savannah*. Babcock & Wilcox supplied the reactor and New York Shipbuilding Corporation constructed the ship. The cargo–passenger vessel was almost 183 m long and was powered by an 80-MWt pressurized water reactor. The core was fueled with 4.4% enriched uranium dioxide in stainless steel tubes, operating at 12.1 Pa (1750 psi). Steam was supplied to a turbine that drove the ship's propeller. Seawater was used to condense steam. Sustained speeds of the ship were 21 knots.

Launched in 1959, *N.S. Savannah* operated very well. Starting in 1962, it made a goodwill voyage around the world. It was able to travel a distance of several times the earth's circumference on one fuel loading. However, the ship was not competitive economically with oil-powered merchant ships. The shielding was quite adequate, so that the reactor was safe (67). Nonetheless the vessel was opposed by antinuclear groups and the *N.S. Savannah* was eventually retired and put on display in Charleston, South Carolina. In 1994, the ship was transferred to Norfolk, Virginia. Radioactive areas are to be cleaned up and the ship preserved as a National Historic Landmark (68).

The Russian icebreaker *Lenin*, launched in 1959, had three 90 MWt PWRs, one of which was a spare. It operated for many years in the Arctic Ocean. The most recent icebreaker is Yamal, launched in 1992 (69). It is used for tourist trips to the Arctic, including a visit to the North Pole arriving January 1, 2000. Since it requires cold seawater, it cannot go to Antarctica.

12.6. Package Power Reactors. Several small, compact power reactor plants were developed during the period 1957–1962 by the U.S. Army for use in remote locations. Designed by Alco Products, Inc., the PWRs produced electrical power of $\sim$ 1 MW along with space heat for military bases. The first reactor, SM-1, was operated at Fort Belvoir, Virginia. Others were located in Wyoming, Greenland, Alaska, and Antarctica. The fuel consisted of highly enriched uranium as the dioxide, dispersed in stainless steel as plates or rods. Details are available in the book by Loftness (7).

12.7. Space Reactors. Two quite different applications of reactors in space have been studied: one for electrical power of a spacecraft mission, and the other for propulsion of spacecraft. Both applications are for long missions where solar power is inadequate or chemical propulsion is impractical (70,71).

The AEC sponsored research in the program known as Systems for Nuclear Auxiliary Power (SNAP) as early as the 1950s. Most of the systems developed involved the radioisotope plutonium-238 as a heat source for a thermoelectric generator. Such electrical supplies permitted radio transmission to earth from spacecraft, eg, *Pioneer* and *Voyager*.

Several actual reactors have been built to provide the heat that can be converted into electricity. One of them, SNAP-10A, was flown in space in 1965. It was placed in 1300-km radius orbit by the launch vehicle Agena. The power unit remained subcritical until the orbit was reached, at which time automatic control took over to start the reactor. Its core was composed of stainless steel clad rods containing a homogeneous mixture of zirconium hydride [7704-99-6], ZrH_2, and uranium-235, serving as both fuel and moderator. Liquid-metal coolant NaK circulated by an electromagnetic pump passed between the rods and to a Si–Ge thermoelectric generator producing $\sim$ 500 W of power. It operated successfully for 43 days until a nonreactor-related failure occurred. A twin reactor on the ground operated at full power for 10,000 h.

Future space missions of long duration and long distance, eg, flights to Mars and back, could conceivably use a solid-core nuclear rocket. The key measure of effectiveness of a rocket for propulsion is the specific impulse, defined as the ratio of thrust to mass flow rate of propellant. Whereas a nuclear reactor cannot produce a gas temperature as high as a chemical fuel can, a reactor can use the light element hydrogen as coolant–propellant, instead of the heavier products of combustion (72).

Reactors for direct propulsion of spacecraft were built and tested in the ROVER project from 1959–1973. These used uranium carbide as fuel and graphite as moderator. Liquid hydrogen served as coolant. The hydrogen was volatilized and exhausted from a nozzle as a propellant (Fig. 12). The most successful reactor was the nuclear engine for rocket vehicle application (NERVA), which operated at a very high (4000 MW) power for a time of 12 min. The ROVER project was canceled in 1973.

NASAs plans for use of a nuclear rocket for manned voyage to Mars were resurrected in the 1990s (73). However, with an administration change, the plans were abandoned.

Ion propulsion of spacecraft with fission power is under study for future long-term distant missions. A reactor named Safe Affordable Fission Engine (SAFE) would provide 400 kWt over 10 years, supplying electrical power for ion propulsion (74). Special features include (a) structure niobium with one percent zirconium, (b) fuel highly enriched uranium as nitride, (c) coolant a mixture of He (72%) and Xe (28%) (d) heatpipe (75) for transfer of fission heat, and (e) Brayton thermal cycle (76).

In 2003, a mission called Prometheus was planned for exploration of those Jupiter's moons suspected to have oceans. It would use a nuclear reactor to provide electricity for an ion drive. The vehicle called Jupiter Icy Moons Orbiter (JIMO) was designed and its ion engine given some preliminary testing. As visualized, the reactor would supply power to a microwave that produced xenon ions (77). A rectangular grid at high potential would serve to accelerate the ions as propellant with a 6000 s specific impulse. According to Zubrin (78) the plan did not take advantage of planetary gravity assist (the slingshot effect), which caused the cost of the project to be excessive. No funds were provided for JIMO in 2005 and the plan was put on hold.

Also the national goal of returning to the Moon and going on to Mars was reestablished. The project visualized involved a slow heavy cargo spacecraft followed later by a fast light manned vehicle. For transportation to Mars a radically new nuclear space idea was advanced. A spacecraft was proposed that used gamma ray heating of propellant from the annihilation of matter (electrons) and antimatter (positrons). It was recognized that producing enough positrons and storing them were major design problems (79).

12.8. Research and Training Reactors. Research reactors generally fall in one of three categories: an experimental reactor to test a concept, a high flux reactor dedicated to basic research, or a reactor used primarily for educational purposes. Reactors at universities or laboratories may be used for purposes, such as, production of radioisotopes, the study of radiation effects, neutron activation analysis, measurements of reactor properties and behavior, and the teaching of nuclear engineering students. Some of these reactors have been in operation for 30 years or more without incident. Many university reactors have been shut down for economic reasons. The International Atomic Energy Agency (IAEA) has an extensive database of research reactors throughout the world, including data on features of individual reactors (80). The American Nuclear Society's position statement lists a number of references on the subject (81).

The determination of critical size or mass of nuclear fuel is important for safety reasons. In the design of the atom bombs at Los Alamos, it was crucial to know the critical mass, ie, that amount of highly enriched uranium or plutonium that would permit a chain reaction. A variety of assemblies were constructed. For example, a bare ^{235}U metal sphere was found to have a critical mass of $\sim$ 50 kg, whereas a natural uranium reflected ^{235}U sphere had a critical mass of only 16 kg. The first critical experiments at Los Alamos are described by Paxton (82).

The reactor Lady Godiva was constructed at Los Alamos for the study of dynamic behavior of a supercritical assembly. This structure was an unreflected, essentially spherical metal of uranium enriched to 93% and having a density of $\sim$ 18 g/cm^3. The reactor had no moderator, thus the neutrons causing fission were fast, $\sim$ 1 MeV. Cooling was only by conduction through the metal and convection in air at the surface. The sudden application of a positive reactivity to such a reactor causes it to rise in power rapidly, and the strong negative temperature feedback effect causes the power to peak and drop back, giving rise to a pulse or burst. The neutrons that accompany the power were used to irradiate detecting equipment related to weapons. Although large ($\sim 10^{17}$) bursts of neutrons are created and the instantaneous power levels are very high (thousands of MW), the time span of the pulse is very short (a few μs) and the heat energy (a few J) is modest.

In the early 1950s, the Argonaut research and training reactor was designed and built by Argonne National Laboratory. It was subsequently adopted by several U.S. universities. Its initial purpose was for nuclear studies conducted by scientists and engineers from many countries as a part of the Atoms for Peace program. The reactor consisted of a ring of plate-type fuel assemblies having graphite fillers interspersed among them. Water within the fuel boxes provided self-limiting safety. Graphite served as internal and external reflector. A peak power of 10 kW for short times was possible.

