CHAPTER 56

NUCLEAR POWER

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56.1 HISTORICAL PERSPECTIVE

56.1.1 The Birth of Nuclear Energy

The first large-scale application of nuclear energy was in a weapon. The second use was in submarine propulsion systems. Subsequent development of fission reactors for electric power production has

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been profoundly influenced by these early military associations, both technically and politically. It appears likely that the military connection, tenuous though it may be, will continue to have a strong political influence on applications of nuclear energy.

Fusion, looked on by many as a supplement to, or possibly as an alternative to fission for producing electric power, was also applied first as a weapon. Most of the fusion systems now being investigated for civilian applications are far removed from weapons technology. A very few are related closely enough that further civilian development could be inhibited by this association.

56.1.2 Military Propulsion Units

The possibilities inherent in an extremely compact source of fuel, the consumption of which requires no oxygen, and produces a small volume of waste products, was recognized almost immediately after World War II by those responsible for the improvement of submarine propulsion units. Significant resources were soon committed to the development of a compact, easily controlled, quiet, and highly reliable propulsion reactor. As a result, a unit was produced which revolutionized submarine capabilities.

The decisions that led to a compact, light-water-cooled and -moderated submarine reactor unit, using enriched uranium for fuel, were undoubtedly valid for this application. They have been adopted by other countries as well. However, the technological background and experience gained by U.S. manufacturers in submarine reactor development was a principal factor in the eventual decision to build commercial reactors that were cooled with light water and that used enriched uranium in oxide form as fuel. Whether this was the best approach for commercial reactors is still uncertain.

56.1.3 Early Enthusiasm for Nuclear Power

Until the passage, in 1954, of an amendment to the Atomic Energy Act of 1946, almost all of the technology that was to be used in developing commercial nuclear power was classified. The 1954 Amendment made it possible for U.S. industry to gain access to much of the available technology, and to own and operate nuclear power plants. Under the amendment the Atomic Energy Commission (AEC), originally set up for the purpose of placing nuclear weapons under civilian control, was given responsibility for licensing and for regulating the operation of these plants.

In December of 1953 President Eisenhower, in a speech before the General Assembly of the United Nations, extolled the virtues of peaceful uses of nuclear energy and promised the assistance of the United States in making this potential new source of energy available to the rest of the world. Enthusiasm over what was then viewed as a potentially inexpensive and almost inexhaustible new source of energy was a strong force which led, along with the hope that a system of international inspection and control could inhibit proliferation of nuclear weapons, to formation of the International Atomic Energy Agency (IAEA) as an arm of the United Nations. The IAEA, with headquarters in Vienna, continues to play a dual role of assisting in the development of peaceful uses of nuclear energy, and in the development of a system of inspections and controls aimed at making it possible to detect any diversion of special nuclear materials, being used in or produced by civilian power reactors, to military purposes.

56.1.4 U.S. Development of Nuclear Power

Beginning in the early 1950s the AEC, in its national laboratories, and with the participation of a number of industrial organizations, carried on an extensive program of reactor development. A variety of reactor systems and types were investigated analytically and several prototypes were built and operated.

In addition to the light water reactor (LWR), gas-cooled graphite-moderated reactors, liquid-fueled reactors with fuel incorporated in a molten salt, liquid-fueled reactors with fuel in the form of a uranium nitrate solution, liquid-sodium-cooled graphite-moderated reactors, solid-fueled reactors with organic coolant, and liquid-metal solid-fueled fast spectrum reactors have been developed and operated, at least in pilot plant form in the United States. All of these have had enthusiastic advocates. Most, for various reasons, have not gone beyond the pilot plant stage. Two of these, the high-temperature gas-cooled reactor (HTGR) and the liquid-metal-cooled fast breeder reactor (LMFBR), have been built and operated as prototype power plants.

Some of these have features associated either with normal operation, or with possible accident situations, which seem to make them attractive alternatives to the LWR. The HTGR, for example, operates at much higher outlet coolant temperature than the LWR and thus makes possible a significantly more efficient thermodynamic cycle as well as permitting use of a physically smaller steam turbine. The reactor core, primarily graphite, operates at a much lower power density than that of LWRs. This lower power density and the high-temperature capability of graphite make the HTGR's core much more tolerant of a loss-of-coolant accident than the LWR core.

The long, difficult, and expensive process needed to take a conceptual reactor system to reliable commercial operation has unquestionably inhibited the development of a number of alternative systems.

56.2 CURRENT POWER REACTORS, AND FUTURE PROJECTIONS

Although a large number of reactor types have been studied for possible use in power production, the number now receiving serious consideration is rather small.

56.2.1 Light-Water-Moderated Enriched-Uranium-Fueled Reactor

The only commercially viable power reactor systems operating in the United States today use LWRs. This is likely to be the case for the next decade or so. France has embarked on a construction program that will eventually lead to productions of about 90% of its electric power by LWR units. Great Britain has under consideration the construction of a number of LWRs. The Federal Republic of Germany has a number of LWRs in operation with additional units under construction. Russia and a number of other Eastern European countries are operating LWRs, and are constructing additional plants. Russia is also building a number of smaller, specially designed LWRs near several population centers. It is planned to use these units to generate steam for district heating. The first one of these reactors is scheduled to go into operation soon near Gorki.

56.2.2 Gas-Cooled Reactor

Several designs exist for gas-cooled reactors. In the United States the one that has been most seriously considered uses helium for cooling. Fuel elements are large graphite blocks containing a number of vertical channels. Some of the channels are filled with enriched uranium fuel. Some, left open, provide a passage for the cooling gas. One small power reactor of this type is in operation in the United States. Carbon dioxide is used for cooling in some European designs. Both metal fuels and graphite-coated fuels are used. A few gas-cooled reactors are being used for electric power production both in England and in France.

56.2.3 Heavy-Water-Moderated Natural-Uranium-Fueled Reactor

The goal of developing a reactor system that does not require enriched uranium led Canada to a natural-uranium-fueled, heavy-water-moderated, light-water-cooled reactor design dubbed Candu. A number of these are operating successfully in Canada. Argentina and India each uses a reactor power plant of this type, purchased from Canada, for electric power production.

56.2.4 Liquid-Metal-Cooled Fast Breeder Reactor

France, England, Russia, and the United States all have prototype liquid-metal-cooled fast breeder reactors (LMFBRs) in operation. Experience and analysis provide evidence that the plutonium-fueled LMFBR is the most likely, of the various breeding cycles investigated, to provide a commercially viable breeder. The breeder is attractive because it permits as much as 80% of the available energy in natural uranium to be converted to useful energy. The LWR system, by contrast, converts at most 3%–4%.

Because plutonium is an important constituent of nuclear weapons, there has been concern that development of breeder reactors will produce nuclear weapons proliferation. This is a legitimate concern, and must be dealt with in the design of the fuel cycle facilities that make up the breeder fuel cycle.

56.2.5 Fusion

It may be possible to use the fusion reaction, already successfully harnessed to produce a powerful explosive, for power production. Considerable effort in the United States and in a number of other countries is being devoted to development of a system that would use a controlled fusion reaction to produce useful energy. At the present stage of development the fusion of tritium and deuterium nuclei appears to be the most promising reaction of those that have been investigated. Problems in the design, construction, and operation of a reactor system that will produce useful amounts of economical power appear formidable. However, potential fuel resources are enormous, and are readily available to any country that can develop the technology.

56.3 CATALOG AND PERFORMANCE OF OPERATING REACTORS, WORLDWIDE

Worldwide, the operation of nuclear power plants in 1982 produced more than 10% of all the electrical energy used. Table 56.1 contains a listing of reactors in operation in the United States and in the rest of the world.

56.4 U.S. COMMERCIAL REACTORS

As indicated earlier, the approach to fuel type and core design used in LWRs in the United States comes from the reactors developed for marine propulsion by the military.

56.4.1 Pressurized-Water Reactors

Of the two types developed in the United States, the pressurized water reactor (PWR) and the boiling water reactor (BWR), the PWR is a more direct adaptation of marine propulsion reactors. PWRs are

Table 56.1 Operating Power Reactors (1995)

Country	Reactor Type[a]	Number in Operation	Net MWe
Argentina	PHWR	3	1627
Armenia	PWR	2	800
Belgium	PWR	7	5527
Brazil	PWR	1	626
Bulgaria	PWR	6	3420
Canada	PHWR	22	15439
China	PWR	3	2100
Czech Republic	PWR	4	1632
Finland	PWR	2	890
	BWR	2	1420
France	PWR	54	57140
Germany	PWR	14	15822
	BWR	7	6989
Hungary	PWR	4	1729
India	BWR	2	300
	PHWR	8	1395
Japan	PWR	22	17298
	BWR	26	22050
Korea	PWR	9	7541
	PHWR	1	629
Lithuania	LGR	2	2760
Mexico	BWR	2	1308
Netherlands	PWR	1	452
	BWR	1	55
Pakistan	PHWR	1	125
Russia	LGR	11	10175
	PWR	13	9064
	LMFBR	1	560
Slovenia	PWR	1	620
Slovokia	PWR	4	1632
South Africa	PWR	2	1840
Spain	BWR	2	1389
	PWR	7	5712
Sweden	BWR	9	7370
	PWR	3	2705
Switzerland	BWR	2	1385
	PWR	3	1665
Taiwan	BWR	4	3104
	PWR	2	1780
UK	GCR	20	3360
	AGR	14	8180
	PWR	1	1188
Ukraine	LGR	2	1850
	PWR	12	10245
United States	BWR	37	32215
	PWR	72	67458

[a]PWR = pressurized water reactor; BWR = boiling water reactor; AGR = advanced gas-cooled reactor; GCR = gas-cooled reactor; HTGR = high-temperature gas-cooled reactor; LMFBR = liquid-metal fast-breeder reactor; LGR = light-water-cooled graphite-moderated reactor; HWLWR = heavy-water-moderated light-water-cooled reactor; PHWR = pressurized heavy-water-moderated-and-cooled reactor; GCHWR = gas-cooled heavy-water-moderated reactor.

operated at pressures in the pressure vessel (typically about 2250 psi) and temperatures (primary inlet coolant temperature is about 564°F with an outlet temperature about 64°F higher) such that bulk boiling does not occur in the core during normal operation. Water in the primary system flows through the core as a liquid, and proceeds through one side of a heat exchanger. Steam is generated on the other side at a temperature slightly less than that of the water that emerges from the reactor vessel outlet. Figure 56.1 shows a typical PWR vessel and core arrangement. Figure 56.2 shows a steam generator.

