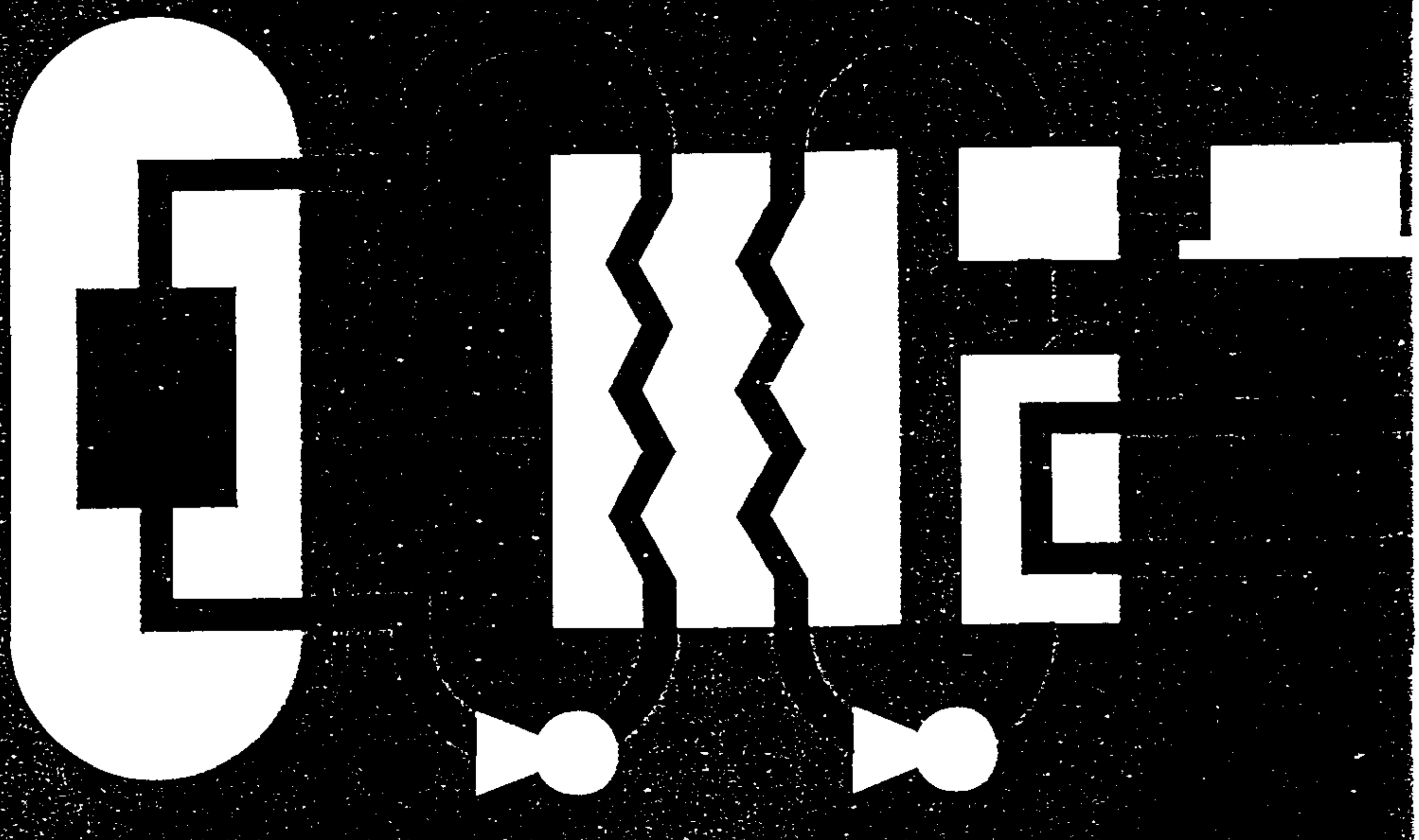
A Guidebook to Nuclear Reactors

ANTHONY V. NERO, JR.

Reactors
Fuel Cycle
The [illegible] of Nuclear Power



CHAPTER FIVE

Pressurized-Water Reactors

MOST OF THE light-water reactor power plants now operating or under construction use pressurized-water reactors. Westinghouse supplies somewhat more than half of the PWRs in the United States, with the remainder split between Babcock & Wilcox and Combustion Engineering. Many of the details of PWRs vary from one vendor to another and even, for the same manufacturer, from one reactor to the next. However, the fundamental characteristic of PWRs remains the same: that the primary coolant raises steam in a heat exchanger called a steam generator and this steam drives the turbine. A basic PWR system is shown schematically in Figure 5-1. Enclosed in a containment structure is the primary coolant system consisting of the reactor vessel and two or more primary coolant loops, each including piping, pumps, and a steam generator (perhaps shared). The safety injection (ECC) systems are also within the containment. Steam from the steam generators is transported out of the containment to the turbogenerator system. Condensate returns to the steam generators. Although three corporations offer PWRs, the system description that follows is based largely on that of Westinghouse. PWRs from the other manufacturers will vary in detail, particularly in the matter of the primary coolant loop arrangement.

BASIC PWR SYSTEM

The basic unit of a PWR core is a fuel pin typical of water-cooled reactors. For such reactors, the uranium dioxide fuel material is pressed into "pellets," cylinders about one-half inch in diameter and of similar height. These pellets are sintered (heated to high temperatures), ground to the proper dimensions, then sealed, along with a helium atmosphere, in a cladding material. This constitutes a fuel rod or pin. The cladding is typically an alloy of zirconium, chosen for its low neutron cross-section, as well as for its structural properties. The fuel pin for a light-water reactor is shown schematically in Figure 5-2. These pins, each more than

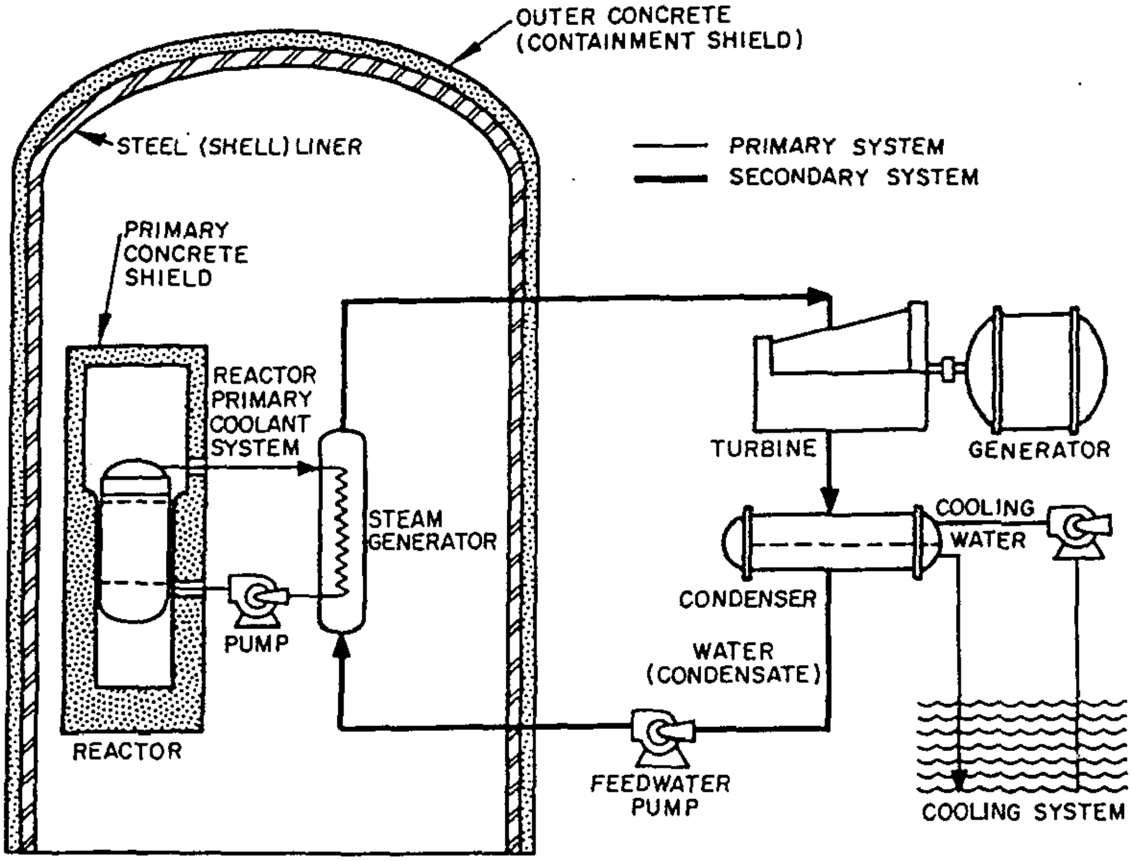
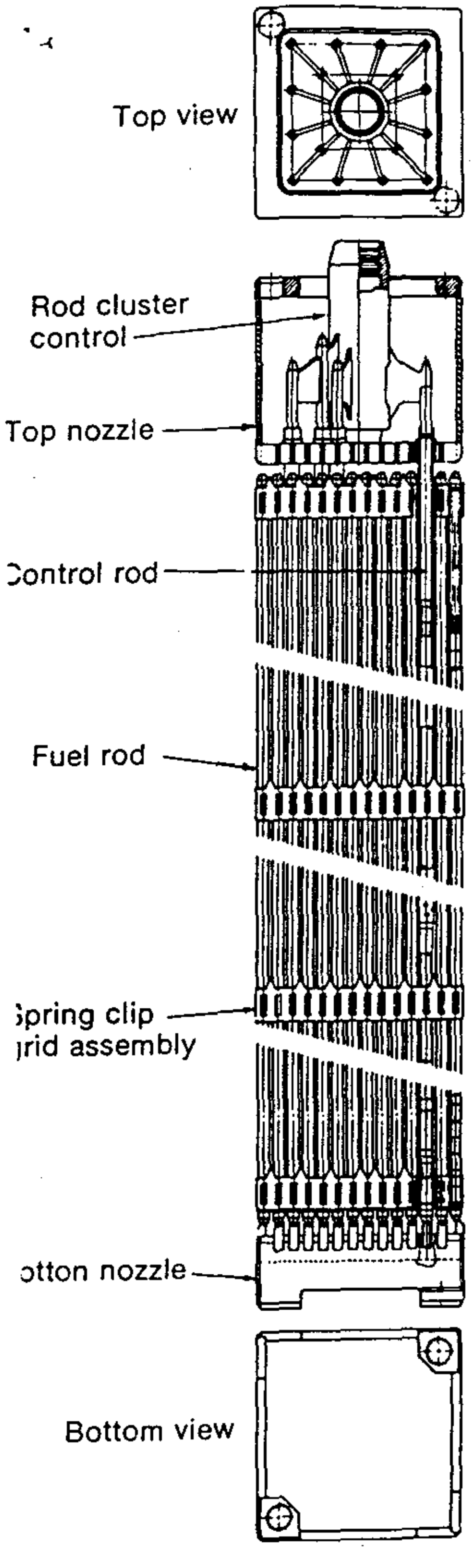


Figure 5-1. SCHEMATIC PRESSURIZED-WATER REACTOR POWER PLANT.
The primary reactor system is enclosed in a steel-lined concrete containment building. Steam generated within the building flows to the turbine-generator system (outside the building), after which it is condensed and returned to the steam generators. (Figure reproduced from ERDA-1541.)

12 feet (3.6 m) long for LWRs, are assembled into bundles or "assemblies," the operational unit for handling, refueling, etc. Should plutonium be recycled into light-water reactors, it would be handled in much the same way. In the United States, it has been proposed that the plutonium oxide be finely mixed with the uranium dioxide before a fuel pellet is formed.

The core of a pressurized water reactor consists of a large number of square fuel assemblies or bundles. Figure 5-3 shows one of these assemblies, in this case containing a control rod cluster. Many PWRs use assemblies that consist of 15 X 15 arrays of fuel pins of the type indicated in Figure 5-2, each somewhat more than 12 feet long. Newer PWRs use 17 X 17 assemblies. These pins or rods are closely held together in a matrix with no outer sheath, by the assembly's top and bottom structures, and by spring clip grid assemblies. A full-sized (about 1000 MWe) PWR may contain nearly 200 assemblies with about 40 or 50 thousand fuel pins, contain-



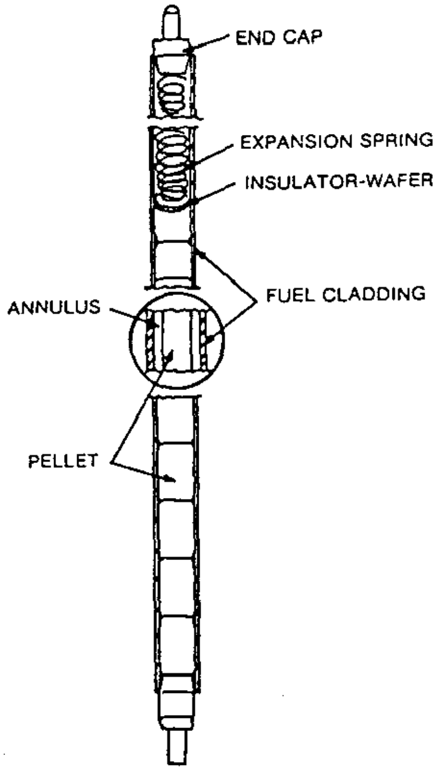


Figure 5-2. CUTAWAY VIEW OF OXIDE FUEL FOR COMMERCIAL LWR POWER PLANTS.
The basic unit in the core of a light-water reactor is a fuel rod containing uranium oxide pellets in a Zircaloy cladding. The rod is filled will helium gas and welded shut. The circled portion exaggerates the annular space between the pellet and the cladding. (Figure reproduced from WASH-1250.)