One of the early popular low power research and training reactors was the AGN-201, supplied by Aerojet General Nuclear. This is a homogeneous solid fuel reactor, consisting of a mixture of polyethylene and uranium at 20% enrichment in ^{235}U. The core ^{235}U loading is around 0.7 kg and the core volume is 12 L. The reflector is graphite and the core has no cooling. Power is limited to 5 W, giving a thermal neutron flux of around $10^8/\text{cm}^2{\cdot}\text{s}$, sufficient for certain experiments. An example AGN-201 is described by Chiovaro et al. (83).

A number of pool reactors, also called swimming pool reactors, have been built at educational institutions and research laboratories. The cores in these reactors are located at the bottom of a large pool of water, 6 m deep, suspended from a bridge. The water serves as moderator, coolant, and shield. The highest power U.S. university reactor is the University of Missouri-Columbia Research Reactor (MURR). First operated in 1966, it has a power of 10 MWt with pool temperature 136°F. Cooling is by natural convection. The reactor operates almost continuously, using a variety of beam tubes, for many research purposes (84).

A variant on the pool reactor is the tank type, in which a limited volume of water surrounds the core. The TRIGA reactor, marketed throughout the world by General Atomic of San Diego, California, is an example. The reactor core contains many fuel rods immersed in water, and has a graphite reflector. The fuel is a mixture of zirconium hydride and 20% enriched uranium, clad with stainless steel. Control rods are boron carbide. TRIGA Mark II is capable of steady operation at 250 kW or pulsing to 250 MW for a brief time. The steady thermal flux is approximately $10^{13}/\text{cm}^2{\cdot}\text{s}$. The core fuel loading is 2.7 kg in a volume of 63 L. Included in experimental facilities are a rotating specimen rack and various beam tubes. The history of TRIGA reactors is given in an article by Fouquet, Razvi, and Whittemore (85).

The Slowpoke reactor, supplied by Atomic Energy of Canada, Ltd., is installed in several Canadian universities. An example Slowpoke is that at the University of Toronto (86). It is a light water moderated reactor having 93% enriched uranium. Various features make it inherently safe and give it its name. The core is composed of aluminum-clad fuel rods 0.473 cm in diameter and 22 cm long. Its reflector is beryllium and it has a single cadmium control rod. Cooling is by natural convection. The power level is 20 kW, giving a thermal flux of $10^{12}/\text{cm}^2{\cdot}\text{s}$. A larger version, the Slowpoke demonstration reactor (SDR), is designed for district heating. It provides thermal power of 2 MW.

12.9. Advanced Power Reactors. Most of the U.S. nuclear reactors were built in the 20-years period 1964–1984. Many are approaching their design life. Increased attention is being given to the aging of components, using preventive maintenance, and replacement of parts and assemblies to extend the life of facilities (87). Almost all reactors are applying to the NRC for license extension up to 20 years. At the same time, the U.S. nuclear industry, in cooperation with the Department of Energy, is developing several advanced reactor designs to supplant and supplement the existing reactors (88). One impetus for the R&D is the increasing concern about global warming resulting in part from fossil-fueled power plants. Another is the fear that the cost of natural gas used in turbine plants will become prohibitive. Finally, the public has come to realize that nuclear plants are being operated safely and efficiently.

Advantage is being taken of the experience and knowledge gained in > 40 years of light water reactor operation, and features that provide inherent passive safety are being included wherever feasible. By incorporating simplicity, economy, and improved safety, the advanced reactors are being designed to be attractive to the public, to utility management, and to the financial community.

Advanced reactors can be classified according to the features that distinguish them, the program plans for achievement of operation, the status of NRC design approval, and the anticipated date of deployment. A designation according to "generation" has become popular, as outlined in the following sections.

Generation I. Prototype reactors of the 1950s and 1960s are in Generation I. Examples are Shippingport, Dresden, EBR II, and Fermi I, which were the culmination of extensive testing.

Generation II. Generation II reactors are the reactors currently in commercial use, deployed in the 1970s and 1980s. Examples are the Westinghouse and General Electric reactors discussed in Sections 7 and 8, and the System 80 reactor originally designed by ABB Combustion Engineering. Others are CANDUs in Canada and AGRs in Britain.

Generation III. Reactor designs that are ready for deployment or have already been built abroad are in Generation III. They are denoted as "evolutionary" in that they have definite improvements over earlier models.

AP600. The Westinghouse AP600 is a pressurized-water reactor of 600-MWe capacity, of the passive safety type (89). The system has far fewer pumps, pipes, valves, and ducts than current designs. It depends greatly on passive natural processes such as gravity, natural circulation, convection, evaporation, and condensation. As sketched in Fig. 13, it has an emergency core cooling system that does not require pumps or electric power. Two large water tanks are located above the reactor. If a loss-of-coolant accident (LOCA) occurs when the reactor is still under pressure, water is driven by pressurized nitrogen into the core. If the reactor pressure is lost, gravity produces water flow from a tank at atmospheric pressure. In the event steam generators are not operable, natural circulation to the large water tank above the reactor removes decay heat. The metal containment is kept cool to condense vapor released in a LOCA by air drawn through a chimney at the top and by a gravity-fed water spray.

Lower power ratings of reactors provide greater flexibility for a utility to add power generation to a system. The AP600 uses prefabricated modules to shorten the construction time. Thus construction and operating costs are expected to be competitive with coal-fired plants.

System 80+. An example of the large reactor concept as applied to the PWR is the System 80+ of ABB Combustion Engineering (acquired by Westinghouse), designed in conjunction with Duke Engineering Services. System 80+ is an improved "evolutionary" version in the category Generation III (90). It is an extension of System 80 that embodies several features, eg, safer design, simpler design, greater reliability, and enhanced operability. It has a large spherical steel containment building, gravity feed for the emergency water, hydrogen control, and decay-heat removal capability. System 80+ is considered highly safe but able to withstand any credible accident. It has two coolant loops, is rated at 1300 MWe, and the fuel contains the burnable poison erbium to enhance the fuel cycle. Lower enrichment ends of the fuel rods give an effective axial "blanket" that reduces neutron leakage. It also has an advanced control console that emphasizes the application of human engineering. Construction times of only four years are expected. Three units are operating in Arizona and several have been built in South Korea.

ABWR. An example of the large reactor is the Advanced Boiling Water Reactor of General Electric Company, designed and built in collaboration with the Japanese companies Hitachi and Toshiba (91). Goals for the reactor are reduced damage frequency by an order of magnitude, simplification of design, reduced costs of construction, fuel, and operation, and reduced radiation exposure and waste. New features are internal recirculation pumps, modern electronics, use of inert nitrogen in the containment to prevent hydrogen explosion, steel lining for the reinforced concrete containment, new control rod drives, a reactor vessel having forged rings instead of welded plates, and a backup gas turbine generator in addition to diesels.

ACR-700. The Advanced CANDU reactor is a pressurized heavy water reactor (PHWR) that is an enhanced version of the traditional CANDU (92). It was designed by Atomic Energy of Canada (AECL). Cooling of the 700-MWe reactor is by light water rather than heavy water, which is still used as moderator. Fuel is uranium at 2% U-235 with an at-power refueling machine.

France, which derives 75% of its electricity from nuclear energy, has chosen to make improvements in standard reactors and to participate in the European Pressurized Water Reactor program (93). Areva NP, the parent of Framatome and Siemens, leads the design process.

Generation III+. Reactors with additional improvements are in Generation III+.

AP1000. Many of the features of the AP600 are adapted for the larger power version of the passive safety type reactor (94). The Westinghouse AP-1000 fuel assemblies are in a 17 $\times$ 17 array with 264 fuel rods and 25 guide tubes for instruments and control rods. The assembly length is 3.7 m (12 ft). Fuel rod cladding is a zirconium alloy ZIRLO. Control is provided by rods with Ag–In–Cd alloy, by a boron coating on some fuel pellets, and burnable absorber rods containing boron.

GT-MHR. The Gas Turbine Modular Helium Reactor is an HTGR design of the company General Atomic (95). It would use uranium enriched to just under 20%. The heated helium would go directly to a turbine, achieving efficiencies of up to 50%. The concept involves no corrosion, produces much less waste per unit of power, and is considered very safe.

PBMR. The Pebble Bed Modular Reactor (96) consists of a vessel holding some 360,000 spherical fuel elements (pebbles) that are fed in and removed from the container. The hollow spheres are the size of a tennis ball, composed of pyrolitic graphite. Within the shells are thousands of 0.5-mm diameter coated fuel particles. These have layers of uranium oxide, pyrolitic carbon, and silicon carbide that seal in fission products. Cooling of the reactor is by helium, allowing for high temperature gas to drive a turbine directly and achieving efficiencies $\sim$ 50%.