The reactor pressure vessel is an especially crucial component. Current U.S. design and operational philosophy assumes that systems provided to ensure maintenance of the reactor core integrity

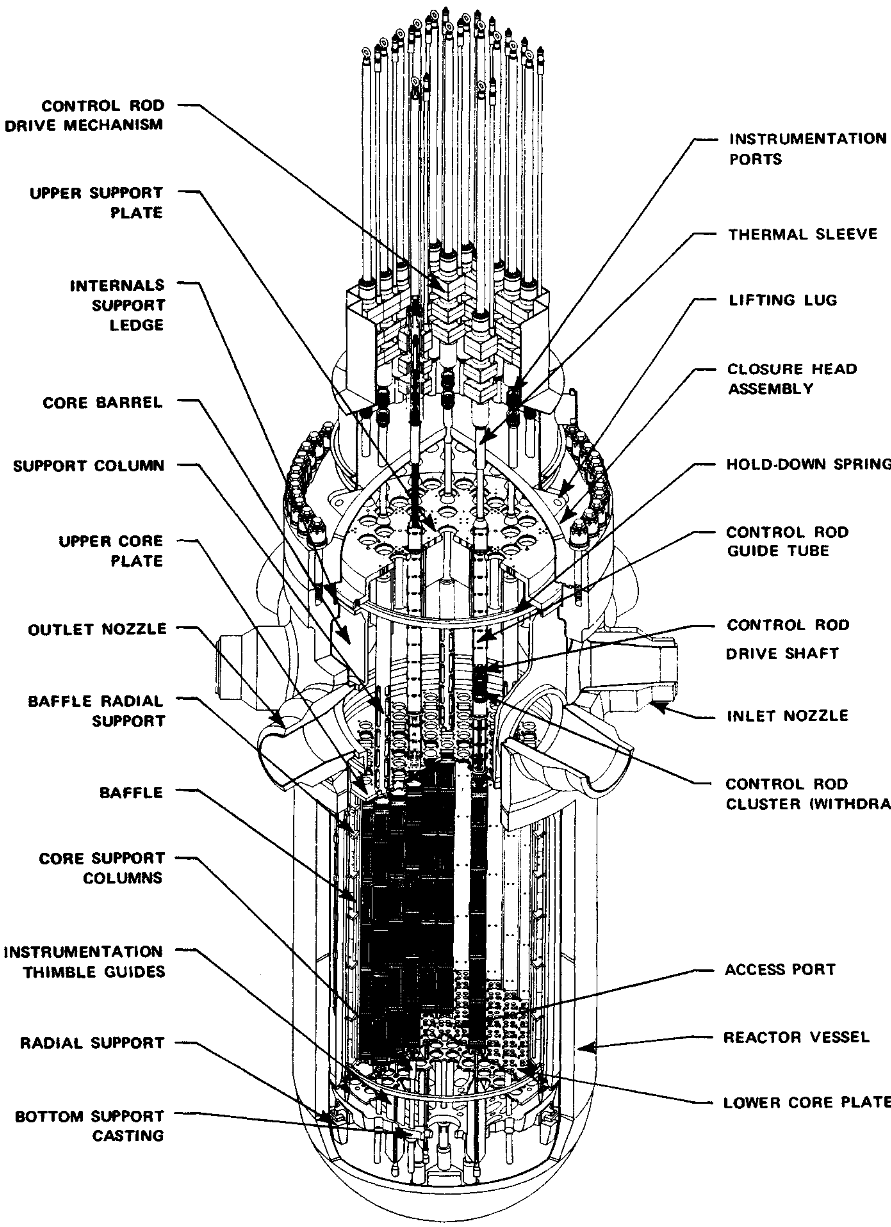


Fig. 56.1 Typical vessel and core configuration for PWR. (Courtesy Westinghouse.)

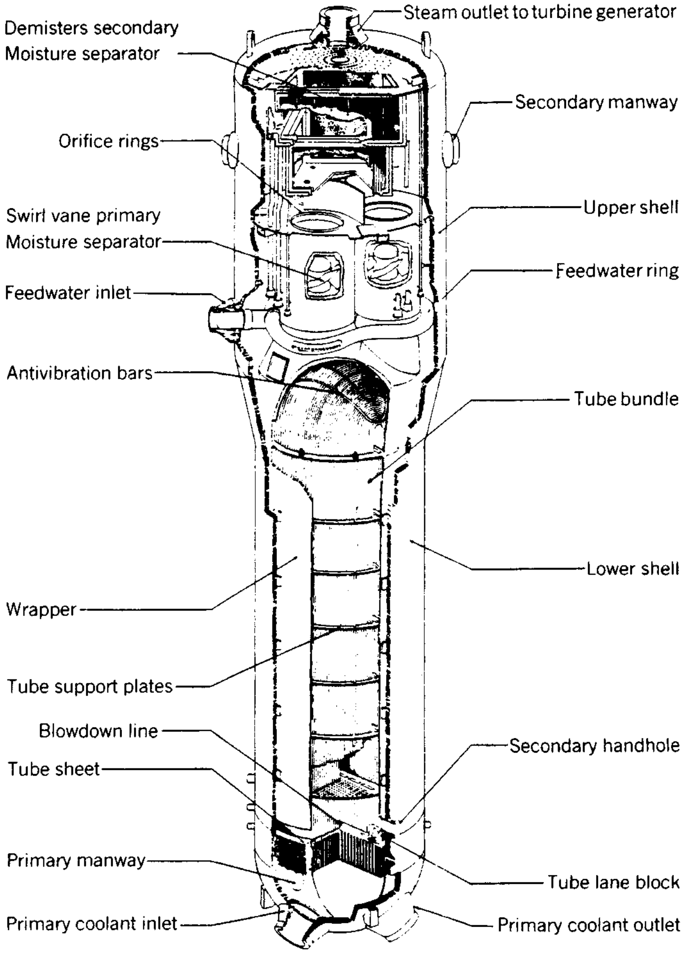


Fig. 56.2 Typical PWR steam generator.

under both normal and emergency conditions will be able to deliver cooling water to a pressure vessel whose integrity is virtually intact after even the most serious accident considered in the safety analysis of hypothesized accidents required by U.S. licensing. A special section of the ASME Pressure Vessel Code, Section III, has been developed to specify acceptable vessel design, construction, and operating practices. Section XI of the code specifies acceptable inspection practices.

Practical considerations in pressure vessel construction and operation determine an upper limit to the primary operating pressure. This in turn prescribes a maximum temperature for water in the primary. The resulting steam temperature in the secondary is considerably lower than that typical of modern fossil-fueled plants. (Typical steam temperatures and pressures are about 1100 psi and 556°F at the steam generator outlet.) This lower steam temperature has required development of massive steam turbines to handle the enormous steam flow of the low-temperature steam produced by the large PWRs of current design.

56.4.2 Boiling-Water Reactors

As the name implies, steam is generated in the BWR by boiling, which takes place in the reactor core. Early concerns about nuclear and hydraulic instabilities led to a decision to operate military propulsion reactors under conditions such that the moderator–coolant in the core remains liquid. In the course of developing the BWR system for commercial use, solutions have been found for the instability problems.

Although some early BWRs used a design that separates the core coolant from the steam which flows to the turbine, all modern BWRs send steam generated in the core directly to the turbine. This arrangement eliminates the need for a separate steam generator. It does, however, provide direct communication between the reactor core and the steam turbine and condenser, which are located outside the containment. This leads to some problems not found in PWRs. For example, the turbine–condenser system must be designed to deal with radioactive nitrogen-16 generated by an (n,p) reaction of fast neutrons in the reactor core with oxygen-16 in the cooling water. Decay of the short-lived nitrogen-16 (half-life 7.1 sec) produces high-energy (6.13-MeV) highly penetrating gamma rays. As a result, the radiation level around an operating BWR turbine requires special precautions not needed for the PWR turbine. The direct pathway from core to turbine provided by the steam pipes also affords a possible avenue of escape and direct release outside of containment for fission products that might be released from the fuel in a core-damaging accident. Rapid-closing valves in the steam lines are provided to block this path in case of such an accident.

The selection of pressure and temperature for the steam entering the turbine that are not markedly different from those typical of PWRs leads to an operating pressure for the BWR pressure vessel that is typically less than half that for PWRs. (Typical operating pressure at vessel outlet is about 1050 psi with a corresponding steam temperature of about 551°F.)

Because it is necessary to provide for two-phase flow through the core, the core volume is larger than that of a PWR of the same power. The core power density is correspondingly smaller. Figure 56.3 is a cutaway of a BWR vessel and core arrangement. The in-vessel steam separator for removing moisture from the steam is located above the core assembly. Figure 56.4 is a BWR fuel assembly. The assembly is contained in a channel box, which directs the two-phase flow. Fuel pins and fuel pellets are not very different in either size or shape from those for PWRs, although the cladding thickness for the BWR pin is somewhat larger than that of PWRs.

56.4.3 High-Temperature Gas-Cooled Reactors

Experience with the high-temperature gas-cooled reactor (HTGR) in the United States is limited. A 40-MWe plant was operated from 1967 to 1974. A 330-MWe plant has been in operation since 1976. A detailed design was developed for a 1000-MWe plant, but plans for its construction were abandoned.

Fuel elements for the plant in operation are hexagonal prisms of graphite about 31 in. tall and 5.5 in. across flats. Vertical holes in these blocks allow for passage of the helium coolant. Fuel elements for the larger proposed plant were similar. Figure 56.5 shows core and vessel arrangement. Typical helium-coolant outlet temperature for the reactor now in operation is about 1300°F. Typical steam temperature is 1000°F. The large plant was also designed to produce 1000°F steam.

The fuel cycle for the HTGR was originally designed to use fuel that combined highly enriched uranium with thorium. This cycle would convert thorium to uranium-233, which is also a fissile material, thereby extending fuel lifetime significantly. This mode of operation also produces uranium-233, which can be chemically separated from the spent fuel for further use. Recent work has resulted in the development of a fuel using low-enriched uranium in a once-through cycle similar to that used in LWRs.