Figure 5-3. FUEL ASSEMBLY FOR A PRESSURIZED-WATER REACTOR.
In a pressurized-water reactor, fuel rods are assembled into a square array, held together by spring clip assemblies and by nozzles at the top and bottom. The structure is open, permitting flow of coolant both vertically and horizontally. All the assemblies in the reactor may have the same mechanical design, including provision for passage of a control rod cluster (shown in the figure). Where there is no cluster, these positions may have neutron sources, burnable poison rods, or plugs. (Figure reproduced from WASH-1250.)

ing about 110 tons (100 metric tons) of uranium dioxide (and plutonium, were recycle to occur).

All the assemblies have provision for the passage of control rods through rod guides which take about 20 of the positions that could otherwise hold fuel rods. If the assembly is used as a control assembly, and about 30% of them are, the rods from that assembly are manipulated from the top as a cluster. The control drives are at the top of the pressure vessel. In case the assembly does not contain a rod cluster, control rod positions may be taken by burnable poison, in this case boron 10 which is used after initial reactor operation to offset excess reactivity, or by neutron sources, used for reactor startup. Otherwise, these positions are left vacant and water flow through them is blocked.

Most of the control rods have silver-indium-cadmium neutron absorber for the full length of the core and are used for operational control of the reactor, including load following, and for quick shutdown capability. Reactor "trip" capability is provided by the fact that the rods can simply be dropped into place gravitationally; somewhat fewer than half the control assemblies are reserved for this shutdown capability, the remaining being used for operational control. Some of the control rods have absorber only in their bottom quarter and are used for shaping the axial (vertical) power distribution. The other basic means of control is to introduce boric acid into the primary coolant. This method is used both for shutdown and for adjusting the reactivity to take account of long-term changes, such as reduction in fissile content and buildup of fission product poisons. Effectively, boron adjustment is used to keep the reactivity within the range of the control rods.

The core has three enrichment zones, with the most highly enriched (slightly greater than 3%) at the periphery and the other enrichments scattered through the interior, all to provide a relatively flat power distribution. The average power generation density in the core is about 98 kW/liter. (See Table 5-1 for other PWR parameters.) This energy is carried away by a very large flow of water, about 140 million pounds per hour (18 Mg/s). The water's operating temperature is about 600 °F (315 °C), which maintains the clad temperature nominally below 700 °F (371 °C).

The core, control rods, and core-monitoring instrumentation are contained in a large pressure vessel, designed to withstand pressures, at operating temperatures, of about 2500 psi (17 MPa). The vessel may be about 40 feet in height (12 m) and 14 feet (4 m) in diameter, with carbon steel walls 8 inches (20 cm) or more thick. All inner surfaces that come into contact with the coolant are clad in stainless steel. (This is also true of all other parts of the primary coolant system, except for those portions that are made of Zircaloy or Inconel, i.e., the fuel cladding and the steam generator tubing, respectively.) The top head of the vessel, which holds all the control rod drives, is removable for refueling. The reactor vessel and its contents are shown in Figure 5-4.

The coolant enters the reactor vessel through nozzles near the top of the core and, constrained by a "core barrel" between the vessel and the core, flows to the bottom of the core. The water then flows up through the core and out exit nozzles

TABLE 5-1.
Representative Characteristics of Pressurized-Water Reactors

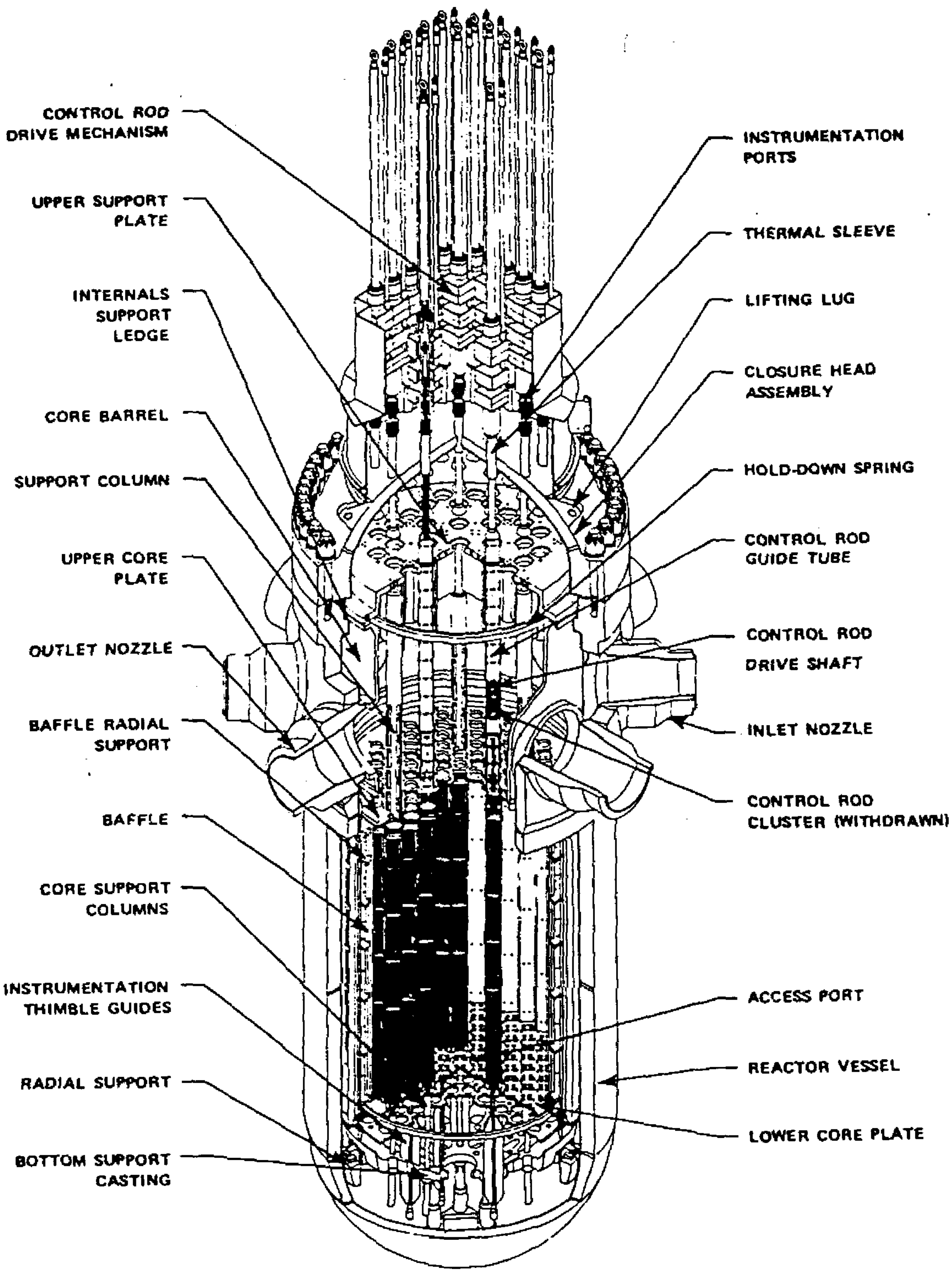
Core thermal power	3,411 MWth
Plant efficiency	32%
Plant electrical output	1,100 MWe
Core diameter	134 in (3.4 m)
Core (or fuel rod) active length	144 in (3.7 m)
Core weight (mass)	276,000 lb (125 Mg)
Core power density	98 kW/liter
Cladding material	Zircaloy-4
Cladding diameter (OD)	0.422 in (1.07 cm)
Cladding thickness	0.024 in (0.06 cm)
Fuel material	UO_2
Pellet diameter	0.37 in (0.9 cm)
Pellet height	0.6 in (1.5 cm)
Assembly array	15 X 15, open structure[a]
Number of assemblies	193
Total number of fuel rods	39,372[a]
Control rod type	B_4C or Ag-In-Cd in cylindrical rod
Number of control rod assemblies	60 (may vary considerably)
Number of control rods per control assembly	20 (may vary considerably)
Total amount of fuel (UO_2)	217,000 lb (98 Mg)
Fuel power density	38 MW/Te
Fuel/coolant ratio	1/4.1
Coolant	Water (liquid phase)
Total coolant flow rate	136 X 10^6 lb/hr (17 Mg/sec)
Core coolant velocity	15.5 ft/sec (4.7 m/sec)
Coolant pressure	2,250 psi (15.5 MPa)
Coolant temperature (inlet at full power)	552 °F (289 °C)
Coolant temperature (outlet at full power)	617 °F (325 °C)
Nominal clad temperature	657 °F (347 °C)
Nominal fuel central temperature	4,140 °F (2,282 °C)
Radial peaking factor (variation in power density)	1.5
Axial peaking factor	1.7
Design fuel burnup	32,000 MWd/Te (heavy metal); varies
Fresh fuel assay	3.2% ^{235}U (less in initial load)
Spent fuel assay (design)	0.9% ^{235}U, 0.6% $^{239,241}Pu$
Refueling sequence	One-third of the fuel per year
Refueling time	17 day (minimum)

a. PWRs now being licensed have a 17 X 17 assembly array, with thinner rods totaling 50,952. Other specifications may be slightly changed.

Source: Taken primarily from Westinghouse Electric Corp. specifications.

to the steam generators. From there, the coolant is recirculated to the core by large primary coolant pumps. The main elements of the primary coolant system are shown in Figure 5-5.

The pressure in the primary system is maintained at about 2250 psi (15.5 MPa), preventing the formation of steam. Instead, steam is raised in a secondary system by allowing heat to flow from the high-pressure primary coolant to the lower pressure secondary fluid. This heat transfer occurs through the walls of large numbers of tubes through which the primary coolant circulates in the steam generators. After the steam has passed through separators to remove water droplets,


CONTROL ROD DRIVE MECHANISM
UPPER SUPPORT PLATE
INTERNALS SUPPORT LEDGE
CORE BARREL
SUPPORT COLUMN
UPPER CORE PLATE
OUTLET NOZZLE
BAFFLE RADIAL SUPPORT
BAFFLE
CORE SUPPORT COLUMNS
INSTRUMENTATION THIMBLE GUIDES
RADIAL SUPPORT
BOTTOM SUPPORT CASTING
INSTRUMENTATION PORTS
THERMAL SLEEVE
LIFTING LUG
CLOSURE HEAD ASSEMBLY
HOLD-DOWN SPRING
CONTROL ROD GUIDE TUBE
CONTROL ROD DRIVE SHAFT
INLET NOZZLE
CONTROL ROD CLUSTER (WITHDRAWN)
ACCESS PORT
REACTOR VESSEL
LOWER CORE PLATE

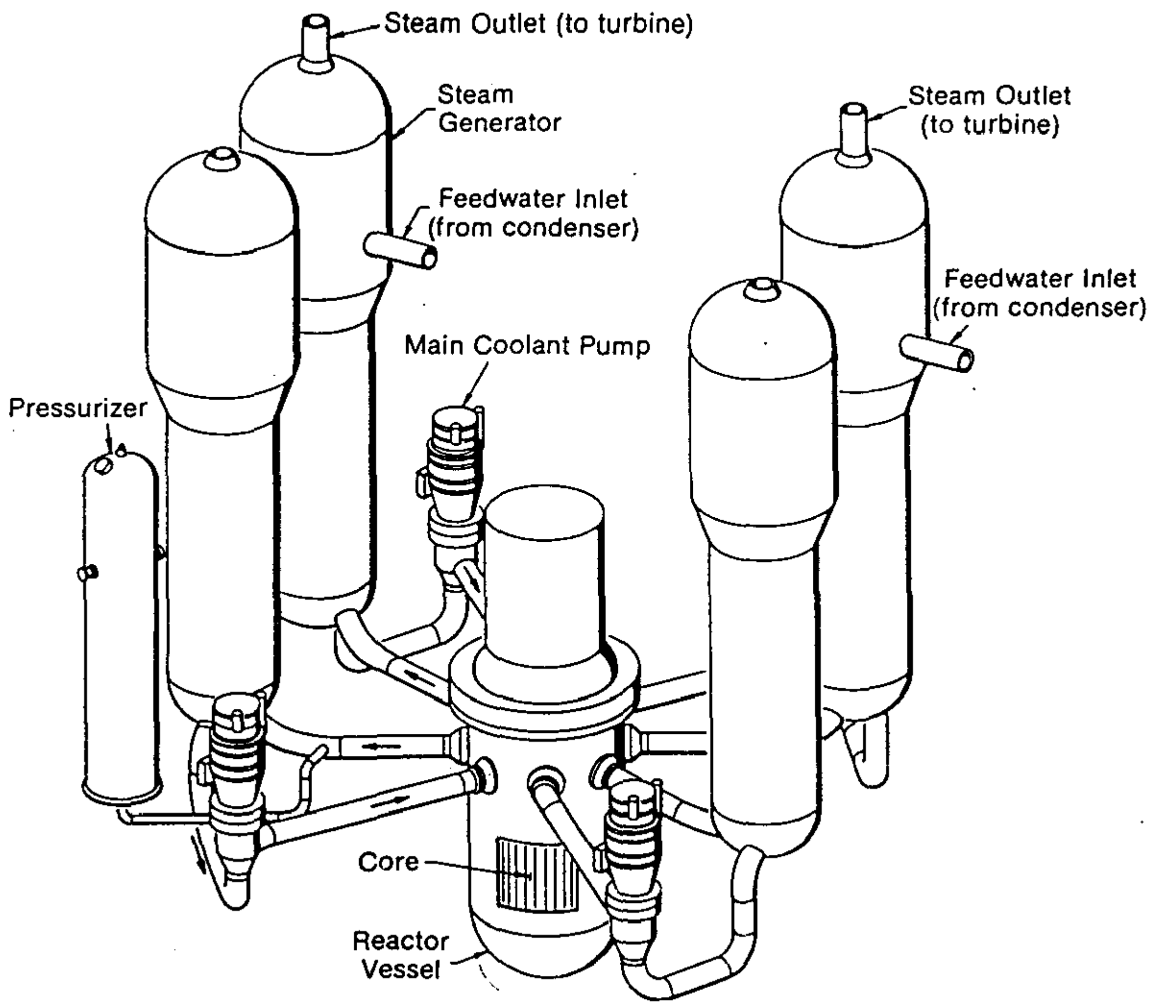


Figure 5-5. ARRANGEMENT OF THE PRIMARY SYSTEM FOR A WESTINGHOUSE PWR. The primary system constitutes the nuclear steam supply system for a PWR plant. In the four-loop arrangement shown in the figure, each loop has its own steam generator and coolant pump. A pressurizer is connected to one of the loops. The primary coolant enters and leaves the steam generator from the bottom; one of the U-tubes in the generator is shown in Figure 5-8. (Figure reproduced from WASH-1250.)