Research on the pebble bed concept was carried out in Germany in the 1970s. Development is in progress by the South Africa company Eskom, in cooperation with Idaho National Laboratory. China is also working on the concept with cooperation between Tsinghua University and MIT.

IRIS. A new reactor design with global implications is IRIS (International Reactor Innovative and Secure). It is a PWR completely redesigned with all components including pressurizer, pumps, and steam generator inside the reactor vessel (97). The concept "Safety-by-Design" is applied, meaning that almost all sources of accident are eliminated and consequences are minimal. High fuel enrichment assures long cycle length. A consortium of reactor vendors, national laboratories, utilities, and universities carries out the R&D and design.

EPR. The European Pressurized Water Reactor was designed by Areva in cooperation with French and German companies (98). It is large, 1600 MWe, with four loops, 241 fuel assemblies, and high efficiency (37%). The EPR has numerous safety enhancements, featuring defense-in-depth and redundancy. Unique is the location of a large tank of borated water within the containment. The core loading involves a central region with checkerboard low and medium enrichment assemblies, surrounded by high enrichment assemblies, some with gadolinium as burnable poison. An EPR is already being built in Finland and the design may be adopted by other countries. A 60-page colorful brochure on EPR may be viewed on the Web.

ESBWR. The Economic Simplified Boiling Water Reactor of General Electric (99) is intended to cut construction and operating costs significantly.

The 1550-MWe reactor features new passive safety features. Cooling is by natural circulation, without recirculation pumps and pipes.

4S. This “Super Safe, Small and Simple” fast reactor is cooled by liquid sodium (100). It was designed by Toshiba for remote locations. Interest in a plant has been expressed by the village of Galena, Alaska, where the cost of diesel fuel is prohibitive. The 10-MWe reactor would use 19.9% enriched uranium or uranium–plutonium and would operate for many years without refueling. It would be placed in an underground vault.

Generation IV. Looking forward toward further improvements in reactor technology, the Department of Energy initiated a study of new designs titled Generation IV, abbreviated Gen IV. Generation IV reactors are considered promising for the more distant future, well into the twenty-first century. The growth in world population and expectations of developing countries will require greater demands for energy. Nuclear reactors can provide part of that energy economically, safely, and without environmental effects.

A DOE committee and an international Forum involving 10 countries selected 6 reactor systems that showed special promise to achieve a set of goals, briefly as follows:

Sustainability: long-term availability of nuclear fuels and favorable disposal of radioactive waste.
Economy: Low construction costs and competitive costs of energy production.
Safety: Use of inherent safety features that prevent accidents.
Proliferation resistance: Avoidance of diversion of fissile material.
Physical security: prevention of access and damage by terrorists.

A report was prepared (101) that emphasized the desire to achieve closed fuel cycles where possible, with nuclear fuel recycled to utilize fissile plutonium and to transmute higher isotopes that affect waste disposal. The potential for desalination of seawater and for production of hydrogen was also of high priority. The six reactor systems selected from a much longer list of suggestions are identified below:

1. Gas-Cooled Fast Reactor (GFR). Helium is used as coolant for fuel at high temperature. Fast neutrons convert uranium-238 and burn actinides. The closed fuel cycle uses advanced processing, including pyrometallurgical.
2. Lead-Cooled Fast Reactor (LFR). Liquid lead or a Pb–Bi mixture serve as coolant for fuel as metal or nitride. The long-life 10–30 years core favors nonproliferation.
3. Molten Salt Reactor (MSR). An adaptation of a design of the 1970s. A graphite moderated thermal reactor with circulating fuel composed of fluorides. Design involves ease in introducing actinides with no need for fuel fabrication.
4. Sodium-Cooled Fast Reactor (SFR). Full recycle with two processing options, aqueous or pyrometallurgical. Early deployment is possible.
5. Supercritical-Water-Cooled Reactor (SCWR). Two options—a thermal reactor with once-through cycle or a fast reactor with closed cycle. Water coolant of low density above the critical point (22.1 MPa, 374°C). High thermal efficiency (44%).

6. Very High Temperature Reactor (VHTR). Helium cooling of graphite prisms or pebbles. Thermal neutrons in an open cycle. Intended for process heat applications such as coal gasification and thermochemical hydrogen production.

Of the above six reactor concepts, the VHTR is considered to be the most promising and thus deserving of greatest emphasis for deployment $\sim$ 2020. It has been designated as the Next Generation Nuclear Plant (NGNP). It could be one of two concepts (1) the Gas Turbine Modular Helium Reactor, or (2) the Pebble Bed Modular Reactor, described earlier. Design studies will determine the choice of a direct thermal cycle with or without an intermediate heat exchanger and the coupling between the reactor and a process for generating hydrogen. Figure 14 shows a schematic view of the VHTR.

13. Safety and Security

A large inventory of radioactive fission products is present in any reactor fuel where the reactor has been operated for times on the order of months. In steady state, radioactive decay heat amounts to $\sim$ 5% of fission heat, and continues after a reactor is shut down. If cooling is not provided, decay heat can melt fuel rods, causing release of the contents. Protection against a loss-of-coolant accident (LOCA), eg, a primary coolant pipe break, is required. Power reactors have an emergency core cooling system (ECCS) that comes into play upon initiation of a LOCA.

Nuclear power has achieved an excellent safety record. In the United States, safety is enhanced by oversight by the Nuclear Regulatory Commission (NRC), which reviews proposed reactor designs, processes applications for licenses to construct and operate plants, and provides surveillance of all safety-related activities of a utility. A technique called probabilistic safety assessment (PSA) has been developed to analyze complex systems and to aid in assuring safe nuclear power plant operation. PSA, which had its origin in a project sponsored by the U.S. Atomic Energy Commission, is a formalized identification of potential events and consequences leading to an estimate of risk of accident. Discovery of weaknesses in the plant allows for corrective action. References include Fullwood and Hall (102), Henley and Kumamoto (103), and Lewis (104).

Reactors are designed to be inherently safe based on physical principles, supplemented by redundant equipment and special procedures. Nuclear power benefits from the application of the concept of defense in depth, ie, by using fuel form, reactor vessel, building containment, and emergency backup procedures to ensure safety.

The accident in 1979 at Three Mile Island Unit 2 (TMI-2), although highly publicized and very costly to clean up, resulted in minimum hazard to the public. The design included a thick steel reactor vessel and a tight containment building. The incident resulted from mechanical failure compounded by misinterpretation of events by the operating crew. A highly regarded description of the accident is the book by Walker (105).

The TMI-2 accident, which prompted a number of improvements in equipment and procedures, also led the nuclear industry to create the Institute of Nuclear Power Operations (INPO), a self-regulatory organization. The INPO maintains extensive safety-related databases, conducts power plant visits, and oversees operator-training programs. The utilities seek continued improvement in capability, use procedures extensively, and analyze any plant incidents for their root causes. Similar programs intended to ensure reactor safety are in place in other countries. The World Association of Nuclear Operators (WANO) performs for nuclear power plants throughout the world the same function as INPO in the United States.

The steam explosion of the Chernobyl reactor in Ukraine in 1986 caused scores of immediate deaths and released large amounts of radioactivity, with resultant contamination and radiation exposure. The accident occurred because of inadequate inherent safety, improper operating practices, and lack of containment. The Chernobyl accident resulted in some design and operation changes in the reactor, making it less vulnerable in future operation. Countries of the former USSR have been encouraged by the International Atomic Energy Agency (IAEA) and the United States to shut down graphite-moderated water-cooled reactors, but demands for electrical power have prevented such action.

The public perceives the risk of nuclear power to be much greater than that determined by experts. Among explanations for the discrepancy are the belief in the possibility of a disaster and the association of reactors with weapons. A more realistic view is provided by the estimate by Cohen (106) that the number of days of lost life expectancy due to smoking is 2400 while that due to living near a nuclear power plant is only 0.5. Despite concerns, the public favors continuation and extension of the use of nuclear power.