The use of graphite as a moderator and helium as coolant allows operation at temperatures significantly higher than those typical of LWRs, resulting in higher thermal efficiencies. The large thermal capacity of the graphite core and the large negative temperature coefficient of reactivity make the HTGR insensitive to inadvertent reactivity insertions and to loss-of-coolant accidents. Operating experience to date gives some indication that the HTGR has advantages in increased safety and in lower radiation exposure to operating personnel. These possible advantages plus the higher thermal efficiency that can be achieved make further development attractive. However, the high cost of developing a large commercial unit, plus the uncertainties that exist because of the limited operating experience with this type reactor have so far outweighed the perceived advantages.

As the data in Table 56.1 indicate, there is significant successful operating experience with several types of gas-cooled reactors in a number of European countries.

56.4.4 Constraints

Reactors being put into operation today are based on designs that were originally conceived as much as 20 years earlier. The incredible time lag between the beginning of the design process and the operation of the plant is one of the unfortunate products of a system of industrial production and federal regulation that moves ponderously and uncertainly toward producing a power plant that may be technically obsolescent by the time it begins operation. The combination of the large capital investment required for plant construction, the long period during which this investment remains unproductive for a variety of reasons, and the high interest rates charged for borrowed money have recently led to plant capital costs some 5–10 times larger than those for plants that came on line in the early to mid 1970s. Added to the above constraints is a widespread concern about dangers of nuclear power. These concerns span a spectrum that encompasses fear of contribution to nuclear weapons proliferation, on the one hand, to a strong aversion to high technology, on the other hand.

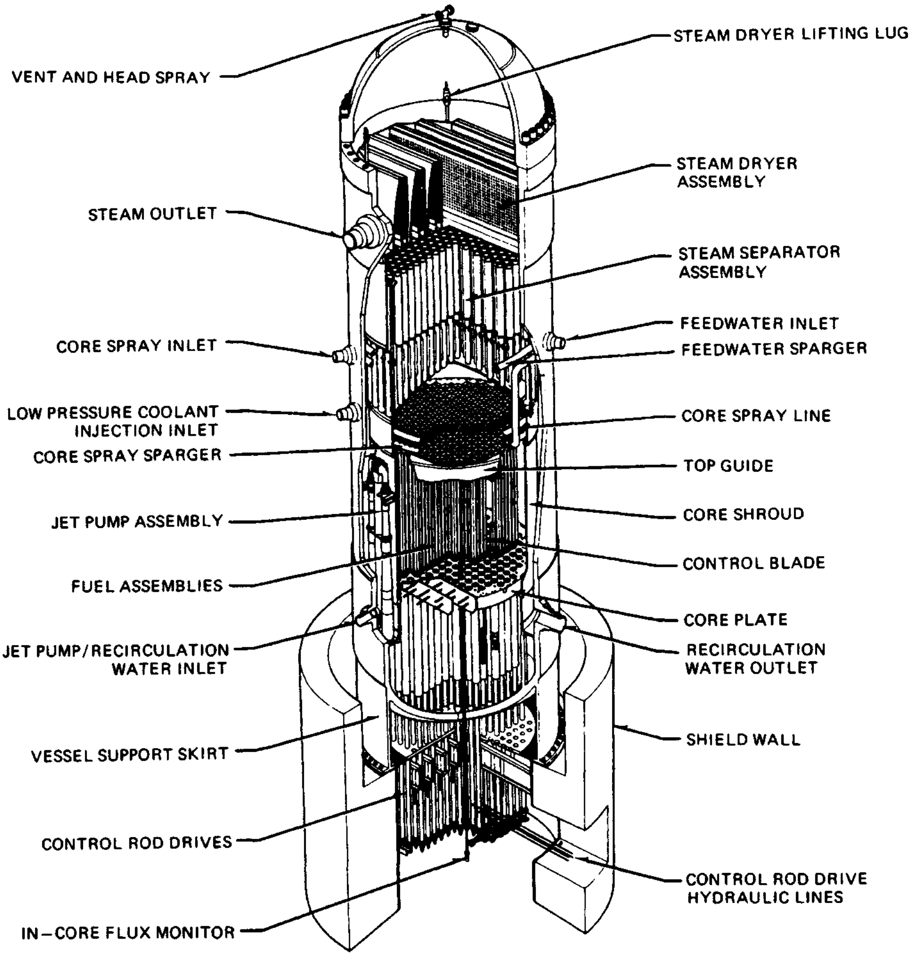


Fig. 56.3 Typical BWR vessel and core configuration. (Courtesy General Electric.)

This combination of technical, economic, and political constraints places a severe burden on those working to develop this important alternative source of energy.

56.4.5 Availability

A significant determinant in the cost of electrical energy produced by nuclear power plants is the plant capacity factor. The capacity factor is defined as a fraction calculated by dividing actual energy production during some specified time period by the amount that would have been produced by continuous power production at 100% of plant capacity. Many of the early estimates of power cost for nuclear plants were made with the assumption of a capacity factor of 0.80. Experience indicates an average for U.S. power plants of about 0.60. The contribution of capital costs to energy production has thus been more than 30% higher than the early estimates. Since capital costs typically represent anywhere between about 40%–80% (depending on when the plant was constructed) of the total energy cost, this difference in goal and achievement is a significant factor in some of the recently observed cost increases for electricity produced by nuclear power. Examination of the experience of individual plants reveals a wide range of capacity factors. A few U.S. plants have achieved a cumulative capacity factor near 0.80. Some have capacity factors as low as 0.40. There is reason to believe that improvements can be made in many of those with low capacity factors. It should also be possible to go beyond 0.80. Capacity factor improvement is a fruitful area for better resource utilization and realization of lower energy costs.

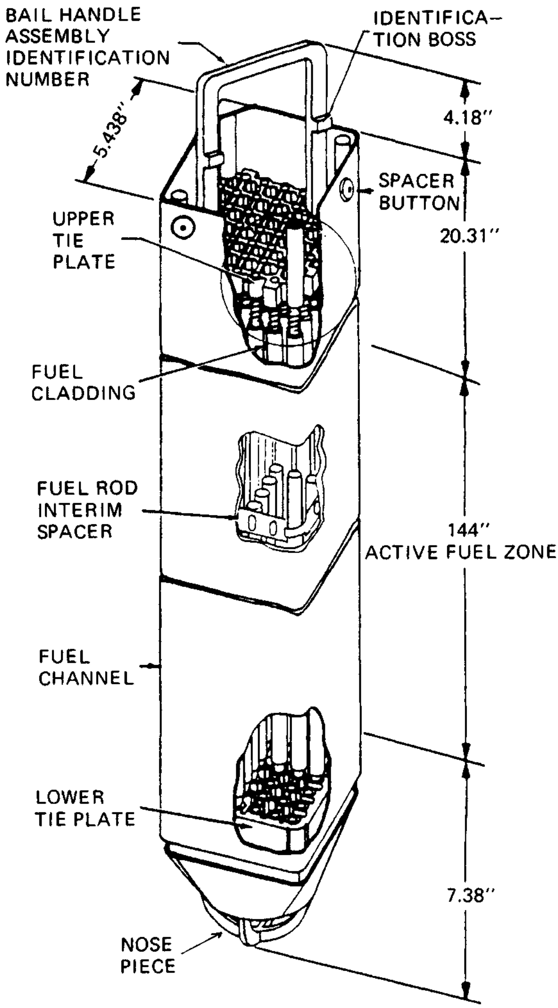


Fig. 56.4 BWR fuel assembly.

56.5 POLICY

The Congress, in the 1954 amendment to the Atomic Energy Act, made the development of nuclear power national policy. Responsibility for ensuring safe operation of nuclear power plants was originally given to the Atomic Energy Commission. In 1975 this responsibility was turned over to a Nuclear Regulatory Commission (NRC), set up for this purpose as an independent federal agency. Nuclear power is the most highly regulated of all the existing sources of energy. Much of the regulation is at the federal level. However, nuclear power plants and their operators are subject to a variety of state and local regulations as well. Under these circumstances nuclear power is of necessity highly responsive to any energy policy that is pursued by the federal government, or of local branches of government, including one of bewilderment and uncertainty.

56.5.1 Safety

The principal safety concern is the possibility of exposure of people to the radiation produced by the large (in terms of radioactivity) quantity of radioactive material produced by the fissioning of the reactor fuel. In normal operation of a nuclear power plant all but a minuscule fraction of this material is retained within the reactor fuel and the pressure vessel. Significant exposure of people outside the plant can occur only if a catastrophic and extremely unlikely accident should release a large fraction

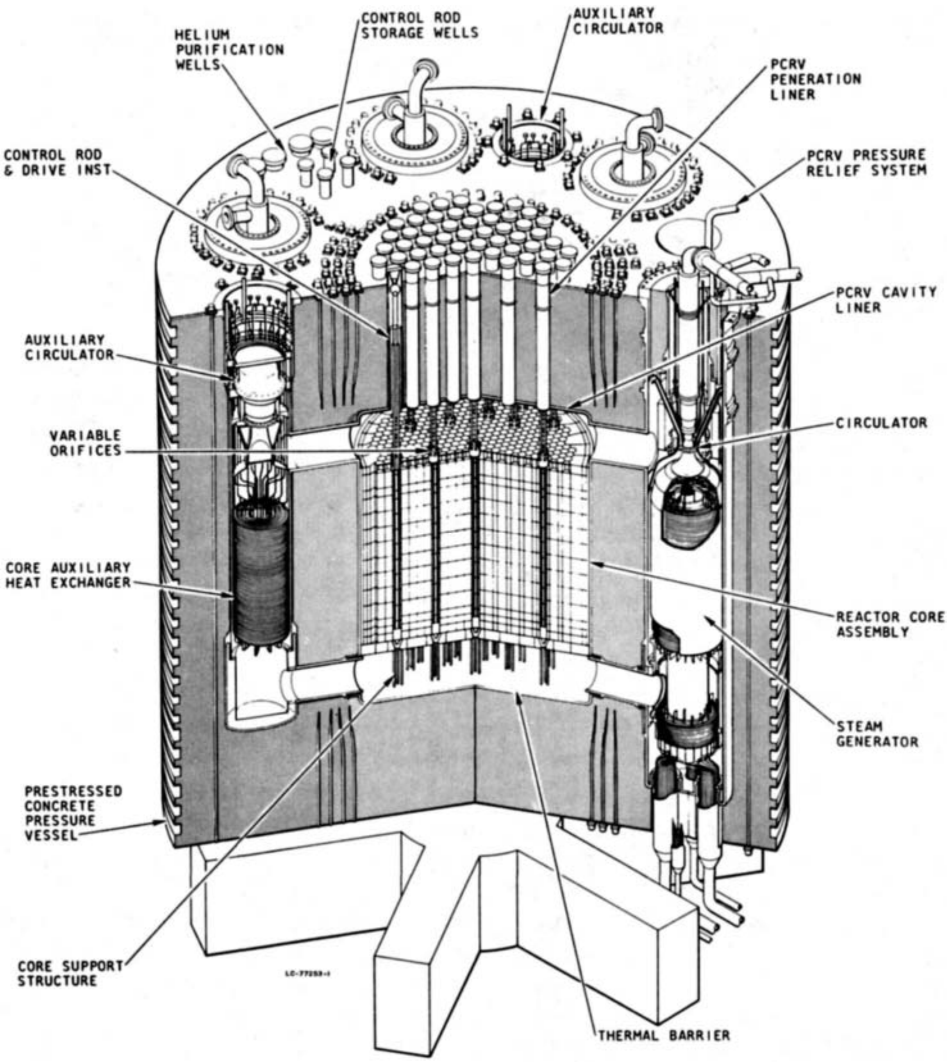


Fig. 56.5 HTGR pressure vessel and core arrangement. (Used by permission of Marcel Dekker, Inc., New York.)