Figure 5-4. PRESSURIZED-WATER REACTOR VESSEL AND INTERNALS. (At Left) The core of a pressurized-water reactor is contained in a large steel vessel through which coolant flows. After passing into an inlet nozzle, the water flows down between the core barrel and the vessel wall, until it reaches the plenum beneath the core; there it turns upward to flow through the core and out one of the outlet nozzles to the steam generators. The top of the reactor vessel, which is removable for refueling, supports mechanisms for driving control rods. (Figure courtesy of Westinghouse Electric Corp.)

thereby reducing its moisture content to less than 1%, it proceeds to the turbogenerator for the production of electricity. After condensation, it returns as liquid to the steam generators. The overall thermal efficiency of a PWR is about 32%. In the steam generators, the primary coolant passes only once through a single tube (i.e., the steam generators are "once through"), which is ordinarily either U-shaped or straight. A large PWR may have four external circuits, indicated schematically in Figure 5-5, each with its own steam generator and pump. As seen in Figure 5-6, this arrangement may vary from one manufacturer to another.

Since maintenance of the pressure near the design value is crucial (to avoid the formation of steam in the primary coolant, on the one hand, and rupture of the primary circuit, on the other), a PWR system also includes a "pressurizer," as shown in Figures 5-5 and 5-6, connected to the "hot" leg of one of the steam generator circuits. The pressurizer volume is occupied partly by water and partly by steam; it has heaters for boiling water and sprayers for condensing steam, as needed, to keep the pressure within specified operating limits.

AUXILIARY SYSTEMS

It is useful to mention the systems that support the main reactor systems and that, in addition, are sometimes intimately connected with the safety systems discussed in the next section. These include particularly the systems for controlling the chemistry and volume of the primary reactor coolant and the decay heat removal system.

The chemistry and volume (C & V) control system provides water for the primary coolant system and reduces the concentration of corrosion and fission products in the coolant, as well as adjusting the boric acid concentration. When the reactor is operating, the system functions by continuously bleeding water from the primary coolant system, passing it through demineralizers and into a volume control tank. Liquid supplied to the primary coolant system is some combination of fluids from this tank, from a fresh demineralized water supply, from the boric acid tanks, and from chemicals needed to maintain coolant chemistry within specifications. The C & V system operates in conjunction with the pressurizer to maintain the proper coolant pressure and volume under normal operation. The system may also maintain the proper concentrations of dissolved gases, particularly hydrogen in the coolant. In connection with this function, the C & V system is a source of gas that must be handled by the gaseous waste processing system; the gaseous waste system provides for storage of gas and, in some cases, ultimate return, if necessary, to the reactor system. A liquid waste processing system whose primary purpose is to process liquids from various drain systems may also be connected with the C & V system; when the liquid may contain tritium, such as the primary coolant does, it may be demineralized and returned to the C & V system. The configuration of the C & V system, or its equivalent, and its connection with the primary coolant system and the waste processing systems can vary significantly from one reactor to another Figures 3-3 and 3-4 provide one example of liquid and gaseous waste control systems.

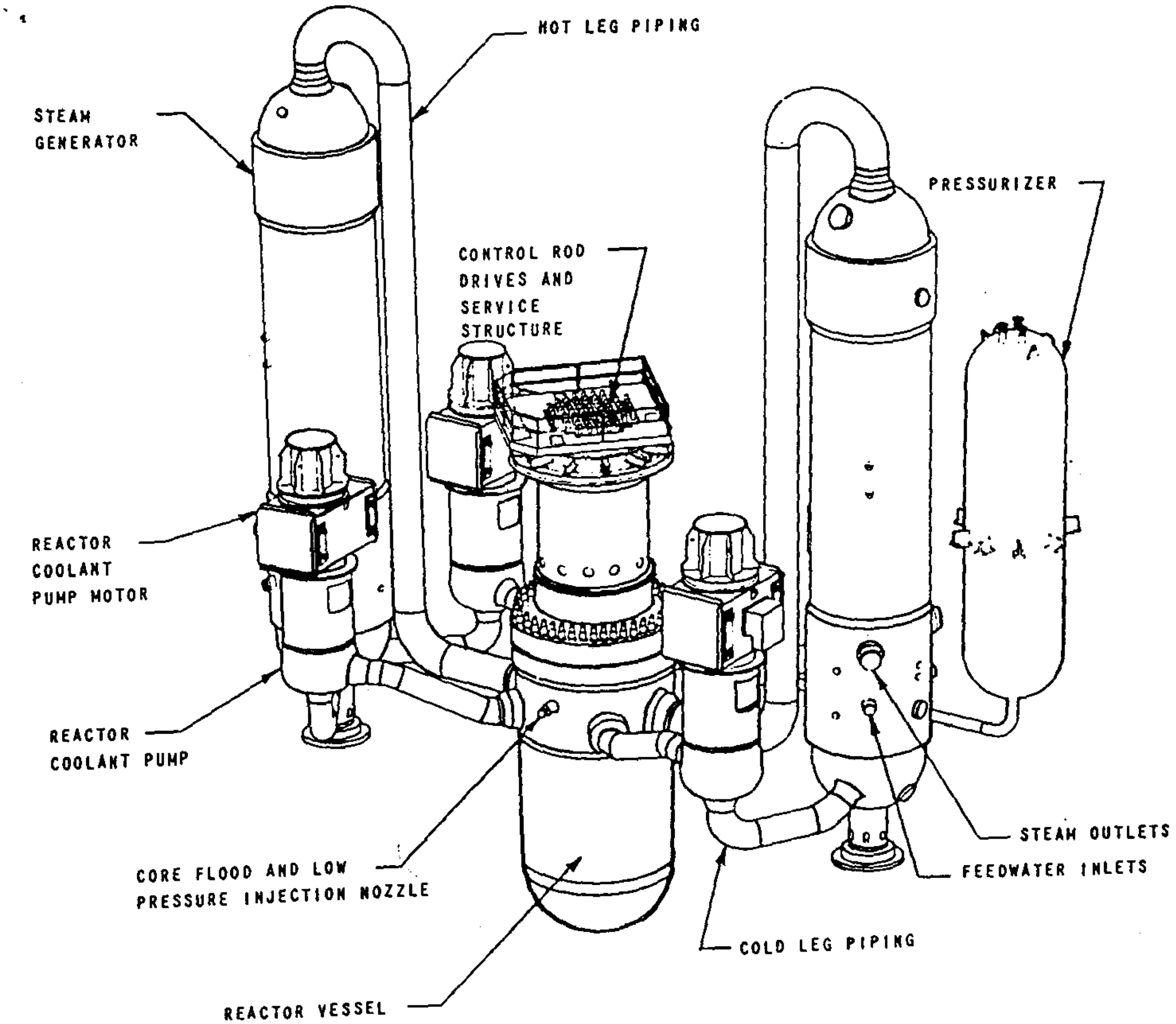


Figure 5-6. ALTERNATIVE ARRANGEMENT FOR A PWR PRIMARY SYSTEM. This PWR system has two outlet nozzles, each leading to a steam generator. The outlet of each generator is connected with two coolant pumps, each of which is connected with an inlet nozzle at the reactor vessel. These steam generators use vertical tubes, rather than the U-tube design of figure 5-5. (Figure courtesy of Babcock & Wilcox Co.)

The residual heat removal (RHR) system removes decay heat from the primary coolant system during plant shutdown. The system consists primarily of heat exchangers and pumps. At the initial stages of shutdown, heat is still removed by the steam generators, and the resulting steam is discharged directly to the condenser, bypassing the turbine. When the reactor coolant has dropped in temperature and, even more significantly, in pressure, the RHR system is turned on. The cooling function of the steam generators is then removed; one of the reactor coolant pumps continues to operate for a time to ensure uniform residual cooldown. The heat removal system may also be used in conjunction with the emergency injection systems discussed in the next section.

In addition to these specific auxiliary systems, a PWR has numerous other auxiliary systems which provide basic services for the major systems. These include systems for cooling specific components, for providing power (even in emergency situations), and for controlling, via complex electrical networks with either manual or automatic supervision, the functions of the basic systems. Although we do not devote attention here to these numerous systems, they must be adequate to constitute a basis for economic and safe operation of nuclear power plants.

SAFETY SYSTEMS

A number of important safety features are added to the basic reactor system in order to minimize the danger from reactor accidents. The immediate safety function following any abnormality is to shut down rapidly (i.e., to "trip" or "scram") the chain reaction. This is accomplished by the shutdown control rods described earlier. In the event that the abnormality continues to the extent of rupturing the primary system or otherwise reducing coolant inventory, emergency injection systems are available to provide continued cooling of the core. Finally, in the event that fuel melting occurs, the containment building and its subsystems are available to minimize the amount of radioactivity that escapes into the general environment.

Before proceeding to a discussion of the emergency core cooling systems, it is worth noting that the components of both the primary coolant system and the various ECC systems are enclosed by the containment building. Such a building is shown schematically in Figure 5-7. This structure is steel-lined reinforced concrete, designed to withstand the overpressure expected if all the primary coolant were released in an accident. Sprays and cooling systems (such as the relatively new ice condenser system of Figure 5-7) are available for washing released radioactivity out of the containment atmosphere and for cooling the internal atmosphere, thereby keeping the pressure below the containment design pressure. At the initial phases of a severe accident, the containment interior is isolated from the outside world. The basic purpose of the containment system, including its spray and cooling functions, is to minimize the amount of released radioactivity that escapes to the external environment. The basic design criterion is the dose limitation specified by 10 CFR 100 (see Chapter 4).

Meeting these criteria depends, however, on successful operation during emergencies of various systems associated with the reactor. Of primary interest is