Nuclear security in the United States received attention well before the terrorist attacks on September 11, 2001 heightened public awareness (107). All of the licensed U.S. commercial reactors have a containment dome to protect the reactor from external damage as well as preventing the release of radiation. The structure protecting a reactor includes massive amounts of concrete reinforced by steel. Since 9/11, in accord with NRC requirements, industry has dramatically increased the protection of nuclear power plants against adversaries. Among the features are more and stronger physical barriers, special detection equipment, increased surveillance, more powerful weapons, and additional security personnel. The Nuclear Energy Institute (NEI) worked with several nuclear power plants to produce a Nuclear Plant Security Video to inform the public about how plants are protected. A free viewer is provided by NEI to observe the film.

For additional information see the article by John J. Taylor titled Nuclear Power Facilities, Safety.

14. Environmental Aspects

In contrast to power plants using fossil fuel, nuclear reactor plants emit no compounds of carbon, nitrogen, or sulfur, and thus do not contribute to acid rain, ozone layer depletion, or global warming. Emissions of radioactive materials during regular operations are within regulatory requirements based on medical knowledge. These emissions do include radionuclides of the noble gases xenon and krypton, which readily disperse throughout the atmosphere. Small quantities of soluble radionuclides are released into lakes or streams that provide very large dilution factors. Plant and animal life are monitored regularly at such facilities. On the other hand, there is the remote possibility of radioactive contamination of the environment in case of a reactor accident in which containment is breached.

As the result of many years of nuclear reactor research and development and weapons production in U.S. defense programs, a large number of sites were contaminated by radioactive materials. A thorough cleanup of this residue of the Cold War will extend well into the twenty-first century and cost many billions of dollars. New technologies are needed to minimize the cost of the cleanup operation.

14.1. Wastes. Nuclear reactors produce unique wastes because these materials undergo radioactive decay and in so doing emit harmful radiation. The origin, handling, and disposal methods for all types of radioactive wastes are described by Murray (108). Spent nuclear fuel has fission products, uranium, and transuranic elements. Plans call for permanent disposal in an underground repository. Geological studies have been completed at Yucca Mountain in Nevada (109). The Department of Energy is required to submit an application for license to the Nuclear Regulatory Commission. Until a repository is completed, spent fuel must be stored in water pools or in dry storage casks at nuclear plant sites.

Nuclear wastes are classified according to the level of radioactivity. Low level wastes (LLW) from reactors arise primarily from the cooling water, either because of leakage from fuel or activation of impurities by neutron absorption. Most LLW will be disposed of in near-surface facilities at various locations around the United States or stored in secure facilities. Mixed wastes are those having both a hazardous and a radioactive component. Transuranic (TRU) waste containing plutonium comes from chemical processes related to nuclear weapons production. These are to be placed in underground salt deposits in New Mexico.

Mill tailings are another form of nuclear waste. The residue from uranium ore extraction contains radium, the precursor of short-lived radon and its daughters. Piles of tailings must be properly covered.

Other wastes are expected to arise from the decontamination and decommissioning of existing nuclear facilities. These include reactors at the time of life extension or at the end of their operating life. Whereas technologies are available for waste disposal, there is much public resistance to the establishment of disposal facilities.

15. Additional Information

For the interested reader there is a wealth of nuclear knowledge.

Some of the general and comprehensive textbooks are Lamarsh and Baratta (110), Murray (111), Knief (112), Bodansky (113), and Glasstone and Sesonske (114).

Organizations dedicated to nuclear matters are the American Nuclear Society (115), Nuclear Energy Institute (116), and World Nuclear Association (117). An annual listing of all nuclear power reactors in the world giving type, organizations, status, and dates is available (118). The web site developed and managed by Joseph Gonyeau provides details about current nuclear power plants (119).

The National Council for Science and the Environment makes available Congressional Research Service Reports, in which topics such as energy, waste management, and climate change are discussed at length (120).

Key government agencies are the Department of Energy (121), the Nuclear Regulatory Commission (122), the Environmental Protection Agency (123) and the Department of Homeland Security (124). Worldwide nuclear activities are carried out by the International Atomic Energy Agency (125).

There are many universities with nuclear engineering programs. University of California at Berkeley maintains the WWW Virtual Library Nuclear Engineering (126); Massachusetts Institute of Technology has advanced research programs in nuclear science and engineering (127); North Carolina State University is the host of ANS Nuclear Engineering Sourcebook for graduate studies (128).

The Internet through the World Wide Web provides a very large source of information, but the investigator will recognize that most of the material has not been peer reviewed and is subject to error. Care is needed in the use of the admittedly valuable resources of Wikipedia (129), Alsos (130), and Google (131).

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Table 1. **Historical and Projected Operable Nuclear Capacities by Region, 2001–2025, Net Gigawatts**[a]

Region	2001[a]	2005	2010	2015	2020	2025
Industrialized	278.7	283.9	290.7	288.5	279.4	260.9
United States	98.2	100.2	99.3	99.5	99.6	99.6
other North America	11.4	14.6	15.9	15.9	15.9	13.0
Japan	43.2	45.0	49.4	52.2	52.2	51.9
France	63.1	63.5	66.6	66.6	66.6	64.7
United Kingdom	12.5	11.0	11.1	7.0	6.0	5.4
other Western Europe	50.3	49.7	48.4	47.3	39.1	26.3
EE/FSU[b]	46.3	46.6	46.4	45.0	39.9	34.7
Eastern Europe	11.6	11.8	10.7	10.7	11.3	11.3
Russia	20.8	22.0	23.5	22.5	16.7	14.5
Ukraine	11.2	11.3	11.9	11.9	11.9	8.9
other FSU	2.7	1.6	0.4	0.0	0.0	0.0
developing	27.6	37.9	44.7	59.6	63.2	70.4
China	2.2	7.6	8.6	16.6	16.6	19.6
South Korea	13.0	16.9	18.0	20.9	23.6	27.6
other	12.4	13.3	18.1	22.2	23.1	23.2
total world	*352.6*	*368.4*	*381.8*	*393.1*	*382.5*	*366.0*

[a]Ref. 35.
[b]EE/FSU = Eastern Europe/Former Soviet Union. Totals may not equal sum of components due to independent rounding.

Table 2. **Westinghouse Model 412 Pressurized Water Reactor**[a]

Parameter	Value
thermal power, MW	3425
electrical power, MWe	1150
reactor vessel ID, m	4.394
primary system pressure, MPa[b]	15.5
coolant flow rate, kg/s	17,438
coolant temperatures, °C	
inlet	291.9
outlet	325.8
rise	33.9
steam pressure, MPa[b]	6.9
fuel dimensions, mm	
fuel rod OD	9.14
Zircaloy-4 cladding thickness	0.572
diametral gap	0.157
UO_2 pellet diameter	7.844
lattice pitch	12.80
fuel assembly array	17×17
rods per assembly	264[c]
number of assemblies in core	193
rods per core	50,952
fuel total weight, kg	81,639
core dimensions, m	
effective diameter	3.38
fuel height	3.658

[a]Ref 45.
[b]To convert MPa to psia, multiply by 145.
[c]25 spaces are taken by control rods, burnable poison rods, or neutron sources, or are plugged.

Table 3. **Design Data for Model BWR/6, General Electric Co.**[a]

Parameter	Value
reactor power, MW	
thermal	3579
electric	1220
reactor vessel pressure, MPa[b]	7–17
temperature, °C	
coolant	288
fuel[b]	1871
linear thermal output,[c] kW/m	44
initial fuel enrichment in U-235, wt%	1.7–2.0
fuel rods, OD, mm	12.27
Zircaloy 2 cladding thickness, mm	0.81
number of fuel assemblies	748
number of B_4C control rods	177
reactor vessel dimensions, m	
height	22
diameter	6

[a]Courtesy of GE Nuclear Energy. For a more complete list, see Refs. (28) and (47).
[b]To convert MPa to psia, multiply by 145.
[c]Value given is maximum.

Table 4. **Data for the SuperPhénix**[a]

Parameter	Value
power, MW	
thermal	3000
electric	1200
sodium coolant temperatures, °C	395–545
coolant flow rate, kg/s	16,900
reactor vessel dimensions, m	
height	19.5
diameter	21
core dimensions	
fuel height, m	1.0
diameter, m	3.7
volume, L	10,766
core fuel composition, wt%	
UO_2	83
PuO_2	17
peak flux, $(cm^2 \cdot s)^{-1}$	6.5×10^{15}
fuel pin diameter, mm	8.5
cladding, type[b]	316
pin pitch (triangular), mm	9.8
pins per assembly	271
number of core assemblies	384
control material	B_4C
number of control assemblies	24
blanket fuel	UO_2
average fuel burnup, MW·d/kg	44
refueling interval, d	320
Doppler coefficient	−0.0086
breeding ratio	1.25
doubling time, yr	23

[a] Ref. 56.
[b] Stainless steel.