of the radioactive fission products from the pressure vessel and from the surrounding containment system, and if these radioactive materials are then transported to locations where people are exposed to their radiation.

The uranium eventually used in reactor fuel is itself radioactive. The radioactive decay process, which begins with uranium, proceeds to produce several radioactive elements. One of these, radon-226, is a gas and can thus be inhaled by uranium miners. Hence, those who work in the mines are exposed to some hazard. Waste products of the mining and milling of uranium are also radioactive. When stored or discarded above ground, these wastes subject those in the vicinity to radon-226 exposure. These wastes or mill tailings must be dealt with to protect against this hazard. One method of control involves covering the wastes with a layer of some impermeable material such as asphalt.

The fresh fuel elements are also radioactive because of the contained uranium. However, the level of radioactivity is sufficiently low that the unused fuel assemblies can be handled safely without shielding.

56.5.2 Disposal of Radioactive Wastes

The used fuel from a power reactor is highly radioactive, although small in volume. The spent fuel produced by a year's operation of a 1000-MWe plant typically weighs about 40 tons and could be stored in a cube less than 5 ft on a side. It must be kept from coming in contact with people or other living organisms for long periods of time. (After 1000 years of storage the residual radioactivity of the spent fuel is about that of the original fresh fuel.) This spent fuel, or the radioactive residue that remains if most of the unused uranium and the plutonium generated during operation are chemically separated, is called high-level radioactive waste. Up to the present a variety of considerations, many of them political, have led to postponement of a decision on the choice of a permanent storage method for this material. The problem of safe storage has several solutions that are both technically and economically feasible. Technical solutions that currently exist include aboveground storage in air-cooled metal cannisters (for an indefinite period if desirable, with no decrease of safety over the period), as well as permanent disposal in deep strata of salt or of various impermeable rock formations. There have also been proposals to place the radioactive materials in deep ocean caverns. This method, although probably technically possible, is not yet developed. It would require international agreements not now in place. As indicated earlier, an operating plant also generates radioactive material in addition to fission products. Some of this becomes part of the various process streams that are part of the plant's auxiliary systems. These materials are typically removed by filters or ion-exchange systems, leaving filters or ion-exchange resins that contain radioactive materials. Tools, gloves, clothing, paper, and other materials may become slightly contaminated during plant operation. If the radioactive contamination has a half-life of more than a few weeks, these materials, described as low-level radioactive waste, must be stored or disposed of. The currently used disposal method involves burial in comparatively shallow trenches. Because of insufficient attention having been given to design and operation of some of the earlier burial sites, small releases of radioactive material have been observed. Several early burial sites are no longer in operation. Current federal legislation provides for compacts among several states that could lead to cooperative operation, by these states, of burial sites for low-level waste.

56.5.3 Economics

Nuclear power plants that began operation in the 1970s produce power at a cost considerably less than coal-burning plants of the same era. The current cost of power produced by oil-burning plants is two to three times as great as that produced by these nuclear plants. Nuclear power plants coming on line in the 1980s are much more expensive in capital cost (in some cases by a factor of 10!). The cost of the power they produce will be correspondingly greater. The two major contributors to the cost increase are high interest rates and the long construction period that has been required for most of these plants. Average construction time for plants now coming on line is about 11 years! It is likely that construction times can be decreased for new plants. The changes that were required as a result of the TMI accident have now been incorporated into regulations, into existing plants, and into new designs, eliminating the costly and time-consuming back fits that were required for plants under construction when the accident occurred. In Japan the average construction time for nuclear power plants is about 54 months. In Russia it is said to be about 77 months.

Standard plants are being designed and licensed that should make the licensing of an individual plant much faster and less involved. Concern over the pollution of the ecosphere caused by fossil-fueled plants (acid rain, CO_2) will call for additional pollution control, which will drive up costs of construction and operation of these plants. It is reasonable to expect nuclear power to be economically competitive with alternative methods of electric power generation in both the near and longer term.

56.5.4 Environmental Considerations

The environmental pollution produced by an operating nuclear power plant is far less than that caused by any other currently available method of producing electric power. The efficiency of the thermodynamic cycle for water reactors is lower than that of modern fossil-fuel plants because current design of reactor pressure vessels limits the steam temperature. Thus, the amount of waste heat rejected is greater for a nuclear plant than for a modern fossil-fuel plant of the same rated power. However, current methods of waste heat rejection (typically cooling towers) handle this with no particular environmental degradation. Nuclear power plants emit no carbon dioxide, no sulfur, no nitrous oxides. No large coal storage area is required. The tremendous volumes of sulfur compounds removed during coal combustion and the enormous quantities of ash produced by coal plants are problems with which those who operate nuclear plants do not have to deal.

Table 56.2 provides a comparison of emissions and wastes from a large coal-burning plant and from a nuclear power plant of the same rated power. Although there is a small release of radioactive material to the biosphere from the nuclear power plant, the resulting increase in exposure to a member of the population in the immediate vicinity of the plant is typically about 1% of that produced by naturally occurring background radiation.

56.5.5 Proliferation

Nuclear power plants are thought by some to increase the probability of nuclear weapons proliferation. It is true that a country with the trained engineers and scientists, the facilities, and the resources required to produce nuclear power can develop a weapons capability more rapidly than one without

Table 56.2 Waste Material from Different Types of 1000-MWe Power Plants (Capacity Factor = 0.8)

	Coal Fired	Water Reactor
Typical thermal efficiency, %	39	32
Thermal wastes (in thermal megawatts)		
To cooling water	1,170	1,970
To atmosphere	400	150
Total	1,570	2,120
Solid wastes		
Fly ash or slag, tons/year	330,000	0
cubic feet/year	7,350,000	0
railroad carloads/year	3,300	0
Radioactive wastes		
Fuel to reprocessing plant,		
assemblies/year	0	160
railroad carloads/year	0	5
Solid waste storage		
From reprocessing plant, cubic feet/year	0	100
From power plant, cubic feet/year	0	5,000
Gaseous and liquid wastes[a]		
(tons per day/10^6 cubic feet per day)		
Carbon monoxide	2/8	0
Carbon dioxide	21,000/53,200	0
Sulfur dioxide: 1% sulfur fuel	140/325	0
2.5% sulfur fuel	350/812	0
Nitrogen oxides	82/305	0
Particulates to atmosphere (tons/day)	0.4	0
Radioactive gases or liquids, equivalent dose mrem/year at plant boundary	Minor	5

[a]For 3,000,000 tons/year coal total ash content of 11%, fly ash precipitator efficiency of 99.5%, and 15% of sulfur remaining in ash.

this background. However, for a country starting from scratch, the development of nuclear power is a detour that would consume needless time and resources. None of the countries that now possess nuclear weapons capability has used the development of civil nuclear power as a route to weapons development.

Nevertheless, it must be recognized that plutonium, an important constituent of weapons, is produced in light-water nuclear power plants. Plutonium is the preferred fuel for breeder reactors. The development of any significant number of breeder reactors would thus involve the production and handling of large quantities of plutonium.

As will be discussed in a later section, plutonium-239 can be produced by the absorption of a neutron in uranium-238. Since most of the uranium in the core of an LWR is uranium-238, plutonium is produced during operation of the reactor. However, if the plutonium-239 is left in a power reactor core for the length of time typical of the fuel cycle used for LWRs or for breeders, neutrons are absorbed by some fraction of the plutonium to produce plutonium-240. This isotope also absorbs neutrons to produce plutonium-241. These heavier isotopes make the plutonium undesirable as weapons material. Thus, although the plutonium produced in power reactors can be separated chemically from the other materials in a used fuel element, it is not what would be considered weapons-grade material. A nation with the goal of developing weapons would almost certainly design and use a reactor and a fuel cycle designed specifically for producing weapons-grade material. On the other hand, if a drastic change in government produced a correspondingly drastic change in political objectives in a country that had a civil nuclear power program in operation, it would probably be possible to make use of power reactor plutonium to produce some sort of low-grade weapon.