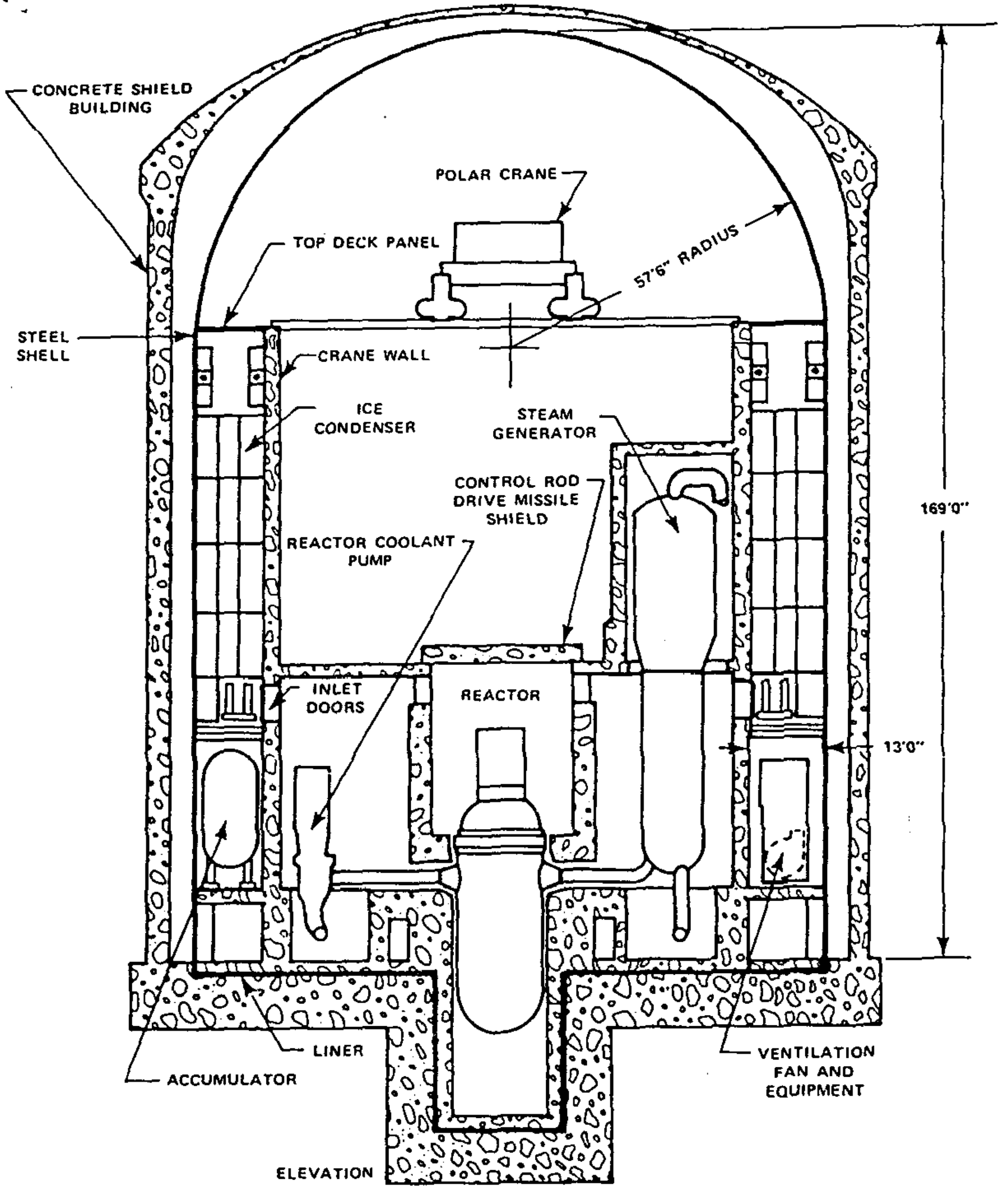
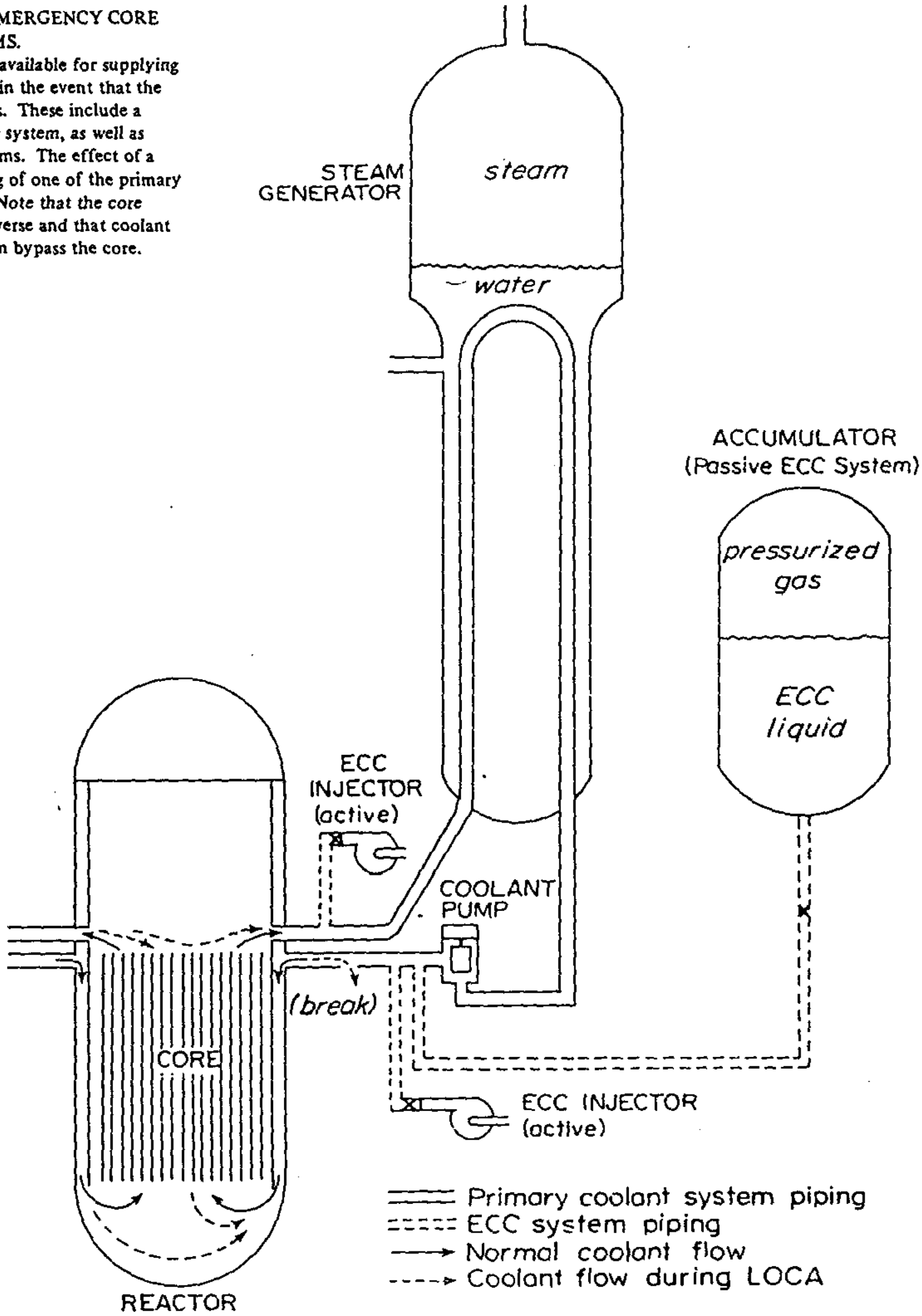


Figure 5-7. CROSS-SECTION OF A PWR CONTAINMENT BUILDING.
The containment building has the entire primary system, as well as various safety systems, in its interior. The building itself is concrete, with a steel shell inside. The safety systems within the building include emergency core cooling systems (note the accumulator), pressure control systems (one form of which may be the ice condenser indicated), and ventilation equipment. (Figure courtesy of Westinghouse Electric Corp.)

Figure 5-8. PWR EMERGENCY CORE COOLING SYSTEMS.
Several systems are available for supplying coolant to the core in the event that the primary system fails. These include a passive accumulator system, as well as active injector systems. The effect of a break in the cold leg of one of the primary loops is indicated. Note that the core coolant flow can reverse and that coolant from other loops can bypass the core.



the behavior of the systems to be called upon during a loss-of-coolant accident. Such an event can vary greatly in degree, and the several ECC systems are intended to cope with a broad range of accidents, ranging from minor leakage from a small pipebreak to a rapid loss of coolant (blowdown) arising from a complete shear of a main coolant return line ("cold leg") in one of the coolant loops. The only LOCA that these systems are not designed to cope with is a catastrophic rupture of the reactor vessel, in which case there is no system that holds water.

Not surprisingly, there should be little difficulty in dealing with a small to intermediate break. It is, rather, the large breaks whose consequences are most difficult to control. Figure 5-8 indicates schematically the major features of interest in a major cold leg break. Whereas coolant normally flows down the annulus between the core barrel and the reactor vessel, then up through the core, and out to the steam generator, the fluid in the core can reverse direction to flow up the barrel and out the broken leg. Indeed, coolant from other loops can bypass the core to escape out the break. The coolant inventory can be exhausted very rapidly, and any ameliorating action must be massive and rapid. Accordingly, the first system to respond is a passive system, consisting of accumulators which are isolated from the primary system by check-valves that open as soon as the primary system pressure drops much below 1000 psi (7 MPa). Each accumulator has about 1000 cubic feet (28 m^3) of liquid, and each reactor system has two or more units. The accumulators act with no delay, and inject fluid either into the cold legs (as shown in Figure 5-1) or into the reactor vessel. Of course, for the case of a cold leg break with cold leg injection, one of the units would be ineffective. It is also conceivable that other units could be ineffective, as would be the case if fluid injected bypassed the core to escape through the cold leg break.

In any case, this accumulator is rapidly exhausted. Long-term cooling would be provided by two active low-pressure injection systems (LPIS), which pump fluid, each at about 3000 gallons per minute (190 liter/s) into either hot or cold legs, or both. These systems require about 20 seconds to become operative; and it is assumed, in accident analysis, that one of the two systems would be effective.

Finally, for small breaks that do not greatly reduce the pressure, two high-pressure injection systems (HPIS) provide makeup water at relatively low rates (about 400 gallons per minute or 25 liter/s). This water is usually injected into a hot or cold leg. However, the HPIS and LPIS water is injected into the reactor vessel in some designs. The source of water for the active injection systems is typically the volume control tanks and the refueling water storage tanks.

It is the emergency core cooling systems whose operation is uncertain and generates much heated controversy. As noted in Chapter 4, transient conditions during a large LOCA are so difficult to model that, for licensing purposes, a "conservative" model and associated criteria are specified. The differences between such a model and one that is "realistic," but very uncertain, is illustrated in Figure 5-9, showing fuel clad temperatures during a large LOCA. The "conservative" model yields high temperatures, presumed by the Nuclear Regulatory Commission to be an upper limit, for use as a criterion for protecting the public.

The existence alone of emergency systems is not sufficient to limit the course of an accident, even assuming the systems are designed adequately. In addition, the overall reactor system must be arranged to ensure that necessary safety systems operate when required. For this reason, individual systems are duplicated as noted above, and their control and power supplies (for active systems) are independent of each other and of the main reactor systems. Unintended dependencies between systems can reduce the overall dependability of emergency response and can, of course, introduce imponderables into an assessment of the risk from nuclear plant accidents. This question of redundancy and independence also arises in connection with those portions of the main reactor systems that would be used during an accident. For example, of the four primary coolant pumps in a large PWR, each is generally large enough to provide alone for sufficient coolant flow for removal of decay heat after shutdown.

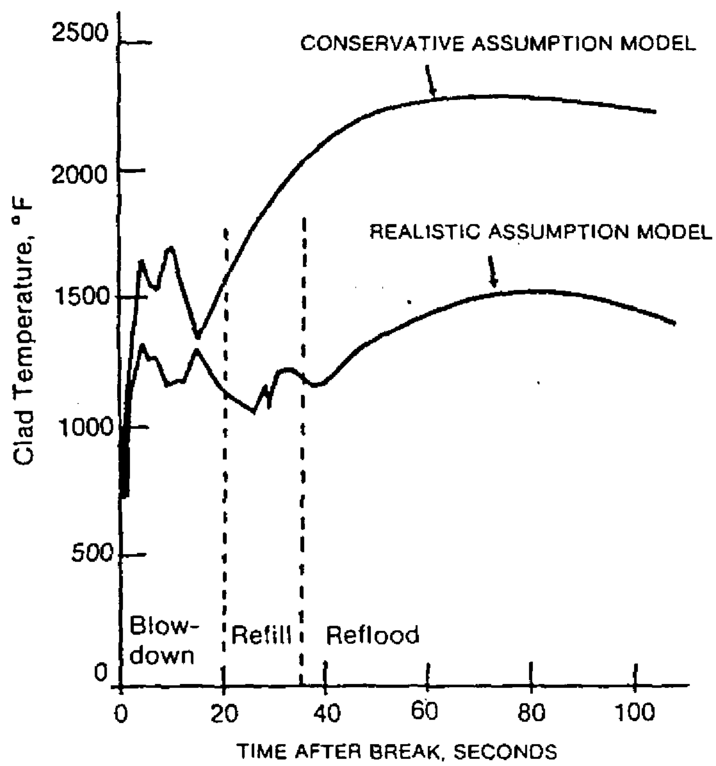


Figure 5-9. SCHEMATIC CALCULATED FUEL CLAD TEMPERATURES FOR A PWR LOCA.

System conditions during an accident may be calculated using both "conservative" and "realistic" models, the first to put an effective limit on the severity of the accident, the second to yield the best available prediction for what will happen. The figure indicates how the models can differ in the results calculated for clad temperatures during the course of a PWR accident. (Figure reproduced from WASH-1250.)

TABLE 5-2.
Approximate Pressurized-Water Reactor Neutronics
(start of life)

Approximately 2.0 fast neutrons are produced following the absorption of 1 neutron by ^{235}U and have the following fate:

0.6[a]	Captured by ^{238}U (largely in the resonance region, leading to ^{239}Pu production)
1	Absorbed by ^{235}U (of which 0.8 result in fissions)
0.1	Absorbed by water
0.1	Absorbed by structural material and fission product poisons
0.2	Absorbed by control poisons
2.0	

a. The conversion ratio is thus 0.6.

NEUTRONICS, FUEL UTILIZATION, AND REACTOR OPERATION

It is typical of light-water reactors, as they operate at present, that the conversion ratio, the ratio of fissile material produced to that destroyed, is about 0.6. Roughly speaking, for each slow neutron absorbed by ^{235}U, about 2.0 fast neutrons are produced.[1] These are rapidly slowed to thermal energies by the water moderator, but in the process a substantial number are captured by ^{238}U resonances. Of the neutrons that reach thermal energies, some are still captured by ^{238}U, but most are captured by ^{235}U, water, structural materials, fission product poisons, and control poisons. Table 5-2 indicates these results for a PWR just after initial fueling. Note that the ratio of ^{238}U captures (yielding ^{239}Pu) to ^{235}U absorptions (de-stroying ^{235}U) is about 0.6. As the reactor runs, fission product poisons build up, :he amount of fissile material decreases slightly, and the amount of control de-:reases, so that the tabulated neutron absorptions change slightly. However, the :onversion ratio does not change drastically, even though the types of fissile and ertile material will change. (For example, initially the only fissile material is ^{235}U, ›ut reactor operation builds up an inventory of ^{239}Pu and other isotopes.) Note lso that the reactor has a large amount of control at startup. Were it possible to ›duce this, the conversion ratio would rise. (See, for example, discussion of ANDU, Chapter 7, and of the light-water breeder reactor, Chapter 14.)