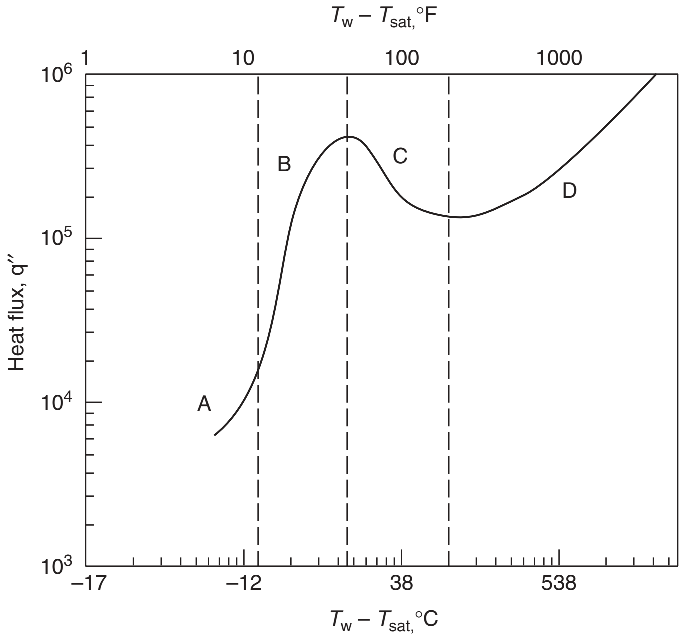

Fig. 1. Variation of heat flux, q'', with temperature difference between heated wall, T_w, and saturation temperature of water, T_{sat}, in regions, where A represents convection; B nucleate boiling; C transition; and D film boiling.

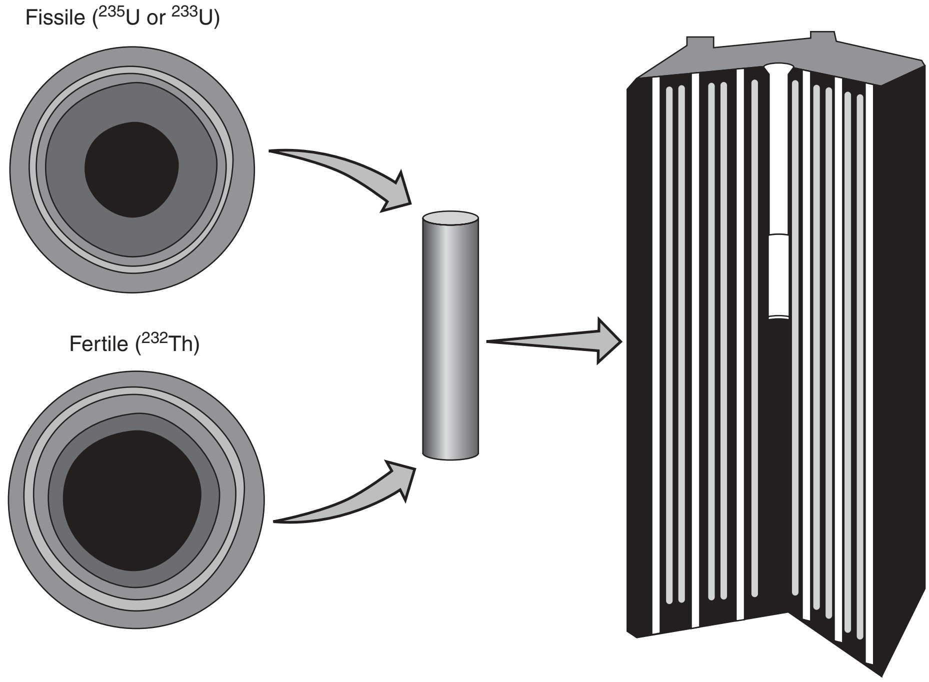

Fig. 2. Fuel for high temperature gas-cooled reactor. Fissile material is coated with carbon and silicon carbide, fertile material with carbon. Particles mixed with carbon form fuel rods inserted in graphite blocks. (Courtesy of General Atomics.)

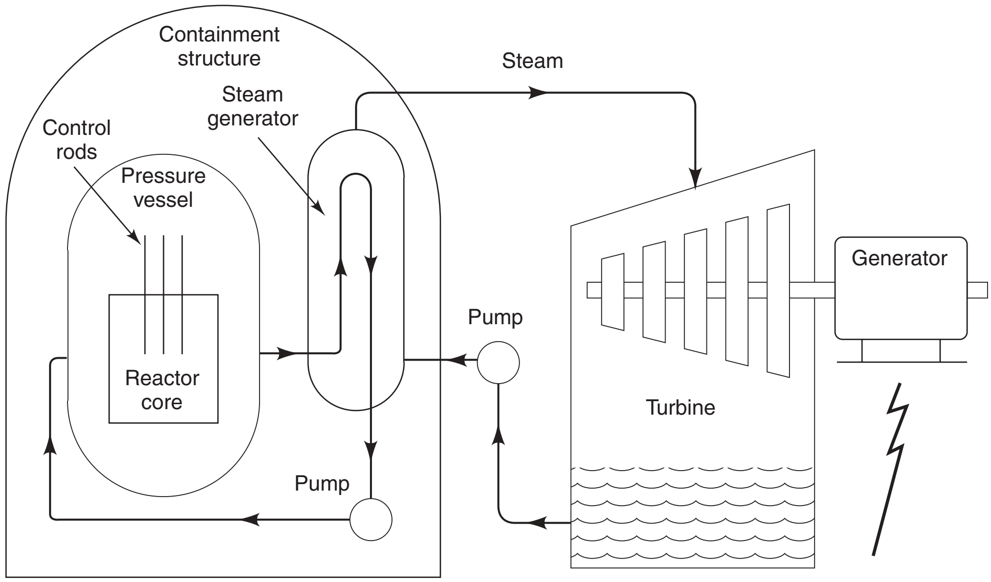

Fig. 3. Schematic of a pressurized water reactor system. Fission heat is extracted by the lightwater coolant. The steam drives the turbine-generator. (Courtesy of the Nuclear Energy Institute.)

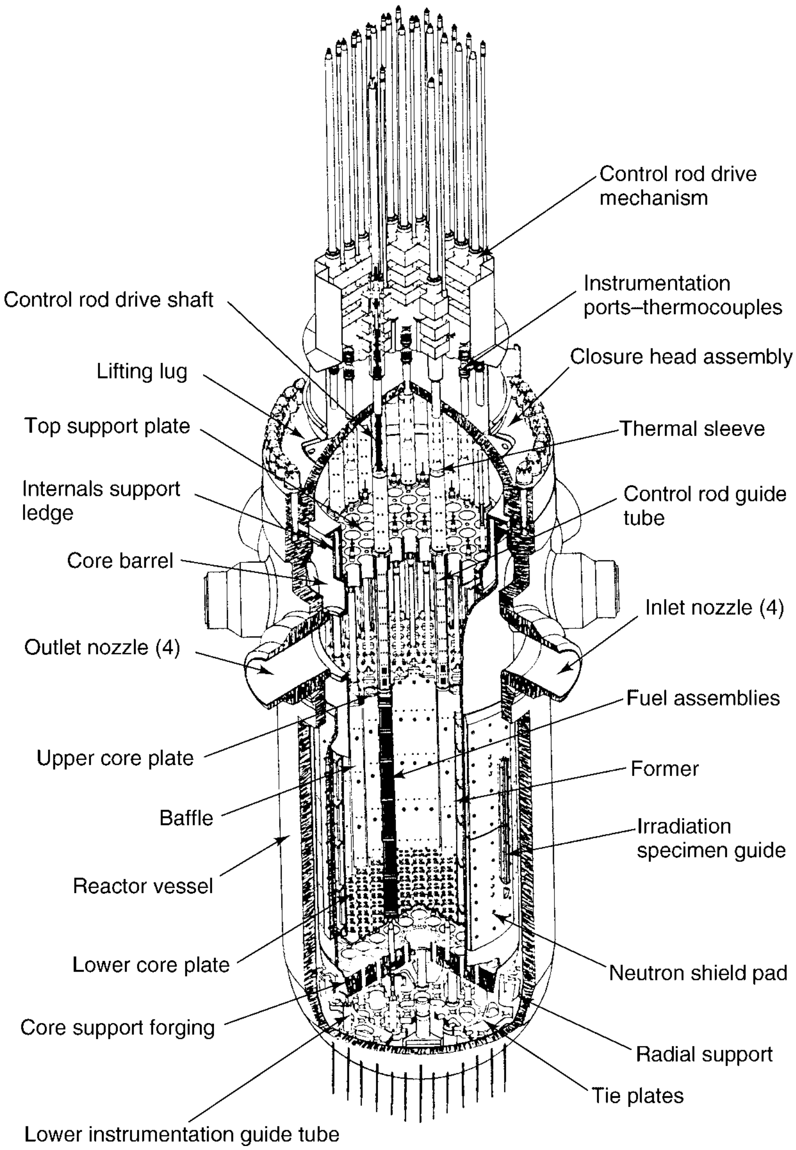

Fig. 4. Cutaway view of the Model 412 four-loop pressurized water reactor vessel (45). (Courtesy of Westinghouse Electric Corp.)