56.6 BASIC ENERGY PRODUCTION PROCESSES

Energy can be produced by nuclear reactions that involve either fission (the splitting of a nucleus) or fusion (the fusing of two light nuclei to produce a heavier one). If energy is to result from fission, the resultant nuclei must have a smaller mass per nucleon (which means they are more tightly bound) than the original nucleus. If the fusion process is to produce energy, the fused nucleus must have a

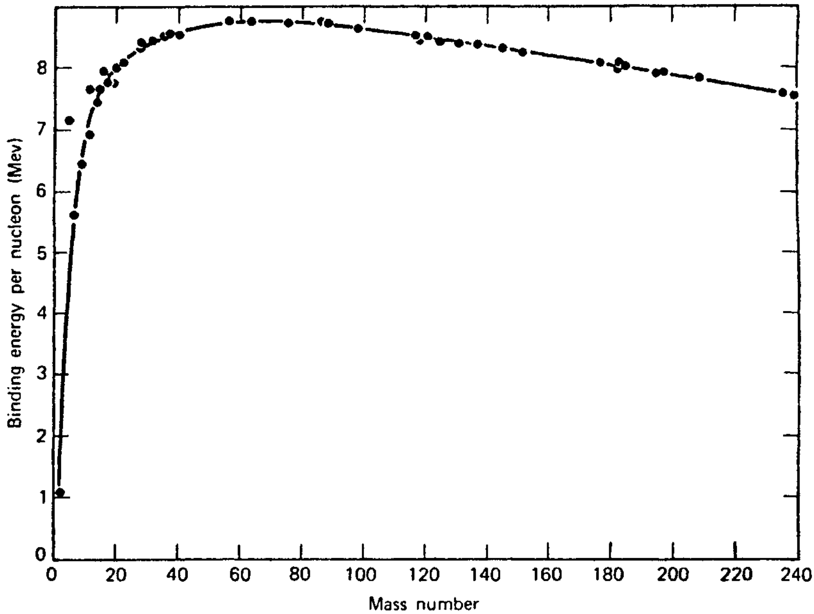


Fig. 56.6 Binding energy per nucleon versus mass number.

smaller mass per nucleon (i.e., be more tightly bound) than the original nuclei. Figure 56.6 is a curve of nuclear binding energies. Observe that only the heavy nuclei are expected to produce energy on fission, and that only the light nuclei yield energy in fusion. The differences in mass per nucleon before and after fission or fusion are available as energy.

56.6.1 Fission

In the fission process this energy is available primarily as kinetic energy of the fission fragments. Gamma rays are also produced as well as a few free neutrons, carrying a small amount of kinetic energy. The radioactive fission products decay (in most cases there is a succession of decays) to a stable nucleus. Gamma and beta rays are produced in the decay process. Most of the energy of these radiations is also recoverable as fission energy. Table 56.3 lists typical energy production due to fission of uranium by thermal neutrons, and indicates the form in which the energy appears. The quantity of energy available is of course related to the nuclear mass change by

$$\Delta E = \Delta mc^2$$

Table 56.3 Emitted and Recoverable Energies from Fission of ^{235}U

Form	Emitted Energy (MeV)	Recoverable Energy (MeV)
Fission fragments	168	168
Fission product decay		
β rays	8	8
γ rays	7	7
Neutrinos	12	—
Prompt γ rays	7	7
Fission neutrons (kinetic energy)	5	5
Capture γ rays	—	3–12
Total	207	198–207

Fission in reactors is produced by the absorption of a neutron in the nucleus of a fissionable atom. In order to produce significant quantities of power, fission must occur as part of a sustained chain reaction, that is, enough neutrons must be produced in the average fission event to cause at least one new fission event to occur when absorbed in fuel material. The number of nuclei that are available and that have the required characteristics to sustain a chain reaction is limited to uranium-235, plutonium-239, and uranium-233. Only uranium-235 occurs in nature in quantities sufficient to be useful. (And it occurs as only 0.71% of natural uranium.) The other two can be manufactured in reactors. The reactions are indicated below:

$$^{238}\mathrm{U} + n \rightarrow {}^{239}\mathrm{U} \rightarrow {}^{239}\mathrm{Np} \rightarrow {}^{239}\mathrm{Pu}$$

Uranium-239 has a half-life of 23.5 min. It decays to produce neptunium-239, which has a half-life of 2.35 days. The neptunium-239 decays to plutonium, which has a half-life of about 24,400 years.

$$^{232}\mathrm{Th} + n \rightarrow {}^{233}\mathrm{Th} \rightarrow {}^{233}\mathrm{Pa} \rightarrow {}^{233}\mathrm{U}$$

Thorium-233 has a half-life of 22.1 min. It decays to protactinium-233, which has a half-life of 27.4 days. The protactinium decays to produce uranium-233 with a half-life of about 160,000 years.

56.6.2 Fusion

Fusion requires that two colliding nuclei have enough kinetic energy to overcome the Coulomb repulsion of the positively charged nuclei. If the fusion rate is to be useful in a power-producing system, there must also be a significant probability that fusion-producing collisions occur. These conditions can be satisfied for several combinations of nuclei if a collection of atoms can be heated to a temperature typically in the neighborhood of hundreds of millions of degrees and held together for a time long enough for an appreciable number of fusions to occur. At the required temperature the atoms are completely ionized. This collection of hot, highly ionized particles is called a plasma. Since average collision rate can be related to the product of the density of nuclei, n, and the average containment time, τ, the $n\ \tau$ product for the contained plasma is an important parameter in describing the likelihood that a working system with these plasma characteristics will produce a useful quantity of energy.

Examination of the fusion probability, or the cross section for fusion, as a function of the temperature of the hot plasma shows that the fusion of deuterium (^{2}H) and tritium (^{3}H) is significant at temperatures lower than that for other candidates. Figure 56.7 shows fusion cross section as a function of plasma temperature (measured in electron volts) for several combinations of fusing nuclei. Table 56.4 lists several fusion reactions that might be used, together with the fusion products and the energy produced per fusion.

One of the problems with using the D–T reaction is the large quantity of fast neutrons that results, and the fact that a large fraction of the energy produced appears as kinetic energy of these neutrons. Some of the neutrons are absorbed in and activate the plasma-containment-system walls, making it highly radioactive. They also produce significant damage in most of the candidate materials for the containment walls. For these reasons there are some who advocate that work with the D–T reaction be abandoned in favor of the development of a system that depends on a set of reactions that is neutron-free.

Another problem with using the D–T reaction is that tritium does not occur in nature in sufficient quantity to be used for fuel. It must be manufactured. Typical systems propose to produce tritium by the absorption in lithium of neutrons resulting from the fusion process. Natural lithium consists of ^{6}Li (7.5%) and ^{7}Li (92.5%). The reactions are

$$^{6}\mathrm{Li} + n \rightarrow {}^{4}\mathrm{He} + {}^{3}\mathrm{H} + 4.8\ \mathrm{MeV}\ \text{(thermal neutrons)}$$

and

$$^{7}\mathrm{Li} + n \rightarrow {}^{4}\mathrm{He} + {}^{3}\mathrm{H} + n + 2.47\ \mathrm{MeV}\ \text{(threshold reaction)}$$

Considerations of neutron economy dictate that most of the neutrons produced in the fusion process be absorbed in lithium in order to breed the needed quantities of tritium. The reactions shown produce not only tritium, but also additional energy. The (^{6}Li,n) reaction, for example, produces 4.7 MeV per reaction. If this energy can be recovered, it effectively increases the average available energy per fusion by about 27%.

56.7 CHARACTERISTICS OF THE RADIATION PRODUCED BY NUCLEAR SYSTEMS

An important by-product of the processes used to generate nuclear power is a variety of radiations in the form of either particles or electromagnetic photons. These radiations can produce damage in

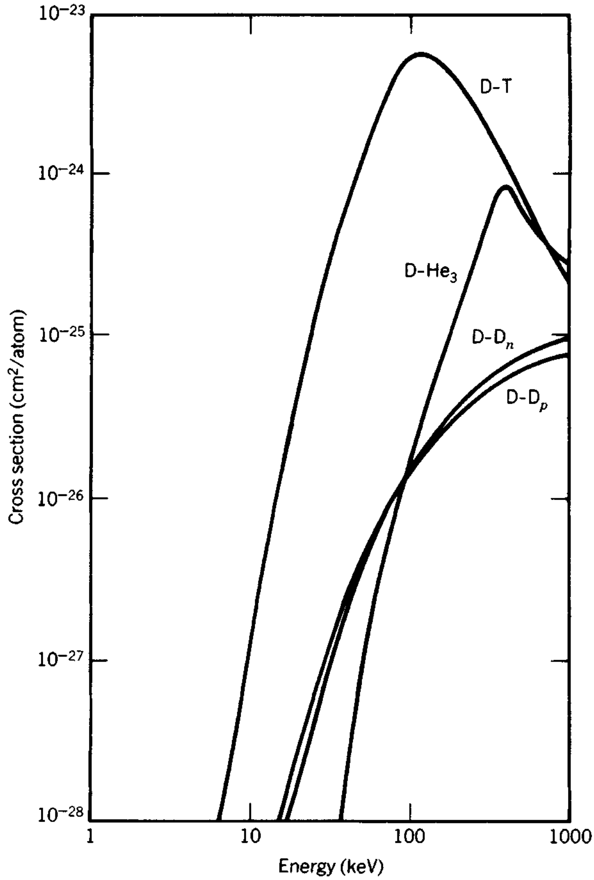


Fig. 56.7 Fusion cross section versus plasma temperature.

the materials that make up the systems and structures of the power reactors. High-energy neutrons, for example, absorbed in the vessel wall make the steel in the pressure vessel walls less ductile.

Radiation also causes damage to biological systems, including humans. Thus, most of the radiations must be contained within areas from which people are excluded. Since the ecosystem to which humans are normally exposed contains radiation as a usual constituent, it is assumed that some additional exposure can be permitted without producing undue risk. However, since the best scientific

Table 56.4 Fusion Reactions

$^{3}H + {}^{2}H \rightarrow n + {}^{4}He$	+17.6 MeV
$^{3}He + {}^{2}H \rightarrow p + {}^{4}He$	+18.4 MeV
$^{2}H + {}^{2}H \rightarrow p + {}^{3}H$	+4.0 MeV
$^{2}H + {}^{2}H \rightarrow n + {}^{3}He$	+3.3 MeV
$^{6}Li + p \rightarrow {}^{3}He + {}^{4}He$	+4.0 MeV
$^{6}Li + {}^{3}He \rightarrow {}^{4}He + p + {}^{4}He$	+16.9 MeV
$^{6}Li + {}^{2}H \rightarrow p + {}^{7}Li$	+5.0 MeV
$^{6}Li + {}^{2}H \rightarrow {}^{3}H + p + {}^{4}He$	+2.6 MeV
$^{6}Li + {}^{2}H \rightarrow {}^{4}He + {}^{4}He$	+22.4 MeV
$^{6}Li + {}^{2}H \rightarrow n + {}^{7}Be$	+3.4 MeV
$^{6}Li + {}^{2}H \rightarrow n + {}^{3}He + {}^{4}He$	+1.8 MeV

judgment concludes that there is likely to be some risk of increasing the incidence of cancer and of other undesirable consequences with any additional exposure, the amount of additional exposure permitted is small and is carefully controlled, and an effort is made to balance the permitted exposure against perceived benefits.