In an important sense, the difference between the conversion ratio and 1 is an :dicator of resource use. This difference, 1-0.6, is approximately 0.4 for LWRs, dicating a substantial deficit. However, the extent to which fissile resources are ;ed involves other factors, such as whether the fuel reaches its design "burnup" ıd whether material in the spent fuel is reprocessed. As was noted in Chapter 2, a VR would require that about 4100 tons of U_3O_8 be supplied to the fuel cycle for . use. Most of this supply is directly associated with the deficit caused by the low

[1] About 0.1 of these result from net neutron production from fission of ^{235}U by fast ›trons.

conversion ratio. Only a small percentage is needed to produce the initial fuel load. If uranium and plutonium are not recycled, this uranium requirement rises by about 50%.

However, all these requirements are based on the assumption that, on the average, the nominal amount of energy is extracted from the fuel rods. The design average burnup[2] of about 32,000 megawatt-days (thermal) per metric ton (MWd/Te) corresponds to a plant capacity factor of about 80%, providing the plant is refueled as scheduled. (Capacity factor is the ratio of actual electrical energy produced to the output if the plant operates continuously at 100% of rated power.) This is substantially higher than the 60% or so that has recently been achieved. If, in spite of relatively low average output for a plant, refueling proceeds on schedule, not as much energy will have been extracted from the fuel. This can represent a net loss of resources if the fuel is not reprocessed and fissile material recycled. Most PWRs have been constructed on the presumption that refueling would occur once yearly, in a low demand period, but possible losses of energy value may cause reexamination of such strict scheduling. The initial design has typically required a burnup of about 10,000 MWd/Te between refuelings, but lower burnup, for whatever reason, may warrant postponement of refueling.

Various factors may cause such low burnup. Most notable from the public health point of view are shutdowns due to difficulties with safety related equipment. Often, though, shutdowns occur because of other maintenance needs. The refueling shutdown (see below) takes a substantial amount of time. On the other hand, low capacity factor (and burnup) may arise from operating the plant at lower than nominal output. Reduced output may occur as a result of safety-related deratings, or as a result of reduced electrical demand. Normally, a nuclear power plant is designed as a base-load[3] unit, so that it ordinarily runs at full output, but as the portion of nuclear units in a utility grid grows, these units may more often be required to follow demand. PWRs can alter load easily enough, using control rods, to accommodate themselves to such needs. However, use in such a mode will reduce the capacity factor.

When refueling occurs, the reactor is unavailable for a substantial period, a minimum of two weeks. During this period, plant workers commonly receive a substantial portion of their annual radiation dose. Standard practice in controlling this dose is to flood the region around the reactor vessel in water, so that fuel is handled underwater. Fuel is moved by a conveyer between an opening in the side of the containment and the point where it is lifted over the edge of the open pressure vessel. In a PWR, the entire head (see Figure 5-4) is removed, along with the control rod drives. A portion of the inner core is removed, assemblies from the periphery are moved into this region, and fresh fuel is added at the periphery. A subsequent period for reconnecting and testing contributes substantially to the shutdown time of about two weeks.

[2] Because burnup will vary from one fuel rod to another, the fuel rods are designed to withstand higher burnup to make the average figure of 32,000 MWd/Te possible.

[3] Base-load plants are operated continuously in order to supply the minimum demand on a utility's grid. However, the utility will experience both daily and seasonal increases above this demand; these increases are met by peak and intermediate load units, usually fossil-fuel fired plants.

Bibliography — Chapter Five

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Includes a brief description of PWR systems, particularly those related to LOCAs.

CHAPTER SIX

Boiling-Water Reactors

ABOUT A THIRD of light-water reactors operating or under construction in the United States are boiling-water reactors. The distinguishing characteristic of a BWR is that the reactor vessel itself serves as the boiler of the nuclear steam supply system. In fact, the reactor vessel and associated equipment is the NSSS, as suggested in Figure 6-1. This vessel is by far the major component in the reactor building, and the steam it produces passes directly to the turbogenerator. The reactor building also contains emergency core cooling equipment, a major part of which is the pressure suppression pool which is – as suggested in Figure 6-2 – an integral part of the containment structure. As noted later in the chapter, earlier BWRs utilized a somewhat different containment and pressure suppression system. All the commercial BWRs sold in the United States have been designed and built by General Electric.

Several types of reactors that use boiling water in pressure tubes have been considered, designed, or built. In a sense, they are similar to the CANDU, described in Chapter 7, which uses pressure tubes and separates the coolant and moderator. The CANDU itself can be designed to use boiling light water as its coolant. The British steam-generating heavy-water reactor (Chapter 7) has such a system. Finally, the principal reactor type now being constructed in the Soviet Union uses a boiling-water pressure tube design, but with carbon moderator.

BASIC BWR SYSTEM

A boiling-water reactor core consists of a large number of fuel assemblies, each a square array as indicated in Figure 6-3. Although many BWRs use a 7 X 7 array, the most recent model (BWR/6) uses an 8 X 8 array, with thinner fuel rods; the cross-sectional size of the newer fuel bundle is therefore similar to the 7 X 7 array. The fuel pin is very similar to that discussed in Chapter 5 (Figure 5-2), with an active length of at least 12 feet (3.6 m). Unlike the typical PWR fuel bundle,

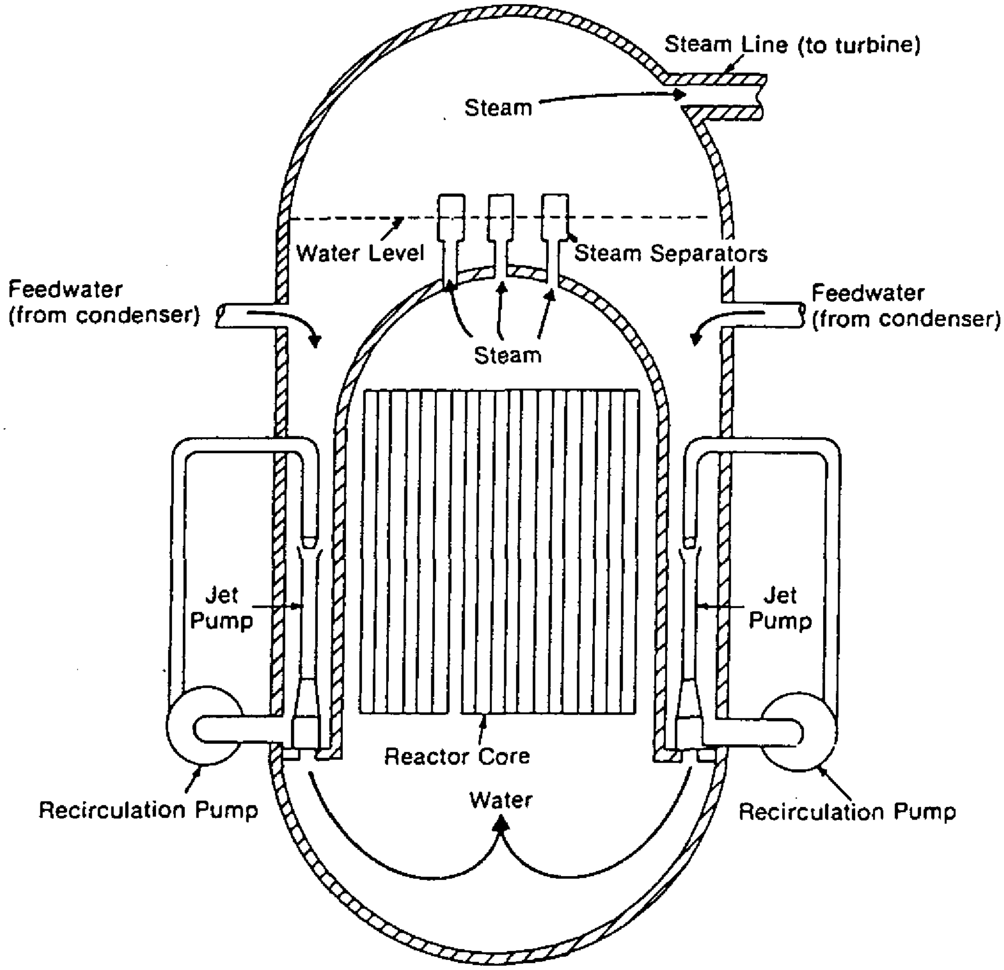


Figure 6-1. SCHEMATIC ARRANGEMENT OF A BOILING-WATER REACTOR.
In a boiling-water reactor, the steam for driving the turbogenerator is formed in the reactor vessel itself. Water passes through the core, forming steam which proceeds to the turbine. Water that is still liquid is recirculated in the vessel through the action of "jet pumps" which surround the core (see text). (Figure reproduced from WASH-1250.)

that of the BWR has an outer sheath (fuel channel) which constrains the flow of water in the assembly. An orifice at the bottom of the bundle then strongly determines the flow rate for a given assembly. The structural stability of the assembly is supplied by upper and lower tie plates, together with tie rods which take eight of the 64 array positions in an 8 X 8 assembly. (See the 4-assembly cross-section of Figure 6-3.) In addition, the assembly has several fuel rod spacers. Assemblies may also contain water rods (rods with water rather than UO_2), providing moderator within the bundle. A large BWR contains 764 assemblies, with 40 or 50 thousand fuel rods, and about 180 tons (160 metric tons) of UO_2.

The cross-shaped object around which the four bundles are arranged in Figure

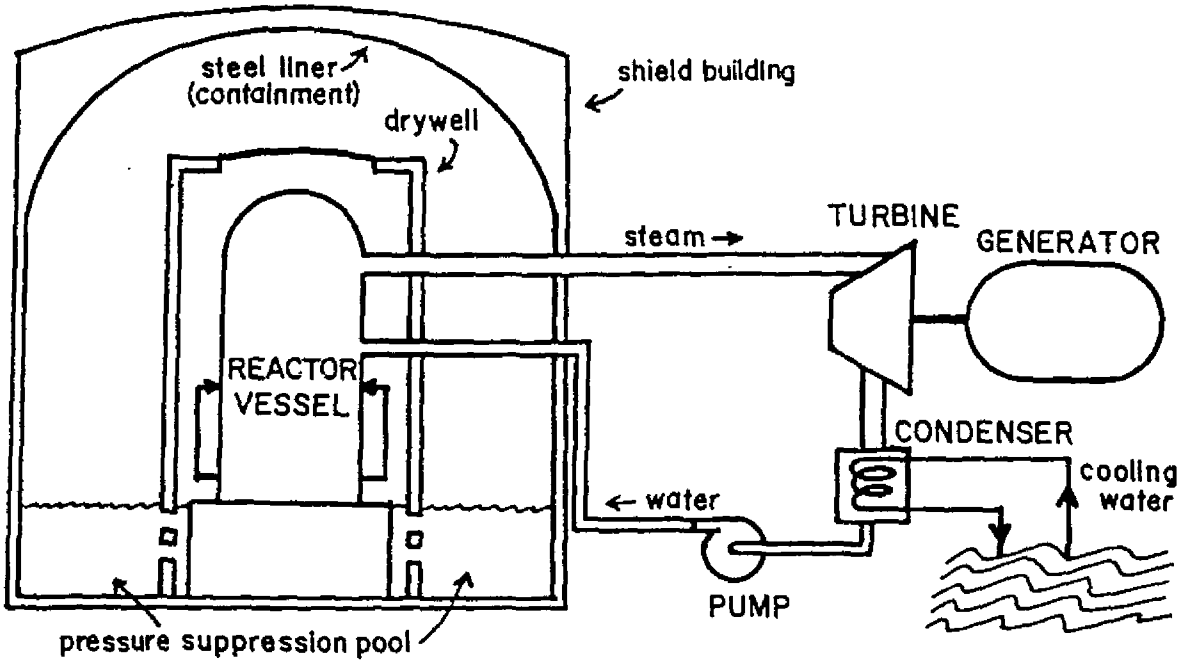


Figure 6-2. SCHEMATIC OF BOILING-WATER REACTOR POWER PLANT.
Steam from a BWR reactor vessel flows to the turbogenerator, after which it is condensed and returned as feedwater to the reactor vessel. The reactor vessel is contained in a dry well which, in turn, is within a reactor building.

6-3 is the cruciform control element used in BWRs. This element actually contains numerous boron-carbide-filled rods, one quarter in each of the blades shown. The cruciform rods are driven from the bottom of the reactor. These rods serve for both reactivity control and power flattening. The reactivity control includes long-term regulation and prompt shutdown ("scram"). Power flattening is needed in particular because, as the coolant rises through the core, it boils, resulting in lower coolant densities, and therefore poorer neutron moderation and lower power densities in the upper portion of the core. Burnable poisons are present as an oxide of gadolinium ("gadolinia") mixed into several of the fuel rods per bundle; this poison is present in all fresh fuel and is completely depleted during one year of operation. The reactor is also controlled by the recirculation rate (see below).

At refueling, assemblies are removed from the central core region and replaced by assemblies from the periphery. Fresh fuel is then added to the periphery of the core. Fresh fuel has an average enrichment of 2.4 to 3.0%. Within an assembly, the enrichment will vary, with lower enrichment fuel in the corners and near the water gaps; it is in these regions that neutrons are more effective because they are better thermalized. A major goal, as usual, is to achieve a relatively flat power distribution. The average power generation density in the core is about 51 kW/liter.

The coolant flow rate is about 105 million pounds per hour (13 Mg/s); the feed-water temperature is about 376 °F (191 °C), and water exiting the core is about 550 °F (288 °C), maintaining the clad temperature below 600 °F (316 °C). (See Table 6-1 for reactor parameters.)