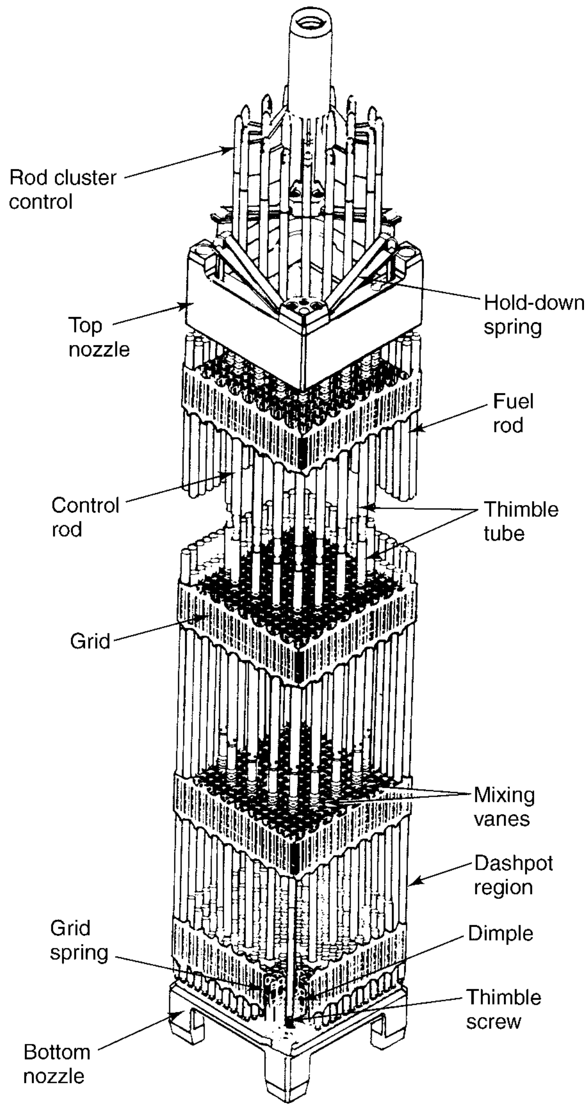

Fig. 5. Fuel assembly of Model 412 PWR having a 17×17 array of rods (45). (Courtesy of Westinghouse Electric Corp.)

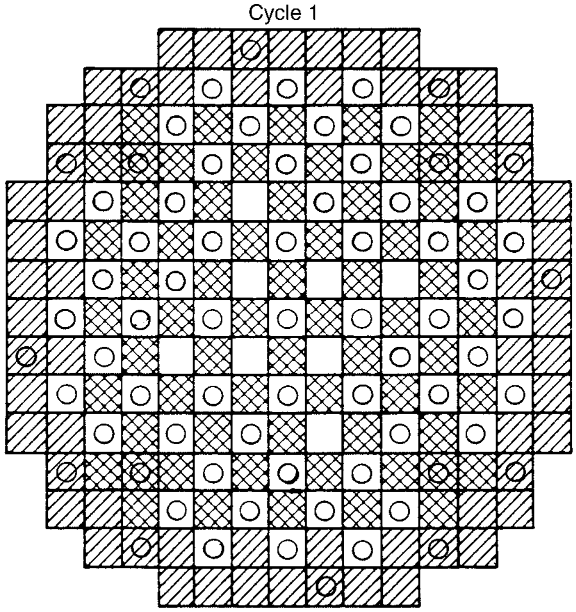

Fig. 6. Initial fuel loading of PWR showing the three enrichments where (▨) represents 3.10 wt% ^{235}U; (⊠) 2.60; (□) 2.10; and (⊡) the cluster openings of control rods (45). (Courtesy of Westinghouse Electric Corp.)

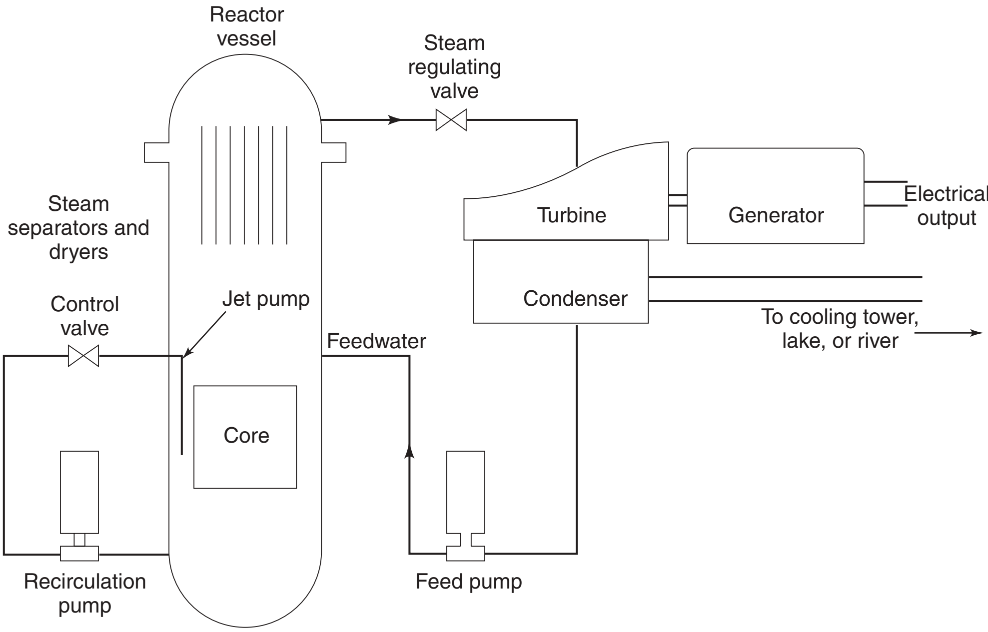

Fig. 7. Flow diagram of a BWR direct-cycle system. The demineralizers, heaters, and one recirculation loop are omitted.

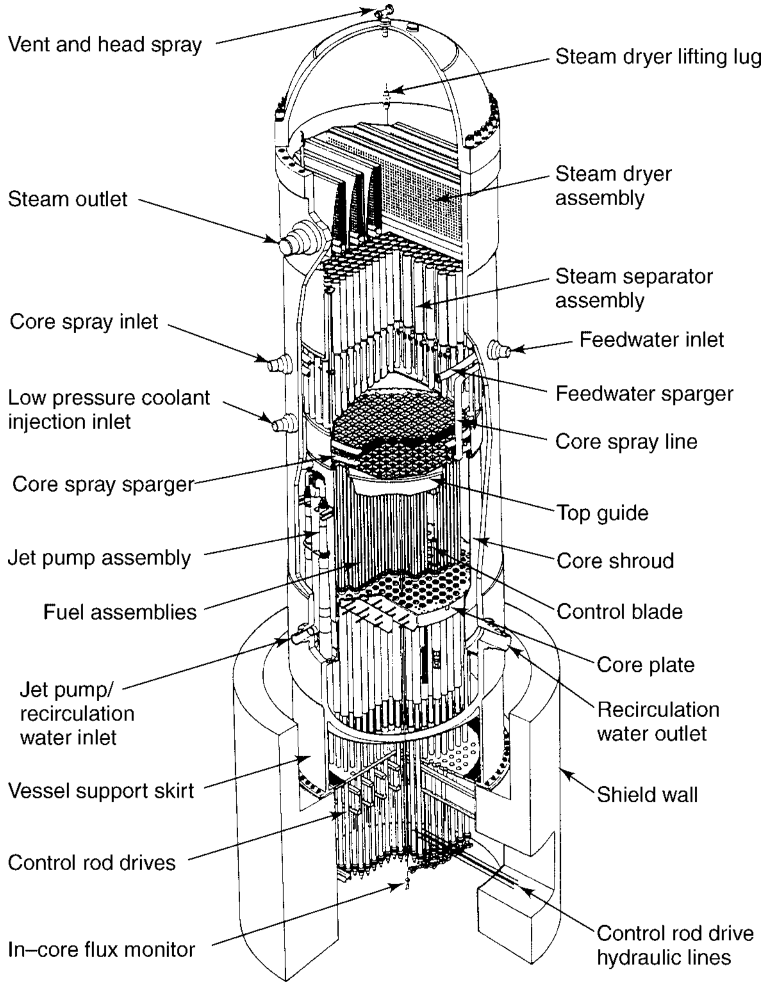

Fig. 8. Cutaway view of the Model BWR/6 pressure vessel (47). (Courtesy of GE Nuclear Energy).