56.7.1 Types of Radiation

The principal types of radiation encountered in connection with the operation of fission and fusion systems are listed in Table 56.5. Characteristics of the radiation, including its charge and energy spectrum, are also given.

Alpha particles are produced by radioactive decay of all of the fuels used in fission reactors. They are, however, absorbed by a few millimeters of any solid material and produce no damage in typical fuel material. They are also a product of some fusion systems and may produce damage to the first wall that provides a plasma boundary. They may produce damage to human lungs during the mining of uranium when radioactive radon gas may be inhaled and decay in the lungs. In case of a catastrophic fission reactor accident, severe enough to generate aerosols from melted fuels, the alpha-emitting materials in the fuel might, if released from containment, be ingested by those in the vicinity of the accident, thus entering both the lungs and the digestive system.

Beta particles are produced by radioactive decay of many of the radioactive substances produced during fission reactor operation. The major source is fission products. Although more penetrating than alphas, betas produced by fission products can typically be absorbed by at most a few centimeters of most solids. They are thus not likely to be harmful to humans except in case of accidental release and ingestion of significant quantities of radioactive material. A serious reactor accident might also release radioactive materials to a region in the plant containing organic materials such as electrical insulation. A sufficient exposure to high-energy betas can produce damage to these materials. Reactor systems needed for accident amelioration must be designed to withstand such beta irradiations.

Gamma rays are electromagnetic photons produced by radioactive decay or by the fission process. Photons identical in characteristics (but not in name) are produced by decelerating electrons or betas. When produced in this way, the electromagnetic radiation is usually called X rays. High-energy (above several hundred keV) gammas are quite penetrating, and protection of both equipment and people requires extensive (perhaps several meters of concrete) shielding to prevent penetration of significant quantities into the ecosystem or into reactor components or systems that may be subject to damage from gamma absorption.

Neutrons are particles having about the same mass as that of the hydrogen nucleus or proton, but with no charge. They are produced in large quantities by fission and by some fusion interactions including the D-T fusion referred to earlier. High-energy (several MeV) neutrons are highly penetrating. They can produce significant biological damage. Absorption of fast neutrons can induce a decrease in the ductility of steel structures such as the pressure vessel in fission reactors or the inner wall of fusion reactors. Fast-neutron absorption also produces swelling in certain steel alloys.

56.8 BIOLOGICAL EFFECTS OF RADIATION

Observations have indicated that the radiations previously discussed can cause biological damage to a variety of living organisms, including humans. The damage that can be done to human organisms includes death within minutes or weeks if the exposure is sufficiently large, and if it occurs during an interval of minutes or at most a few hours.

Radiation exposure has also been found to increase the probability that cancer will develop. It is considered prudent to assume that the increase in probability is directly proportional to exposure. However, there is evidence to suggest that at very low levels of exposure, say an exposure comparable to that produced by natural background, the linear hypothesis is not a good representation. Radiation exposure has also been found to induce mutations in a number of biological organisms. Studies of the survivors of the two nuclear weapons exploded in Japan have provided the largest body of data

Table 56.5 Radiation Encountered in Nuclear Power Systems

Name	Description	Charge (in Units of Electron Charge)	Energy Spectrum (MeV)
Alpha	Helium nucleus	+2	0 to about 5
Beta	Electron	+1, −1	0 to several
Gamma	Electromagnetic radiation	0	0 to about 10
Neutron		0	0 to about 20

available for examining the question of whether harmful mutations are produced in humans by exposure of their forebears to radiation. Analyses of these data have led those responsible for the studies to conclude that the existence of an increase in harmful mutations has not been demonstrated unequivocally. However, current regulations of radiation exposure, in order to be conservative, assume that increased exposure will produce an increase in harmful mutations. There is also evidence to suggest that radiation exposure produces life shortening.

The Nuclear Regulatory Commission has the responsibility for regulating exposure due to radiation produced by reactors and by radioactive material produced by reactors. The standards used in the regulatory process are designed to restrict exposures to a level such that the added risk is not greater than that from other risks in the workplace or in the normal environment. In addition, effort is made to see that radiation exposure is maintained as "low as reasonably achievable."

56.9 THE CHAIN REACTION

Setting up and controlling a chain reaction is fundamental to achieving and controlling a significant energy release in a fission system. The chain reaction can be produced and controlled if a fission event, produced by the absorption of a neutron, produces more than one additional neutron. If the system is arranged such that one of these fission-produced neutrons produces, on the average, another fission, there exists a steady-state chain reaction. Competing with fission for the available neutrons are leakage out of the fuel region and absorptions that do not produce fission.

We observe that if only one of these fission-neutrons produces another fission, the average fission rate will be constant. If more than one produces fission, the average fission rate will increase at a rate that depends on the average number of new fissions produced for each preceding fission and the average time between fissions.

Suppose, for example, each fission produced two new fissions. One gram of uranium-235 contains 2.56×10^{21} nuclei. It would therefore require about 71 generations ($2^{71} \sim 2.4 \times 10^{21}$) to fission 1 g of uranium-235. Since fission of each nucleus produces about 200 MeV, this would result in an energy release of about 5.12×10^{23} MeV or 5.12×10^{10} J. The time interval during which this release takes place depends on the average generation time. Note, however, that in this hypothesized situation only about the last 10 generations contribute any significant fraction of the total energy. Thus, for example, if a generation could be made as short as 10^{-8} sec, the energy production rate could be nearly 5.12×10^{17} J/sec/g.

In power reactors the generation time is typically much larger than 10^{-8} sec by perhaps four or five orders of magnitude. Furthermore, the maximum number of new fissions produced per old fission is much less than two. Power reactors (in contrast to explosive devices) cannot achieve the rapid energy release hypothesized in the above example, for the very good reasons that the generation time and the multiplication inherent in these machines make it impossible.

56.9.1 Reactor Behavior

As indicated, it is neutron absorption in the nuclei of fissile material in the reactor core that produces fission. Furthermore, the fission process produces neutrons that can generate new fissions. This process sustains a chain reaction at a fixed level, if the relationship between neutrons produced by fission and neutrons absorbed in fission-producing material can be maintained at an appropriate level.

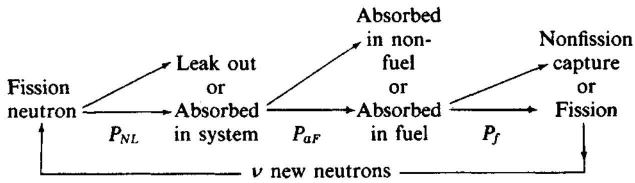
One can define neutron multiplication k as

$$k = \frac{\text{neutrons produced in a generation}}{\text{neutrons produced in the preceding generation}}$$

A reactor is said to be critical when k is 1. We examine the process in more detail by following neutron histories. The probability of interaction of neutrons with the nuclei of some designated material can be described in terms of a mean free path for interaction. The inverse, which is the interaction probability per unit path length, is also called macroscopic cross section. It has dimensions of inverse length.

We designate a cross section for absorption, Σ_a, a cross section for fission, Σ_f, and a cross section for scattering, Σ_s. If, then, we know the number of path lengths per unit time, per unit volume, traversed by neutrons in the reactor (for monoenergetic neutrons this will be nv, where n is neutron density and v is neutron speed), usually called the neutron flux, we can calculate the various interaction rates associated with these cross sections and with a prescribed neutron flux, as a product of the flux and the cross section.

A diagrammatic representation of neutron history, with the various possibilities that are open to the neutrons produced in the fission process, is shown below:



where P_{NL} = probability that neutron will not leak out of system before being absorbed
P_{aF} = probability that a neutron absorbed is absorbed in fuel
P_f = probability that a neutron absorbed in fuel produces a fission

In terms of the cross sections for absorption in fuel, Σ_a^F and for absorption, Σ_a

$$P_{aF} = f = \Sigma_a^F/\Sigma_a$$

where f is called the utilization factor. We can describe P_f as

$$P_f = \Sigma_f^F/\Sigma_a^F$$

Making use of the average number of neutrons produced per fission ν, we calculate the quantity η, the average number of neutrons produced per neutron absorbed in fuel, as

$$\eta = \nu\, \Sigma_f^F/\Sigma_a^F$$

With these definitions, and guided by the preceding diagram, we conclude that the number of offspring neutrons produced by a designated fission neutron can be calculated as

$$N = \eta f P_{NL}$$

We conclude that the multiplication factor k is thus equal to $N/1$ and write

$$k = \eta f P_{NL}$$

Alternatively, making use of the earlier definitions we write

$$k = (\upsilon\Sigma_f^F/\Sigma_a^F)(\Sigma_a^F/\Sigma_a)$$

and if we describe Σ_a as

$$\Sigma_a = \Sigma_a^F + \Sigma_a^{nF}$$

where Σ_a^{nF} is absorption in the nonfuel constituents of the core, we have

$$k = \nu\Sigma_f^F\, P_{NL}/(\Sigma_a^F + \Sigma_a^{nF})$$

Observe that from this discussion one can also define a neutron generation time l as

$$l = N(t)/L(t)$$

where $N(t)$ and $L(t)$ represent, respectively, the neutron population and the rate of neutron loss (through absorption and leakage) at a time t.

For large reactors, the size of those now in commercial power production, the nonleakage probability is high, typically about 97%. For many purposes it can be neglected. For example, small changes in multiplication, produced by small changes in concentration of fissile or nonfissile material in the core, can be assumed to have no significant effect on the nonleakage probability, P_{NL}. Under these circumstances, and assuming that appropriate cross-sectional averaging can be done, the following relationships can be shown to hold. If we rewrite an earlier equation for k as

$$k = \eta f \left(n_f\sigma_f \Big/ \sum_i n_i\sigma_i \right)$$

where η_f, η_i represent, respectively, the concentration of fissile and nonfissile materials, and

$$\eta_f \sigma_f = \Sigma_f$$
$$n_i \sigma_i = \Sigma_{a_i}$$

where the last equation in the macroscopic cross section of the ith nonfissile isotope.