The core and associated equipment are contained in a large, steel reactor

TABLE 6-1.
Representative Characteristics of Boiling-Water Reactors

Core thermal power	3,579 MWth
Plant efficiency	34%
Plant electrical output (nominal)	1,220 MWe
Core diameter	193 in (4.9 m)
Core (or fuel rod) active length	150 in (3.8 m)
Core weight (fuel assemblies)	524,000 lb (238 Mg)
Core power density	54 kW/liter
Cladding material	Zircaloy-2
Cladding diameter (OD)	0.483 in (1.23 cm)
Cladding thickness	0.032 in (0.81 mm)
Fuel material	UO_2
Pellet diameter	0.410 in (1.04 cm)
Pellet height	0.41 in (1.04 cm)
Assembly array	8 X 8, with fuel channel enclosing array
Number of assemblies	748
Total number of fuel rods	46,376
Control rod type	"Cruciform" control rods inserted from the bottom between sets of four assemblies.
Number of control rods	177
Total amount of fuel (UO_2)	342,000 lb (155 Mg)
Fuel/coolant ratio	1/2.7, blades out; 1/2.5, blades in (cold)
Coolant	Water (two phase)
Total coolant flow rate	104×10^6 lb/hr (13 Mg/sec)
Coolant pressure	1,040 psia (7.0 MPa)
Coolant temperature (steam system design)	551 °F (288 °C)
Feed water temperature	420 °F (216 °C)
Average coolant exit quality (percent steam weight)	14.7%
Average clad temperature	579 °F (304 °C)
Maximum fuel central temperature	3,330 °F (1,832 °C)
Average volumetric fuel temperature	1,130 °F (610 °C)
Axial peaking factor	1.4 approx.
Design fuel burnup	28,400 MWd/Te
Fresh fuel assay	Average 2.8% ^{235}U (initial core: 1.7-2.1% ave.)
Spent fuel assay	0.8% ^{235}U, 0.6% $^{239,241}Pu$
Refueling sequence	Approximately one-fourth of the fuel per year to one-third per 18 months
Refueling time	188 hrs @ 100% efficiency
Vessel wall thickness min/max	5.7 in/6.46 in (14.5 cm/16.4 cm)
Vessel material	Manganese-molybdenum-nickel steel internally clad with 1/8 in austenitic stainless steel
Vessel diameter (ID)	19 ft 10 in (6.0 m)
Vessel height	71 ft (22 m)
Vessel weight (including head)	1,950,000 lb (884,500 Kg)

Source: General Electric Co. specifications.

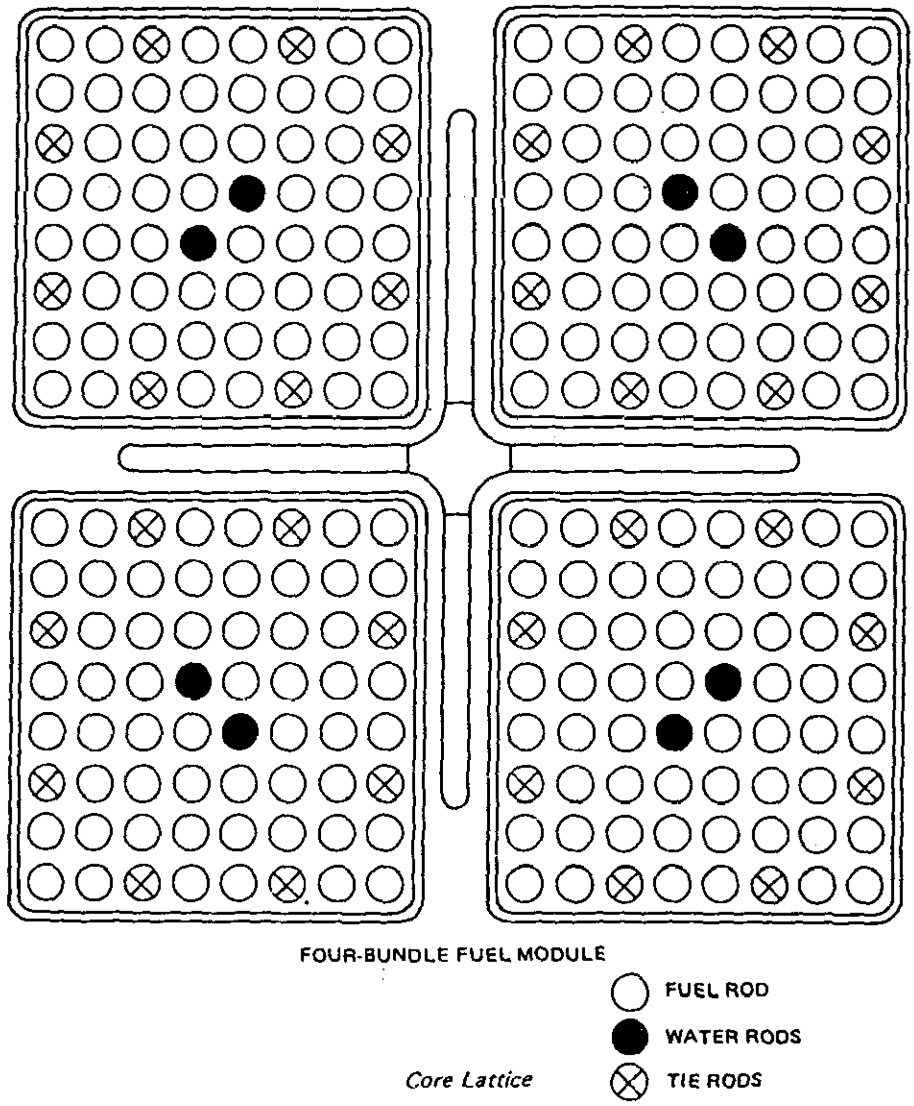


Figure 6-3. BOILING-WATER REACTOR CORE COMPONENTS.

CORE LATTICE

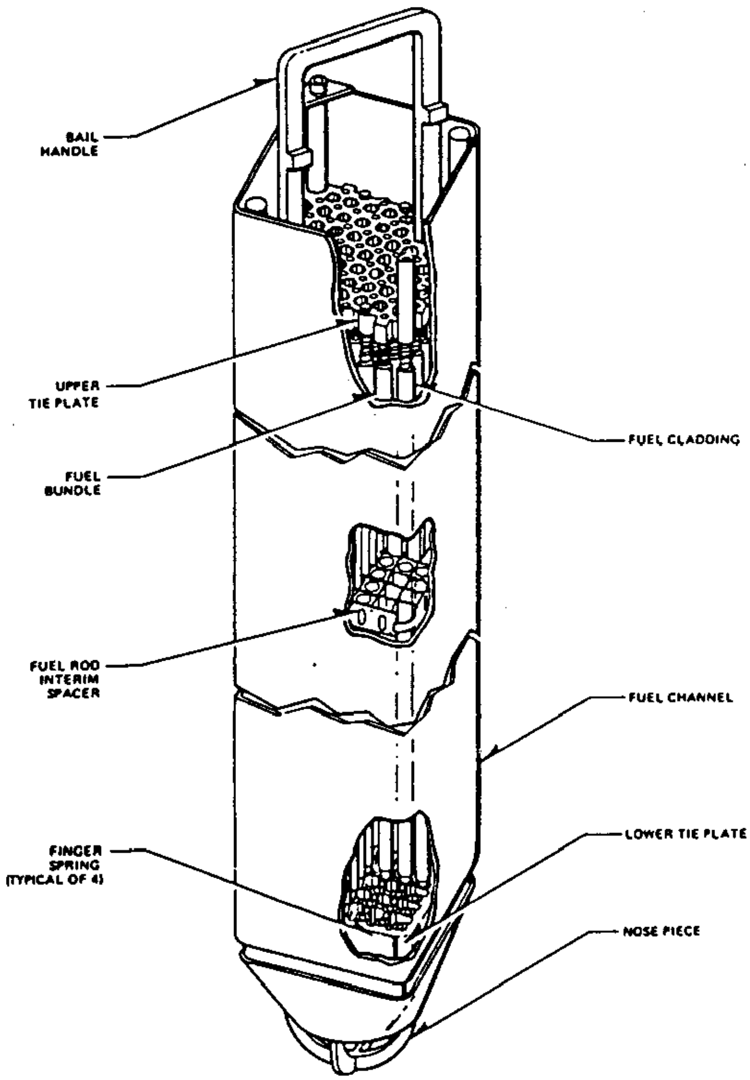
The basic module of a BWR core is a set of four fuel bundles, with a control assembly at the point where they meet. Note that some of the positions in the fuel assemblies are taken by tie rods and others are occupied by water rods, which serve to flatten the power distribution in the assembly.

FUEL ASSEMBLY

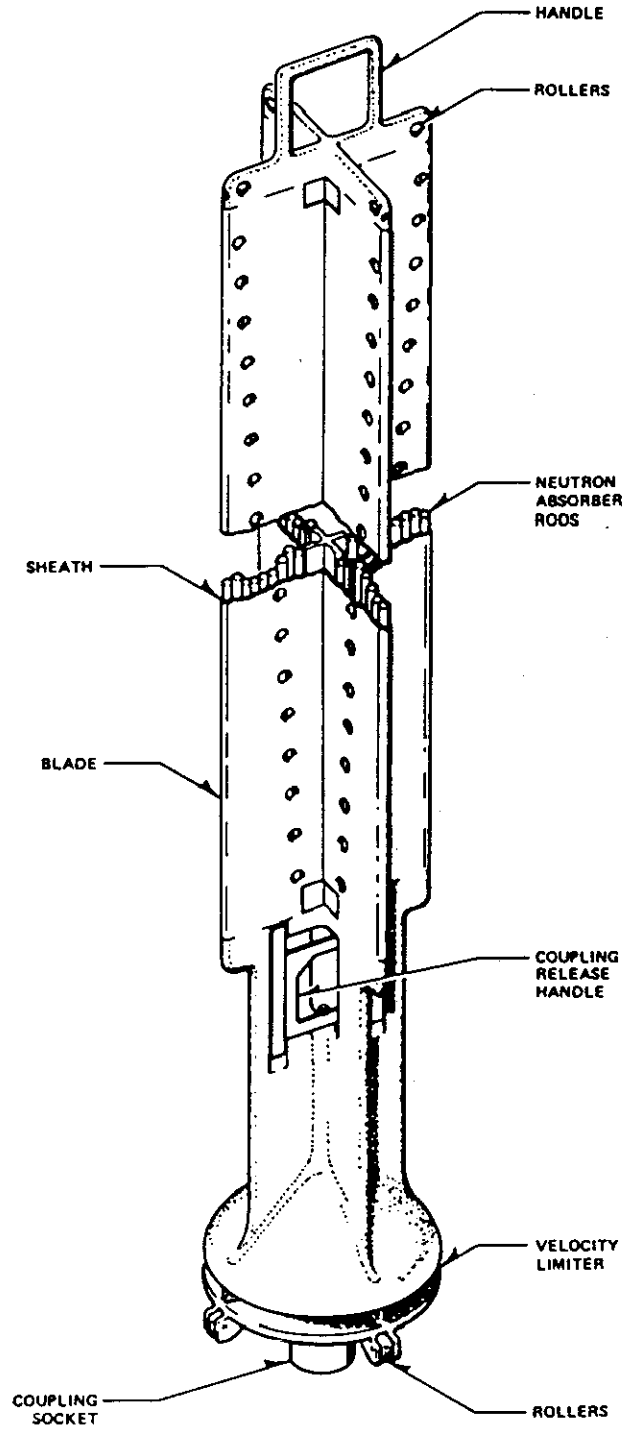
A BWR fuel assembly consists of a square array of fuel rods, held together by upper and lower tie plates and interim spacers, and surrounded by a fuel channel. The bottom of the assembly serves to regulate the flow through the assembly.

CONTROL ROD

The BWR control rod is a four-bladed assembly containing neutron absorber rods. This assembly is driven from the bottom of the reactor vessel. (Figure courtesy of General Electric Co.)