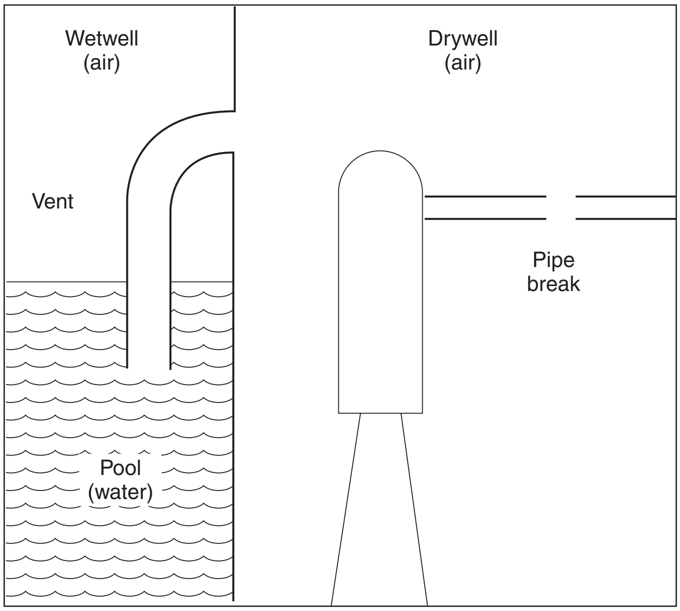

Fig. 9. Principle of pressure suppression containment. Steam from a broken pipe escapes from the drywell through a vent and is condensed in the water pool.

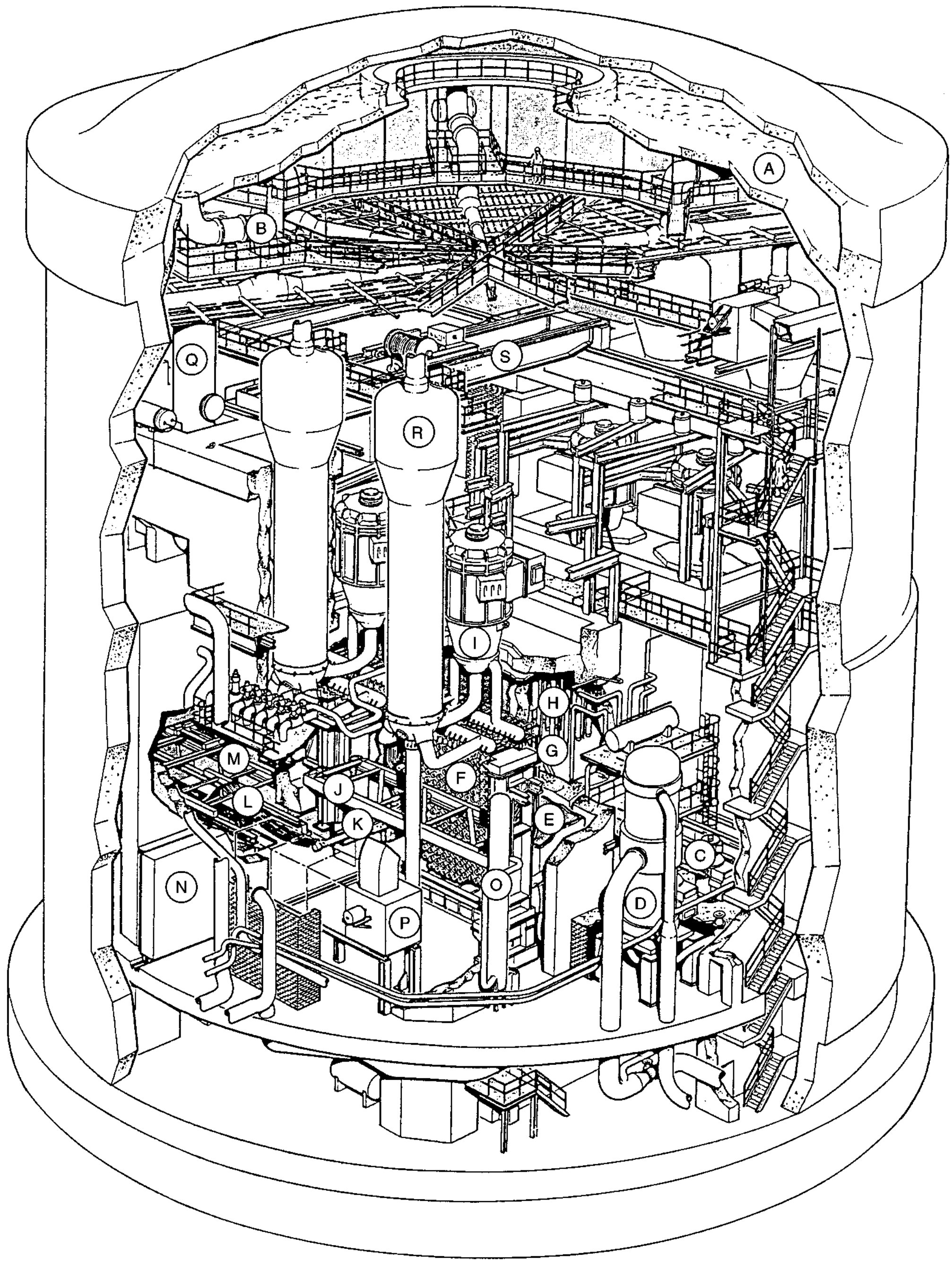

Fig. 10. Cutaway view of containment building of CANDU reactor where A is the dousing water tank; B, dousing water valves; C, moderator pump; D, moderator heat exchanger; E, feeder cabinets; F, reactor face; G, reactor; H, reactivity mechanism; I, heat transport system pump; J, fueling machine bridge; K, fueling machine carriage; L, fueling machine catenary; M, fueling machine maintenance lock; N, fueling machine maintenance lock door; O, end shield cooling water delay tank; P, vault cooler; Q, pressurizer; R, steam generator; and S, steam generator room crane. Courtesy of Atomic Energy of Canada, Ltd.