Variation of k with the variation in concentration of the fissile material (i.e., n_f) is given by

$$\frac{\delta k}{k} = \frac{\delta n_f}{n_f} \left(\frac{\Sigma_i n_i \sigma_i}{\eta_f \sigma_f + \Sigma_i \eta_i \sigma_i} \right)$$

This says that the fractional change in multiplication is equal to the fractional change in concentration of the fissile isotope times the ratio of the neutrons absorbed in all of the nonfissile isotopes to the total neutrons absorbed in the core.

Variation of k with variation in concentration of the jth nonfissile isotope is given by

$$\frac{\delta k}{k} = -\frac{\delta m_j}{m_j} \left(\frac{n_j \sigma_j}{n_f \sigma_f + \Sigma_i n_i \sigma_i} \right), \qquad i \neq j$$

This says that the fractional increase in multiplication is equal to the fractional decrease in concentration of the jth nonfissile isotope times the ratio of neutrons absorbed in that isotope to total neutrons absorbed. Although these are approximate expressions, they provide useful guidance in estimating effects of small changes in the material in the core on neutron multiplication.

56.9.2 Time Behavior of Reactor Power Level

We assume that the fission rate, and hence the reactor power, is proportional to neutron population. We express the rate of change of neutron population $N(t)$ as

$$l \left(\frac{dN}{dt} \right) = (k - 1)N$$

that is, in one generation the change in neutron population should be just the excess over the previous generation, times the multiplication k. The preceding equation has as a solution

$$N = N_0 \exp \left[\left(\frac{k-1}{l} \right) t \right]$$

where N_0 is the neutron population at time zero. One observes an exponential increase or decrease, depending on whether k is larger or smaller than unity. The associated time constant or e-folding time is

$$\tau = l/(k - 1)$$

For a $k - 1$ of 0.001 and l of 10^{-4} sec, the e-folding time is 0.10 sec. Thus in 1 sec the power level increases by e^{10} or about 10^4.

56.9.3 Effect of Delayed Neutrons on Reactor Behavior

Dynamic behavior as rapid as that described by the previous equations would make a reactor almost impossible to control. Fortunately there is a mode of operation in which the time constant is significantly greater than that predicted by these oversimplified equations. A small fraction of neutrons produced by fission, typically about 0.7%, come from radioactive decay of fission products. Six such fission products are identified for uranium fission. The mean time for decay varies from about 0.3 to about 79 sec. For an approximate representation, it is reasonable to assume a weighted mean time to decay for the six of about 17 sec. Thus, about 99.3% of the neutrons (prompt neutrons) may have a generation time of, say, 10^{-4} sec, while 0.7% have an effective generation time of 17 sec plus that of the prompt neutrons. An effective mean lifetime can be estimated as

$$\bar{l} = (0.993)l + 0.007(l + \bar{\lambda}^{-1})$$

For an l of 10^{-4} sec and a $\bar{\lambda}^{-1}$ of 17 sec we calculate

$$l \simeq 0.993 \times 10^{-4} + 0.007(10^{-4} + 17) \simeq 0.12 \text{ sec}$$

This suggests, given a value for $k - 1$ of 10^{-3}, an e-folding time of 120 sec. Observe that with this model the delayed neutrons are a dominant factor in determining time behavior of the reactor power level. A more detailed examination of the situation reveals that for a reactor slightly subcritical on prompt neutrons alone, but supercritical when delayed neutrons are considered (such a reactor is said to be "delayed supercritical"), the delayed neutrons almost alone determine time behavior. If, however, the multiplication is increased to the point that the reactor is critical on prompt neutrons alone (i.e., "prompt critical"), the time behavior is determined by the prompt neutrons, and changes in power level may be too rapid to be controlled by any external control system. Reactors that are meant to be controlled are designed to be operated in a delayed critical mode. Fortunately, if the reactor should inadvertently be put in a prompt critical mode, there are inherent physical phenomena that decrease the multiplication to a controllable level when a power increase occurs.

56.10 POWER PRODUCTION BY REACTORS

Most of the nuclear-reactor-produced electric power in the United States, and in the rest of the world, comes from light-water-moderated reactors (LWRs). Nuclear power reactors produce heat that is converted, in a thermodynamic cycle, to electrical energy. The two types now in use, the pressurized water reactor (PWR) and the boiling water reactor (BWR), use fuel that is very similar, and produce steam having about the same temperature and pressure. In both types water serves both as a coolant and a moderator. We will examine some of the salient features of each system and identify some of the differences.

56.10.1 The Pressurized-Water Reactor

The arrangement of fuel in the reactor core, and of the core in the pressure vessel, are shown in Fig. 56.1. As indicated earlier, bulk boiling is avoided by operation at high pressures. Liquid water is circulated through the core by large electric-motor-driven pumps located outside the pressure vessel in the cold leg of the piping that connects the vessel to a steam generator. Current designs use from two to four separate loops, each containing a steam generator. Each loop contains at least one pump. One current design uses two pumps in the cold leg of each loop. A schematic of the arrangement is shown in Fig. 56.8. Reactor pressure vessel and primary coolant loops, including the steam generator, are located inside a large containment vessel. A typical containment structure is shown in Fig. 56.9.

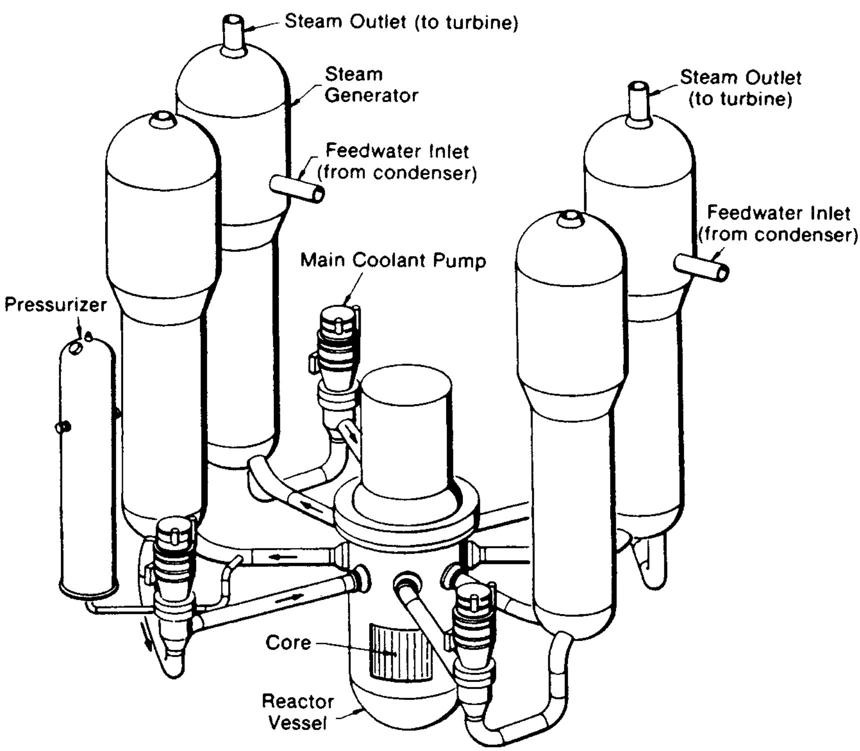


Fig. 56.8 Typical arrangement of PWR primary system.

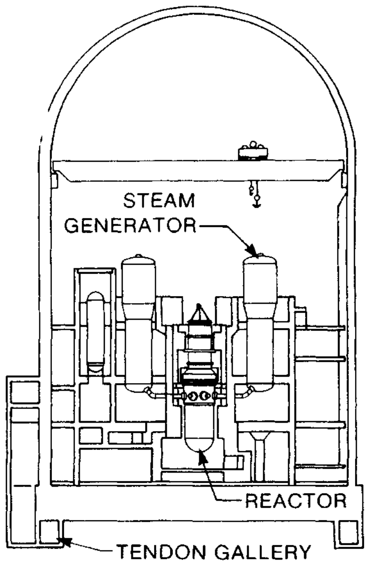


Fig. 56.9 Typical large dry PWR containment.

The containment, typically a massive 3–4-ft-thick structure of reinforced concrete, with a steel inner-liner, has two principal functions: protection of pressure vessel and primary loop from external damage (e.g., tornadoes, aircraft crashes) and containment of fission products that might be released outside the primary pressure boundary in case of serious damage to the reactor in an accident.

The steam generator is markedly different from the boiler in a fossil-fueled plant. It is essentially a heat exchanger containing several thousand metal tubes that carry the hot water coming from the reactor vessel outlet. Water surrounding the outside of these tubes is converted to steam. The rest of the energy-conversion cycle is similar in principle to that found in a fossil-fueled plant.

Experience with reactor operation has indicated that very careful control of water chemistry is necessary to preclude erosion and corrosion of the steam generator tubes (SGT). An important contributor to SGT damage has been leakage in the main condenser, which introduces impurities into the secondary water system. A number of early PWRs have retubed or otherwise modified their original condensers to reduce contamination caused by in-leakage of condenser cooling water.

The performance of steam generator tubes is of crucial importance because: (1) These tubes are part of the primary pressure boundary. SGT rupture can initiate a loss of coolant accident. (2) Leakage or rupture of SGTs usually leads to opening of the steam system safety valves because of the high primary system pressure. Since these valves are located outside containment, this accident sequence can provide an uncontrolled path for release of any radioactive material in the primary system directly to the atmosphere outside containment.

The reactor control system controls power level by a combination of solid control rods, containing neutron-absorbing materials that can be moved into and out of the core region, and by changing the concentration of a neutron-absorbing boron compound (typically boric acid) in the primary coolant. Control rod motion is typically used to achieve rapid changes in power. Slower changes, as well as compensation for burnup of uranium-235 in the core, are accomplished by boron-concentration changes. In the PWR the control rods are inserted from the top of the core. In operation enough of the absorber rods are held out of the core to produce rapid shutdown, or scram, when inserted. In an emergency, if rod drive power should be lost, the rods automatically drop into the core, driven by gravity.

The PWR is to some extent load following. Thus, for example, an increase in turbine steam flow caused by an increase in load produces a decrease in reactor coolant–moderator temperature. In the usual mode of operation a decrease in moderator temperature produces an increase in multiplication (the size of the effect depends on the boron concentration in the coolant-moderator), leading to an increase in power. The increase continues (accompanied by a corresponding increase in moderator-coolant temperature) until the resulting decrease in reactor multiplication leads to a return to criticality at an increased power level. Since the size of the effect changes significantly during the operating cycle (as fuel burnup increases the boron concentration is decreased), the inherent load-following characteristic must be supplemented by externally controlled changes in reactor multiplication.