Fuel Assembly



Control Rod

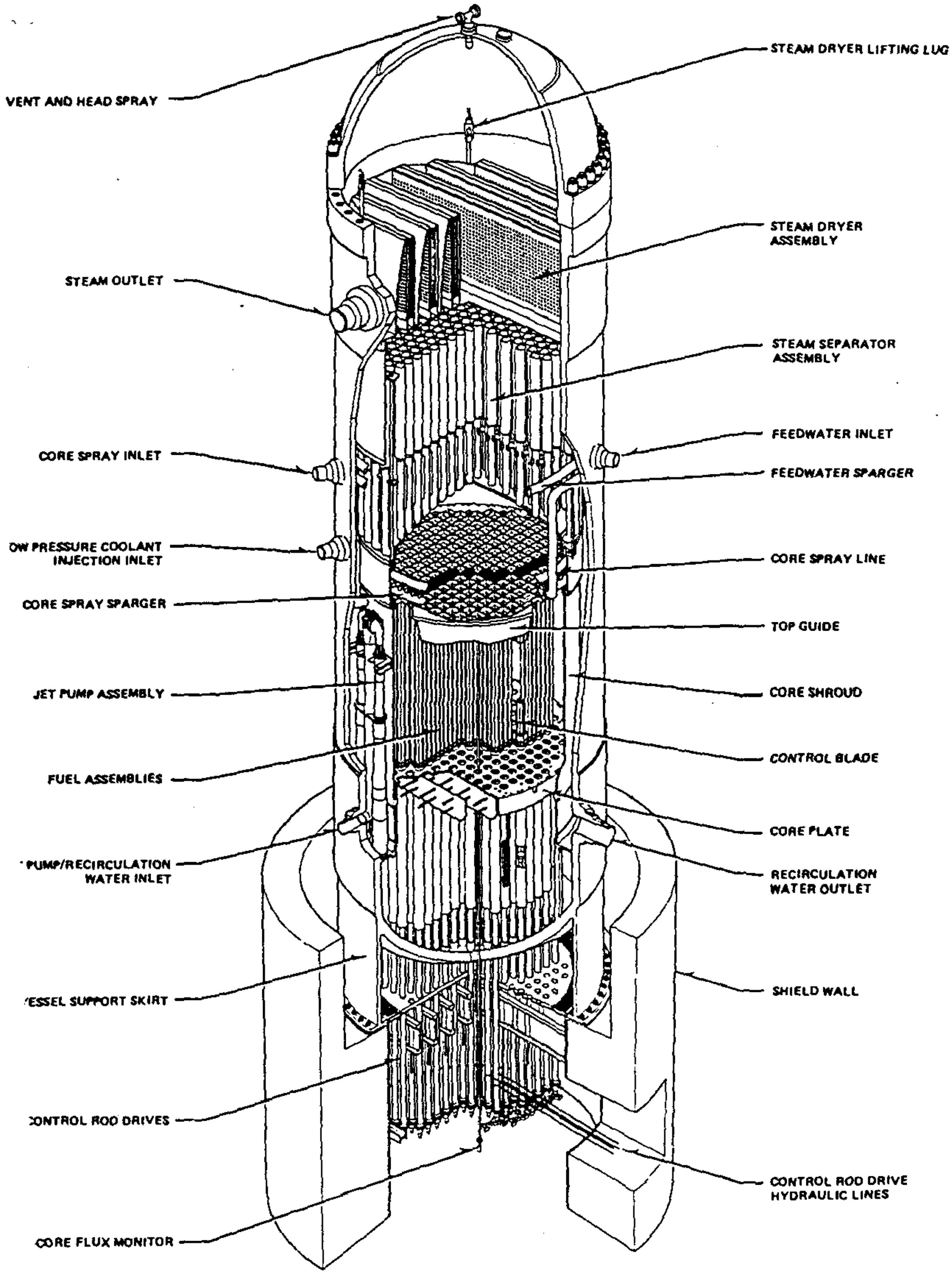
vessel (see Figure 6-4). In addition to the fuel assemblies, the other nuclear components of major interest are the control rods, which are mounted on the bottom of the reactor vessel, with drives below. The top head of the vessel is removable for refueling and contains no large equipment. Above the core are steam separators and dryers, comparable to devices in a PWR steam generator. The vessel containing all this equipment is very large, about 72 feet (22 m) in height and 21 feet (6 m) in diameter for a large BWR. It is made of carbon steel, 6 to 7 inches (16 cm) thick and all but the top, which comes into contact only with high-quality steam, is clad with 1/8 in. (0.3 cm) stainless steel. The vessel can withstand pressures greater than 1000 psi (7 MPa) at operating temperatures.

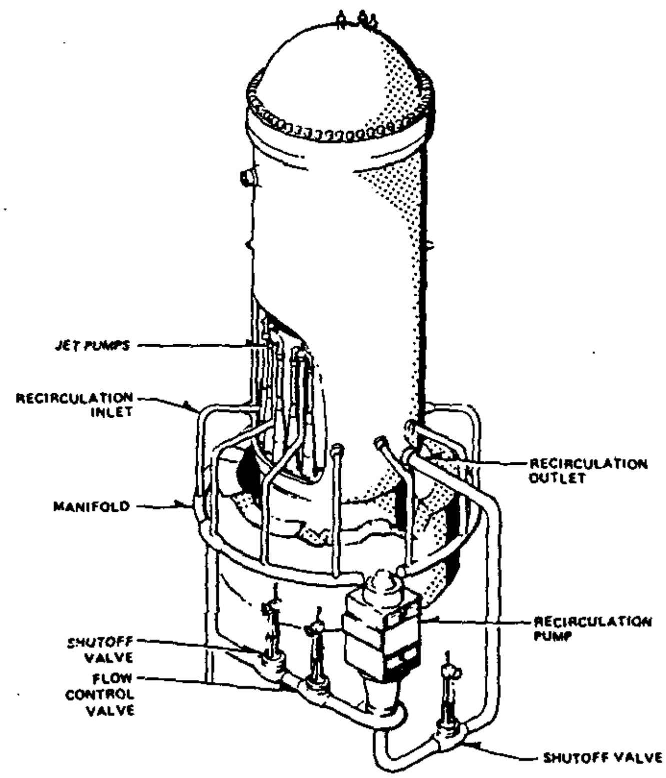
As suggested by Figure 6-1, the water in the vessel boils as it rises through the core. The BWR system is maintained at a pressure of about 1000 psi (7 MPa), at which pressure water boils at a temperature of 545 °F (285 °C). Of course, not all the water passing through the core is vaporized. About 13% (by weight) of the fluid leaving the core is steam. The remainder is recirculated down an annulus formed between the core "shroud" and the reactor vessel, to the plenum beneath the core. The fluid then passes up again through the core.

The steam generated is separated from the remaining liquid by a structure of steam separators which are positioned above the core, at the interface between the predominately liquid and gaseous phases. Steam from the separators then passes through a dryer assembly which removes moisture. The dried steam proceeds out of the vessel, through the drywell wall and reactor building (see below), to the turbogenerator. (Unlike the PWR system, the steam from a BWR – coming as it does directly from the core – is radioactive, primarily because of the presence of nitrogen 16, an isotope with a 7-second half-life.) Steam from the turbines is condensed and returned as feedwater to the reactor vessel, where it joins the flow recirculating to the bottom of the vessel. The thermal efficiency of a BWR is about 33%.

As we have indicated, most of the coolant recirculates within the reactor vessel, rather than in an external loop. This flow is pumped by a series of jet pumps in the annulus outside the core shroud. The jet pumps are basically reactor inlet nozzles for two external recirculation systems, each with a recirculation pump and associated valves and piping (see Figure 6-5). About one third of the core flow is taken from the reactor vessel and pumped through the manifold and jet pumps, thereby driving the annular flow as a whole. The water then turns upwards into the individually orificed fuel assemblies, as discussed above. The recirculation rate serves as one of the control systems. If the flow rate is decreased, a greater percentage of the water rising through the core is changed to steam, so that neutrons are less effectively moderated. The reaction rate and core power therefore tend to drop.

Figure 6-4. REACTOR ASSEMBLY OF A BWR POWER PLANT.
The reactor vessel of a BWR contains not only the core assembly but also devices for separating and drying steam. This steam is generated as the coolant flows up through the core. As the remaining liquid returns along the outside of the core, a portion of it is drawn off to the recirculation system and returned through the jet pumps, which thereby cause the bulk recirculation within the reactor vessel. (Figure courtesy of General Electric Co.)


STEAM DRYER LIFTING LUG
VENT AND HEAD SPRAY
STEAM DRYER ASSEMBLY
STEAM OUTLET
STEAM SEPARATOR ASSEMBLY
FEEDWATER INLET
CORE SPRAY INLET
FEEDWATER SPARGER
OW PRESSURE COOLANT INJECTION INLET
CORE SPRAY LINE
CORE SPRAY SPARGER
TOP GUIDE
JET PUMP ASSEMBLY
CORE SHROUD
CONTROL BLADE
FUEL ASSEMBLIES
CORE PLATE
PUMP/RECIRCULATION WATER INLET
RECIRCULATION WATER OUTLET
'ESSEL SUPPORT SKIRT
SHIELD WALL
:ONTROL ROD DRIVES
CONTROL ROD DRIVE HYDRAULIC LINES
CORE FLUX MONITOR



'igure 6-5. JET PUMP RECIRCU-
.ATION SYSTEM.
Water from a recirculation outlet is pumped back into the reactor vessel via several jet pumps, thereby driving the coolant within the reactor vessel. (Figure courtesy of General Electric Co.)

AUXILIARY SYSTEMS

Like the other water-cooled reactors, the BWR has systems for controlling water chemistry (and volume) and for removing decay heat. In the following brief discussion, aspects of these systems that are peculiar to the BWR are emphasized.

The coolant cleanup system removes fission products, corrosion products, and other impurities from a stream of water that is drawn off via the recirculation pump line and returns via the feedwater line. Cleaning is accomplished by filter-demineralizer units. In addition to performing a cleaning function, this system is also used to remove the excess water volume caused by lowering of the coolant density (due to boiling) as the reactor is brought up to power.

Decay heat removal after reactor shutdown is accomplished by a residual heat removal (RHR) system that is largely a part of the emergency core cooling system discussed below. Decay heat removal is ordinarily accomplished by drawing water from the recirculation line, cooling it in a heat exchanger, and returning it to the feedwater line. Emergency functions of the RHR system are discussed below.

Of the various other BWR systems, most provide basic services such as power, component cooling, and system control. The only system that is peculiar to the BWR is the system for cleaning and cooling the fuel storage and containment pools. The containment pools are a distinctive aspect of BWRs and are discussed in the context of safety design.

SAFETY SYSTEMS

The basic containment configuration for BWRs is shown in Figure 6-6, a schematic drawing of the Mark III containment and shield building. The reactor vessel and immediately associated equipment, such as the recirculation system and the pressure relief valves on the main steam lines, are enclosed in a drywell, which seals the reactor from the rest of the reactor building. The atmosphere in the drywell is in contact with a pressure suppression pool which forms an annulus around the drywell. In recent designs (Figure 6-6), the drywell is a concrete structure, and the suppression pool is on the floor of the reactor building between the containment liner and the drywell wall. The pool connects to the interior of the drywell through horizontal vents, but is prevented from covering the drywell floor by a "weir" wall; an upper containment pool sits atop the drywell. In earlier designs (Figure 6-7), the drywell consists of a steel primary containment, and the pressure suppression pool (with large numbers of downcomer tubes) is contained in a large torus connected to the drywell by several large vent pipes. In either case, blowdown of the reactor coolant inventory into the drywell tends to raise the pressure, thus forcing fluid into the pressure suppression pool. There steam is condensed, thus controlling the pressure increase.

In current designs, a steel containment shell surrounds all the equipment of the reactor building. This containment provides a sealed barrier against radioactive releases and is designed to withstand temperatures and pressures that could be caused by a loss-of-coolant accident. Surrounding the containment is the reactor building itself, a reinforced concrete structure which further limits radioactive releases and also protects the containment from external agents (weather, missiles).

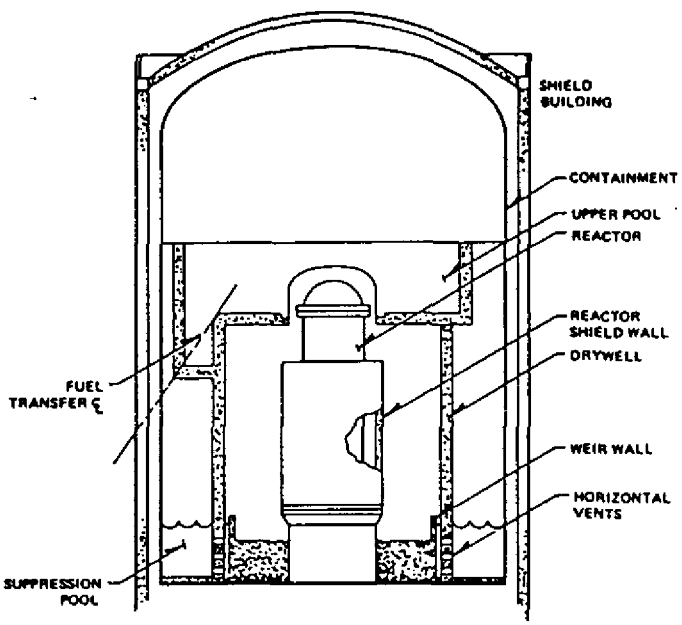
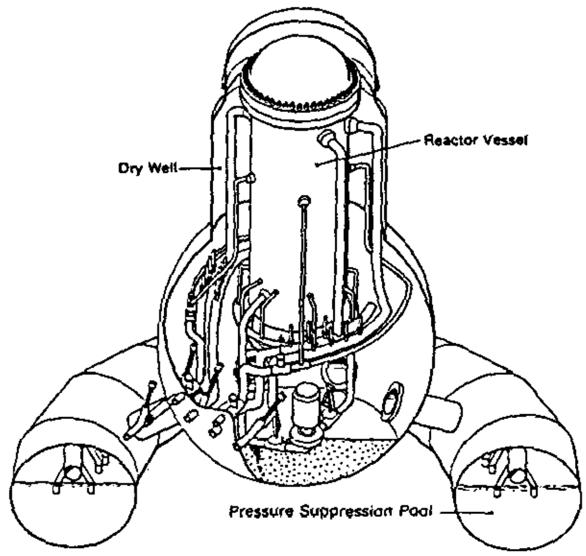


Figure 6-6. BWR MARK III CONTAINMENT AND SHIELD BUILDING.
The BWR reactor vessel is contained within a concrete drywell, which in turn is contained within a reactor building with a steel containment. The drywell is surrounded by a pressure suppression pool, which communicates with the drywell interior through horizontal vents. There is, in addition, a pool above the reactor. (Figure courtesy of General Electric Co.)