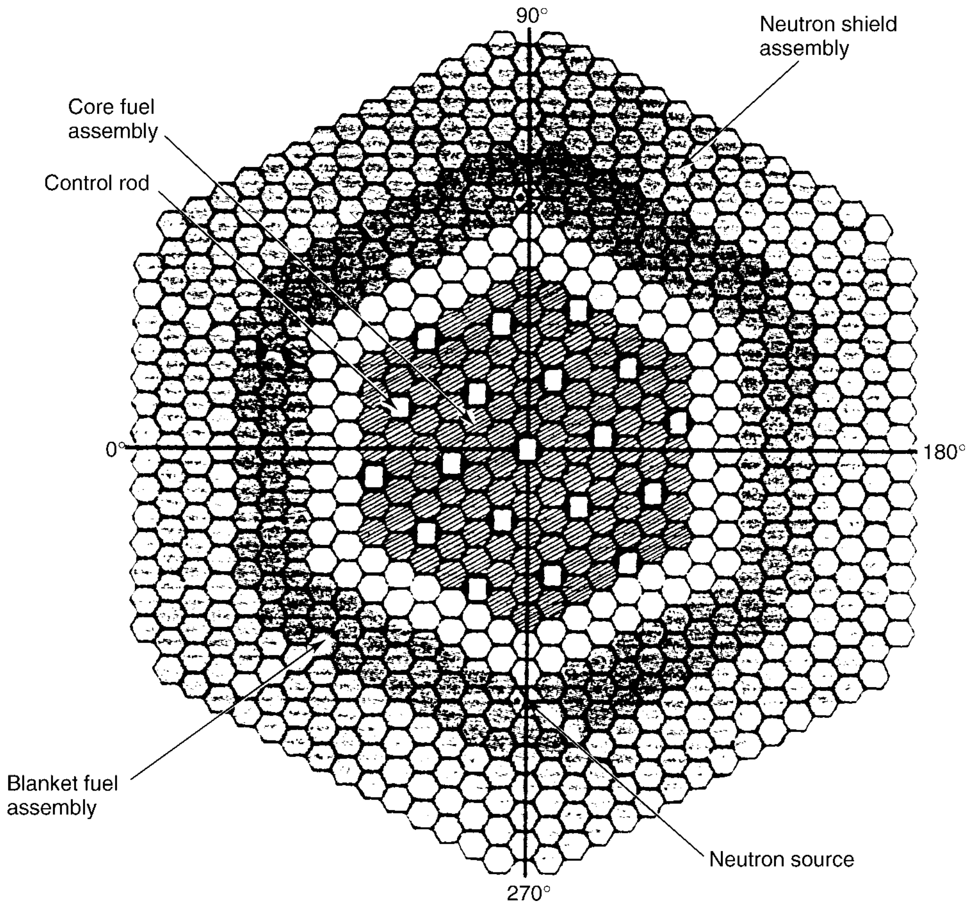

Fig. 11. Reactor core of MONJU, the Japanese fast-breeder reactor. Courtesy of Power Reactor and Nuclear Fuel Development Corp.

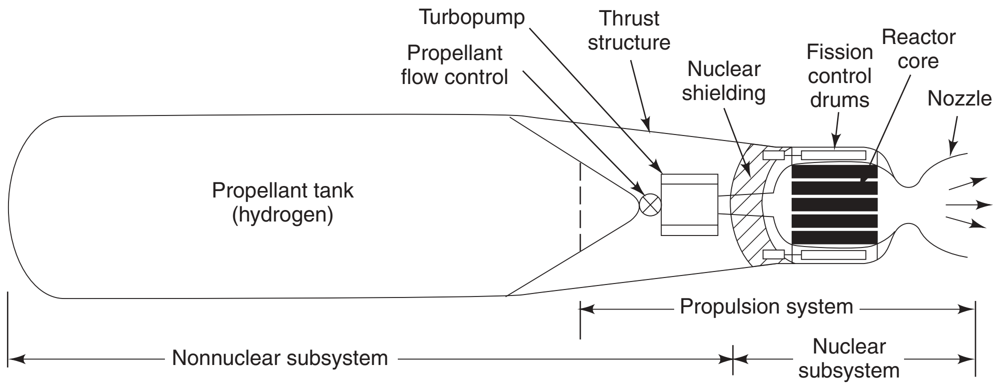

Fig. 12. Schematic of a nuclear rocket (72). Liquid hydrogen is heated by a reactor and expelled as a gas through the nozzle.

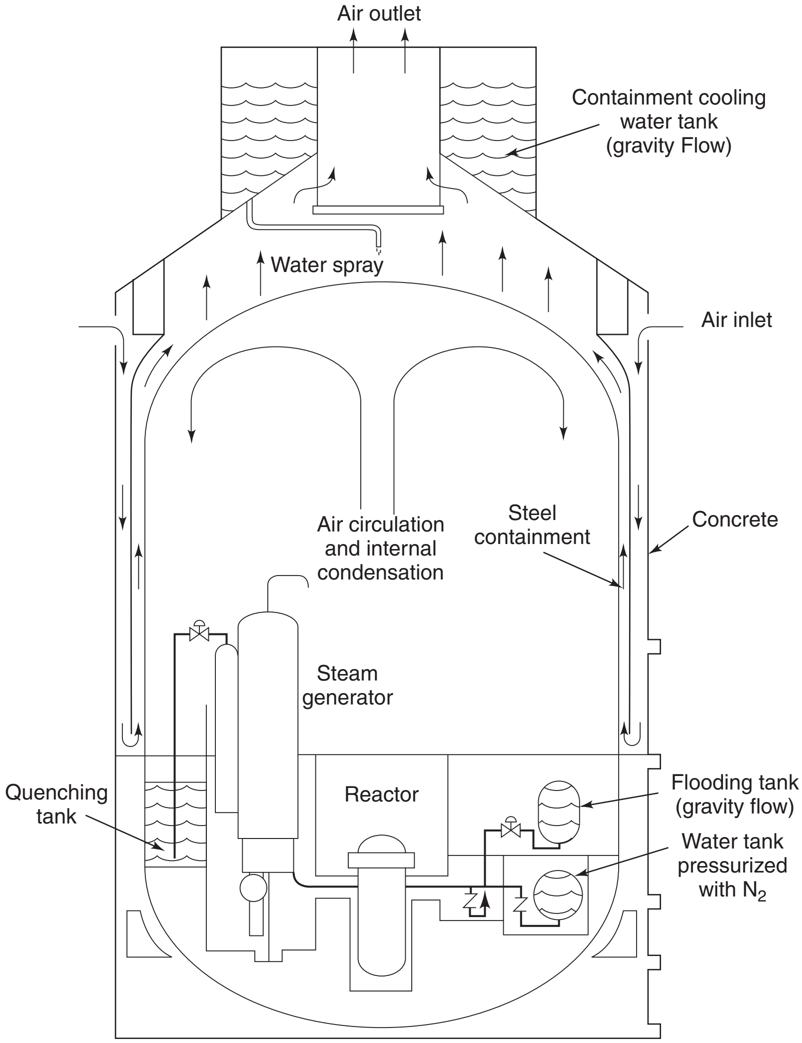

Fig. 13. Proposed advanced PWR design having passive safety features. (Courtesy of Westinghouse Electric Corp.)

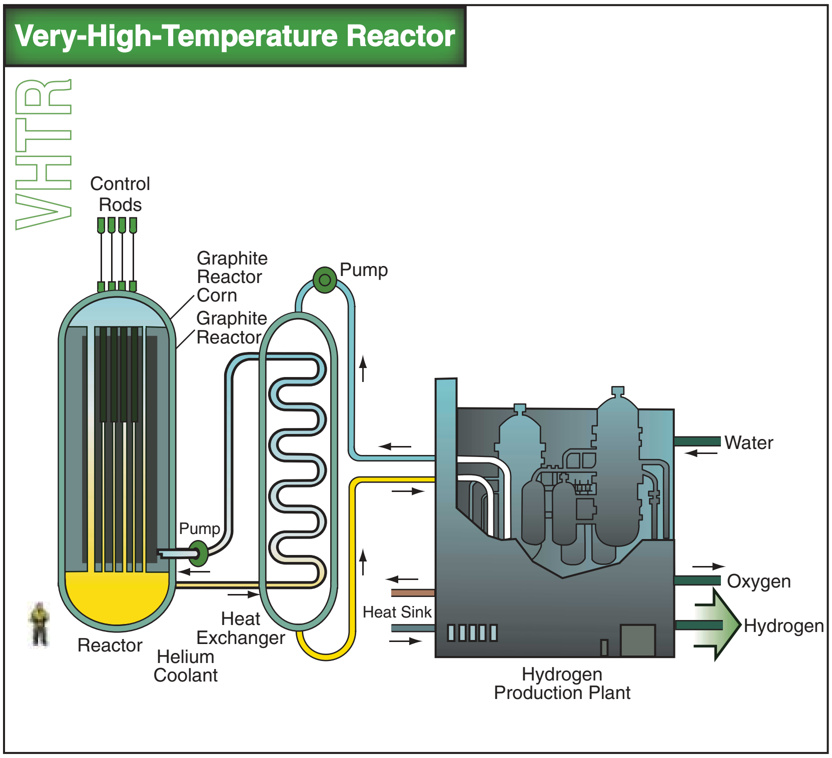

Fig. 14. Diagram of a very high temperature reactor that may be used for hydrogen production. From the Idano National Laboratory (http://nuclear.inl.gov/gen4/vhtr.shtml.)