A number of auxiliaries are associated with the primary. These include a water purification and makeup system, which also permits varying the boron concentration for control purposes, and an emergency cooling system to supply water for decay heat removal from the core in case of an accident that causes loss of the primary coolant.

Pressure in the primary is controlled by a pressurizer, which is a vertical cylindrical vessel connected to the hot leg of the primary system. In normal operation the bottom 60% or so of the pressurizer tank contains liquid water. The top 40% contains a steam bubble. System pressure can be decreased by water sprays located in the top of the tank. A pressure increase can be achieved by turning on electric heaters in the bottom of the tank.

56.10.2 The Boiling-Water Reactor

Fuel and core arrangement in the pressure vessel are shown in Fig. 56.3. Boiling in the core produces a two-phase mixture of steam and water, which flows out of the top of the core. Steam separators above the vessel water level remove moisture from the steam, which goes directly to the turbine outside of containment. Typically about one-seventh of the water flowing through the core is converted to steam during each pass. Feedwater to replace the water converted to steam is distributed around the inside near the top of the vessel from a spray ring. Water is driven through the core by jet pumps located in the annulus between the vessel wall and the cylindrical core barrel that surrounds the core and defines the upward flow path for coolant.

Because there is direct communication between the reactor core and the turbine, any radioactive material resulting, for example, from leakage of fission products out of damaged fuel pins, from neutron activation of materials carried along with the flow of water through the core, or from the nitrogen-16 referred to earlier, has direct access to turbine and condenser. Systems must be provided for removal from the coolant and for dealing with these materials as radioactive waste.

Unlike the PWR, the BWR is not load following. In fact, normal behavior in the reactor core produces an increase in the core void volume with an increase in steam flow to the turbine. This increased core voiding will increase neutron leakage, thereby decreasing reactor multiplication, and leading to a decrease in reactor power in case of increased demand. To counter this natural tendency of the reactor, a control system senses an increase in turbine steam flow and increases coolant flow through the core. The accompanying increase in core pressure decreases steam voiding, increasing multiplication and producing an increase in reactor power.

Pressure regulation in the BWR is achieved primarily by adjustment of turbine throttle setting to achieve constant pressure. An increase in load demand is sensed by the reactor control system and produces an increase in reactor power. Turbine valve position is adjusted to maintain constant steam pressure at the throttle. In rapid transients, which involve decreases in load demand, a bypass valve can be opened to send steam directly to the condenser, thus helping to maintain constant pressure.

BWRs in the United States make use of a pressure-suppression containment system. The hot water and steam released during a loss-of-coolant accident are forced to pass through a pool of water, condensing the steam. Pressure buildup is markedly less than if the two-phase mixture is released directly into containment. Figure 56.10 shows a Mark III containment structure of the type being used with the latest BWRs. Passing the fission products and the hot water and steam from the primary containment through water also results in significant removal of some of the fission products. The designers of this containment claim decontamination factors of 10,000 for some of the fission products that are usually considered important in producing radiation exposure following an accident.

As previously indicated, the control system handles normal load changes by adjusting coolant flow in the core. For rapid shutdown and for compensating for core burnup, the movable control rods are used. In a normal operating cycle several groups of control rods will be in the core at the beginning of core life, but will be completely out of the core at the end of the cycle. At any stage in core life some absorber rods are outside the core. These can be inserted to produce rapid shutdown.

Because of the steam-separator structure above the core, control rods in the BWR core must be inserted from the bottom. Insertion is thus not gravity assisted. Control rod drive is hydraulic. Compressed gas cylinders provide for emergency insertion if needed.

No neutron absorber is dissolved in the coolant, hence, control of absorber concentration is not required. However, cleanup of coolant containments, both solid and gaseous, is continuous. Maintaining a low oxygen concentration is especially important for the inhibition of stress-assisted corrosion cracking that has occurred in the primary system piping of a number of BWRs.

56.11 REACTOR SAFETY ANALYSIS

Under existing law the Nuclear Regulatory Commission has the responsibility for licensing power reactor construction and operation. (Those who operate the controls of the reactor and those who exercise immediate supervision of the operation must also be licensed by the Commission.) Commission policy provides for the granting of an operating license only after it has been formally determined by Commission review that the reactor power plant can be operated without undue risk to the health and safety of the public.

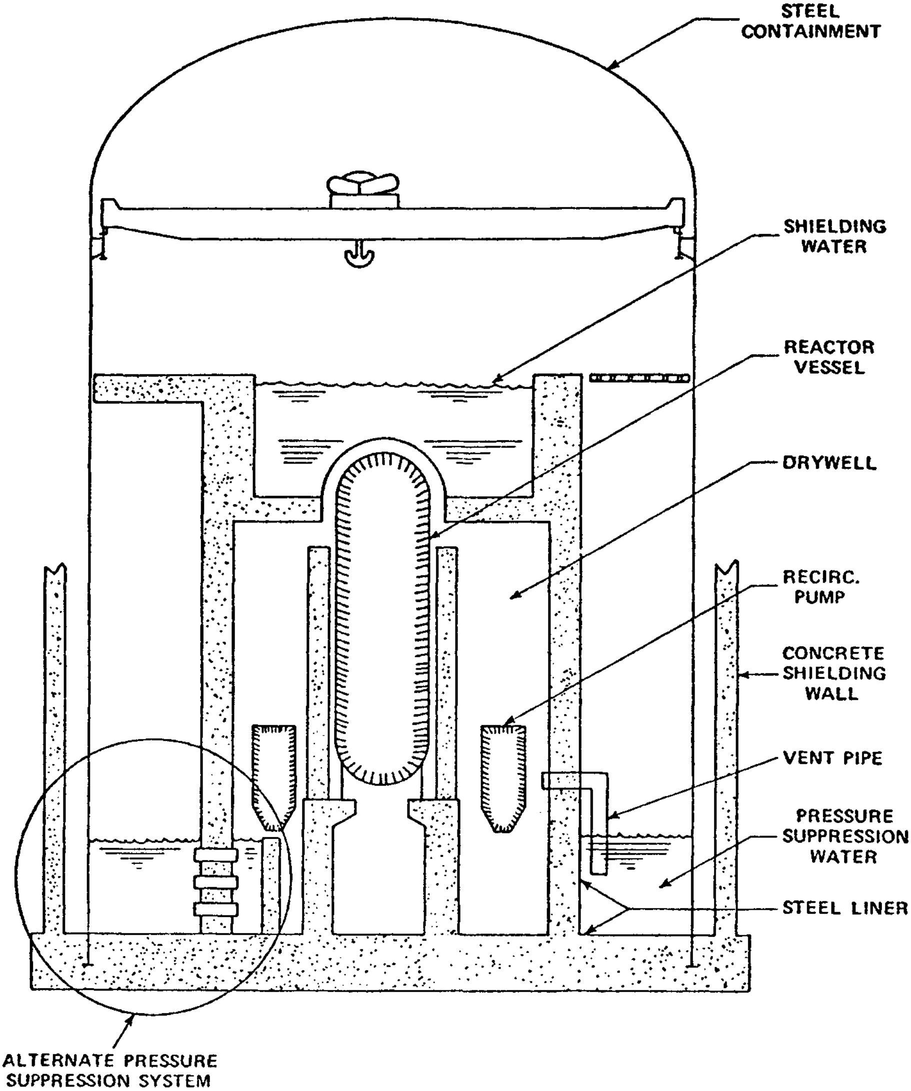


Fig. 56.10 Mark III containment for BWR. (Used by permission of Pergamon Press, Inc., New York.)

The current review process includes a detailed analysis of reactor system behavior under both normal and accident conditions. The existing approach involves the postulating of a set of design basis accidents (DBAs) and carrying out a deterministic analysis, which must demonstrate that the consequences of the hypothesized accidents are within a defined acceptable region. A number of the accident scenarios used for this purpose are of sufficiently low probability that they have not been observed in operating reactors. It is not practical to simulate the accidents using full-scale models or existing reactors. Analysis of reactor system behavior under the hypothesized situations must depend on analytical modeling. A number of large and complicated computer codes have been developed for this purpose.

Although the existing approach to licensing involves analysis of DBAs that can cause significant damage to the reactor power plant, none of the DBAs produces any calculable damage to personnel. Indeed core damage severe enough to involve melting of the core is not included in any of the sequences that are considered. However, in the design of the plant allowance is made, on a nonmechanistic basis, for consequences beyond those calculated for the DBA. This part of the design is not based on the results of an analytical description of a specific serious accident, but rather on nonmechanistic assumptions meant to encompass a bounding event.

This method of analysis, developed over a period of about two decades, has been used in the licensing and in the regulation of the reactors now in operation. It is likely to be a principal component of the licensing and regulatory processes for at least the next decade. However, the accident at Three Mile Island in 1979 convinced most of those responsible for reactor analysis, reactor operation, and reactor licensing that a spectrum of accidents broader than that under the umbrella of the DBA should be considered.

In the early 1970s, under the auspices of the Atomic Energy Commission, an alternative approach to dealing with the analysis of severe accidents was developed. The result of an application of the method to two operating reactor power plants was published in 1975 in an AEC report designated as WASH-1400 or the Reactor Safety Study. This method postulates accident sequences that may lead to undesirable consequences such as melting of the reactor core, breach of the reactor containment, or exposure of members of the public to significant radiation doses. Since a properly designed and operated reactor will not experience these sequences unless multiple failures of equipment, serious operator error, or unexpected natural calamities occur, an effort is made to predict the probability of the required multiplicity of failures, errors, and calamities, and to calculate the consequences should such a sequence be experienced. The risk associated with the probability and the consequences can then be calculated.

A principal difficulty associated with this method is that the only consequences that are of serious concern in connection with significant risk to public health and safety are the result of very-low-probability accident sequences. Thus data needed to establish probabilities are either sparse or nonexistent. Thus, application of the method must depend on some appropriate synthesis of related experience in other areas to predict the behavior of reactor systems. Because of the uncertainty introduced in this approach, the results must be interpreted with great care. The method, usually referred to as probabilitistic risk analysis (PRA), is still in a developmental stage, but shows signs of some improvement. It appears likely that for some time to come PRA will continue to provide useful information, but will be used, along with other forms of information, only as one part of the decision process used to judge the safety of power reactors.

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