Figure 6-7. BWR MARK I PRIMARY CONTAINMENT.
In older versions of the BWR, the reactor vessel is enclosed in a dry well which communicates, via vent pipes and a downcomer system, with a pressure suppression pool contained in a large torus. This entire structure is contained in a reactor building. (Figure reproduced from WASH-1250.)



The annulus between the building and containment is maintained at negative pressure to serve as a collector of radioactivity during accident conditions. The atmosphere of this annulus is filtered to collect suspended radioactive materials.

Numerous systems are available for controlling abnormalities. In the event that control rods cannot be inserted, liquid neutron absorber (containing a boron compound) may be injected into the reactor to shut down the chain reaction. Heat removal systems are available for cooling the core in the event the drywell is isolated from the main cooling systems. Closely related to the heat removal systems are injection systems for coping with decreases in coolant inventory.

Both abnormalities associated with the turbine system and actual loss of coolant accidents can lead to closing of the steam lines and feedwater line, effectively isolating the reactor vessel within the drywell. Whenever the vessel is isolated, and indeed whenever feedwater is lost, a reactor core isolation cooling system is available to maintain coolant inventory by pumping water into the reactor via connections in the pressure vessel head. This system operates at normal pressures and initially draws water from tanks that store condensate from the turbine, from condensate from the residual heat removal system, or, if necessary, from the suppression pool.

A network of systems performs specific ECC functions to cope with LOCAs. (See Figure 6-9.) These all depend on signals indicating low water level in the pressure vessel or high pressure in the drywell, or both. The systems include low-pressure injection, utilization of the RHR system, and high- and low-pressure core spray systems. The high-pressure core spray is intended to lower the pressure within the pressure vessel and provide makeup water in the event of a LOCA. In the event the core is uncovered, the spray can directly cool the fuel assemblies. Water is taken from the condensate tanks and from the suppression pool. On the other hand, should it become necessary to use the low-pressure systems, the vessel must be depressurized. This can be accomplished by opening relief valves to blow down the vessel contents into the drywell (and hence the suppression pool). Once this is done, the low-pressure core spray may be used to cool the fuel assemblies (drawing water from the suppression pool) or RHR low-pressure injection (again from the suppression pool) may be initiated, or both. The RHR system may also be used

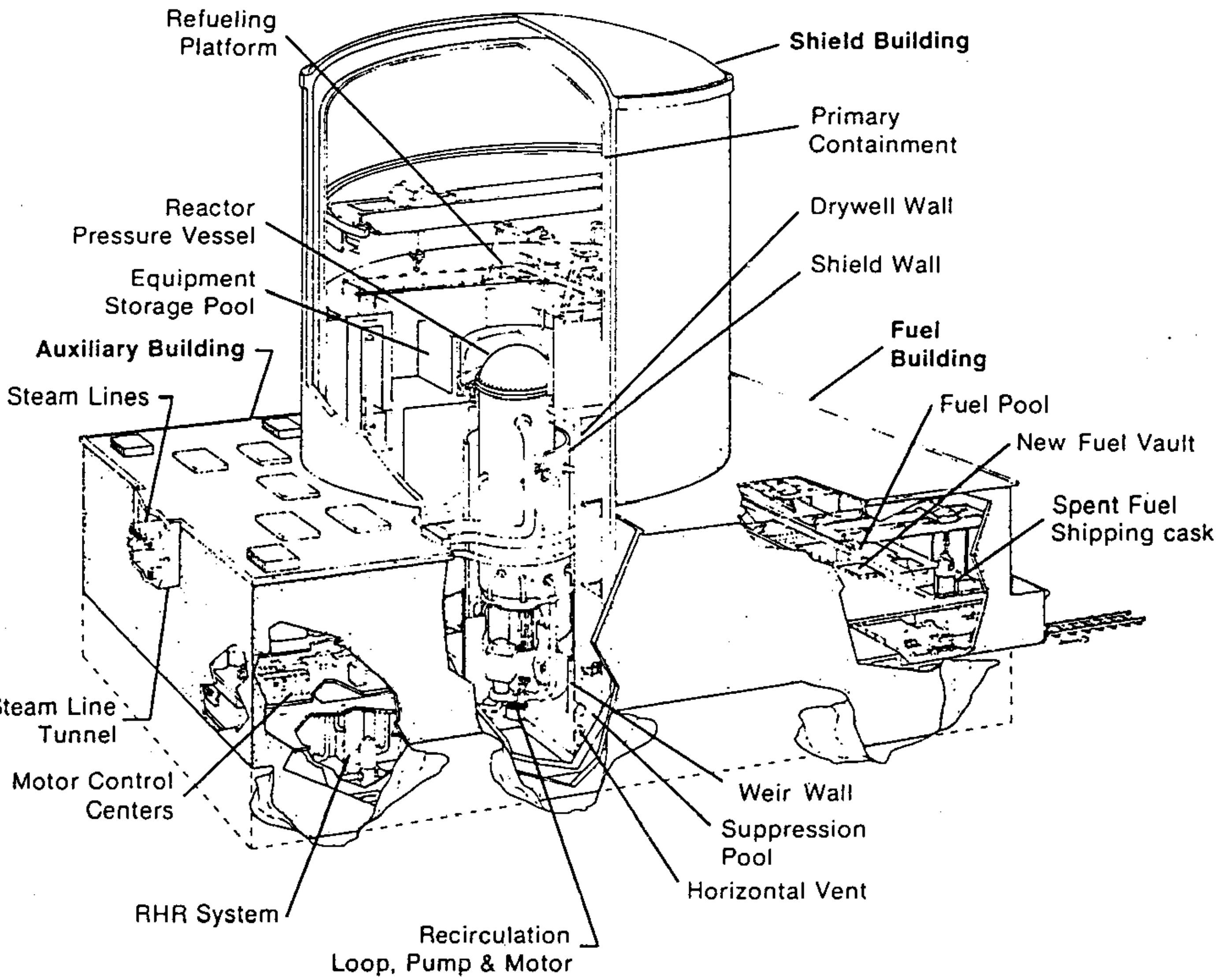
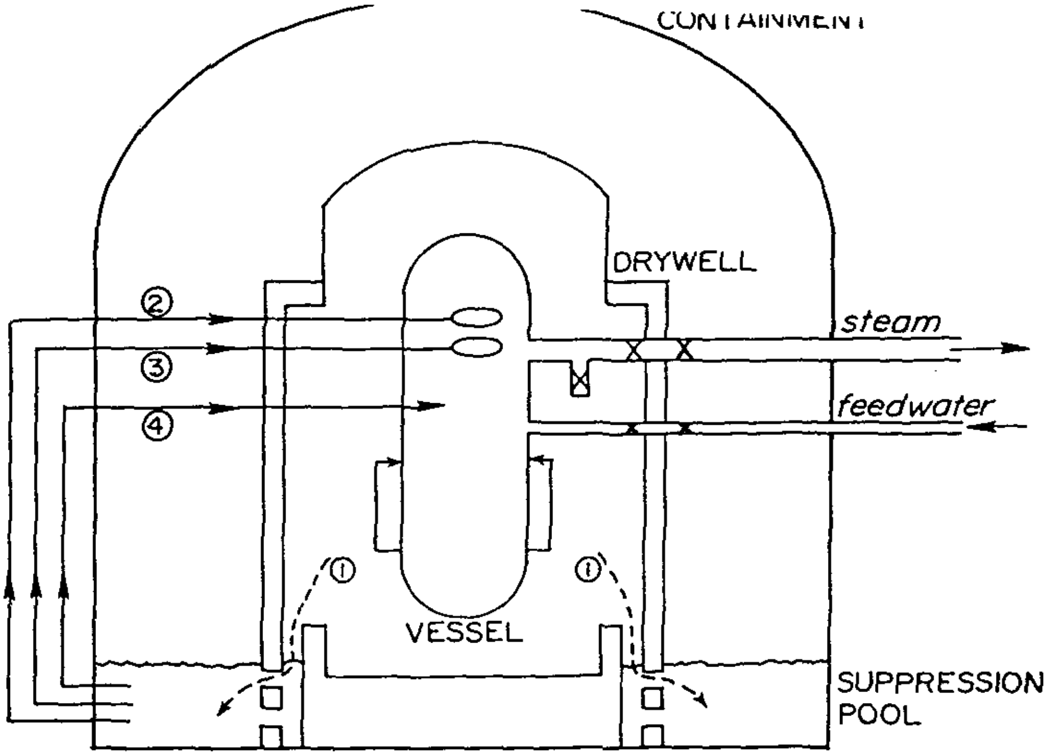


Figure 6-8. BWR REACTOR BUILDINGS.
Directly connected with the containment and shield building of a BWR are a fuel building and an auxiliary building. The turbine building is not shown. (Figure courtesy of General Electric Co.)



Emergency core cooling functions:

① overpressure injection into pressure suppression pool
② high pressure core spray
③ low pressure core spray
④ low pressure coolant injection

(x valves)

Figure 6-9. BWR EMERGENCY CORE COOLING SYSTEMS.
Several systems are available for supplying coolant to the core in the event that the basic BWR systems fail. The basic system for condensing and collecting coolant, thereby limiting drywell pressure, is the passive pressure suppression pool. In addition, active systems provide for high- and low-pressure core spray and for low-pressure coolant injection.

simply to cool the suppression pool. (Two other functions of the RHR are to provide decay heat removal under ordinary shutdown conditions and, when necessary, to supplement the cooling system for the spent fuel pool and the upper containment pool.)

The various ECC systems are thus designed to cool the core adequately under any conditions that are apt to occur. Ultimately, the water supply for any of the injection or spray systems is the suppression pool. This is also where reactor coolant losses should flow, so that a closed loop should exist. Thus the pressure suppression pool acts, not only to condense steam, thereby controlling containment pressure, but also to provide an emergency coolant reservoir.

BWR ECC systems have not been as controversial as those of the PWR, partly because the performance of the BWR spray systems, located above the core, is easier to analyze than the PWR ECC systems. However, it is interesting to note that the Reactor Safety Study (see Chapter 4) concluded that, within the uncertainties of their results, the risk from BWRs and PWRs were not markedly dissimilar. Moreover, the 1975 experience at Brown's Ferry, where burned cables led to a situation whereby coolant inventory was slowly being lost, showed that unexpected circumstances can circumvent multiple safety systems.

In many ways, the remarks at the end of the discussion of PWR safety systems (Chapter 5) apply equally well to BWRs. Both conservative and realistic models of emergency core cooling function exist, and as in any reactor system, great attention is given to assuring redundancy and independence of safety systems.

NEUTRONICS, FUEL UTILIZATION, AND REACTOR OPERATION

The neutronics and fuel utilization of a BWR are grossly similar to those of a PWR, for which the reader is referred to Chapter 5. As in a PWR, the actual burnup achieved by a BWR depends on how the reactor is operated. A BWR has a somewhat unusual capability for varying output to meet demand in that alteration of the coolant flow rate changes the reaction rate. This method of load following is not available to other types of reactors. A BWR also differs from a PWR in that it has a larger volume of fuel available for a given rated output. As a result, not only is the BWR power density lower, but the residence time of the fuel may be longer, particularly if comparable burnups are achieved. Since this appears to be the case (the BWR is designed for 27,500 MWd/Te versus the PWR's 32,000), refueling may not have to occur as often. General Electric does, in fact, cite one possible refueling sequence as replacing about one third of the core every 18 months. Moreover, on the newer systems (BWR/6 in Mark III containment) a refueling time of one week is specified. Older BWRs take a longer time.

In general, refueling entails opening the top of the drywell and removing the vessel head, steam dryers, and steam separators. The reactor well area is filled with water, and fresh and spent fuel bundles are exchanged in the upper containment pool area. A refueling tube connects this area with the fuel storage areas in the fuel building attached to the shield building (see Figure 6-8).

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Includes a brief description of BWR systems, including those related to LOCAs.

CHAPTER SEVEN

Heavy-Water Reactors

AN ALTERNATIVE to using ordinary water as the moderator and coolant of a thermal reactor is to choose "heavy" water for one or both of these purposes. Because heavy water absorbs fewer neutrons than ordinary water, heavy water moderated reactors (HWR) can be designed with natural uranium (0.7% ^{235}U) as the fuel. Moreover, because of the lower absorption and because the heavy water is a somewhat less effective moderator, it is feasible and advantageous to have larger separations between fuel bundles than in an LWR. This leads to the possibility of having individually cooled fuel channels, one bundle thick, with heavy-water moderator surrounding the channels. This is the basic configuration of commerical HWRs. These HWRs typically utilize a pressurized (as opposed to boiling) primary coolant system, so that a schematic of the reactor coolant and generating system is identical to that of a pressurized-water reactor (see Figures 1-4 and 5-1) except that the primary coolant may be heavy water. This is the case in the reactor now being marketed by Atomic Energy of Canada, Limited (AECL), the CANDU, for "Canadian deuterium-uranium" reactor. Most of the discussion in this chapter focuses on the CANDU, particularly its newer versions.

Although current CANDUs use heavy water, not only as the moderator, but also as the coolant, other cooling fluids are possible. Two that have been seriously considered, both in Canada and elsewhere, are light water and organic coolant. Light water is much less expensive than heavy water. Organic materials can operate at higher temperatures, thereby improving the thermal efficiency of the power plant.

In recent years, a significant portion of the British nuclear program has been directed to development of a "steam generating heavy-water reactor" (SGHWR). The SGHWR uses light-water coolant in vertical pressure tubes, which are immersed in heavy water moderator. The coolant is permitted to boil, and steam is separated in a steam drum, from which it goes to the turbine, as in a boiling-water reactor.

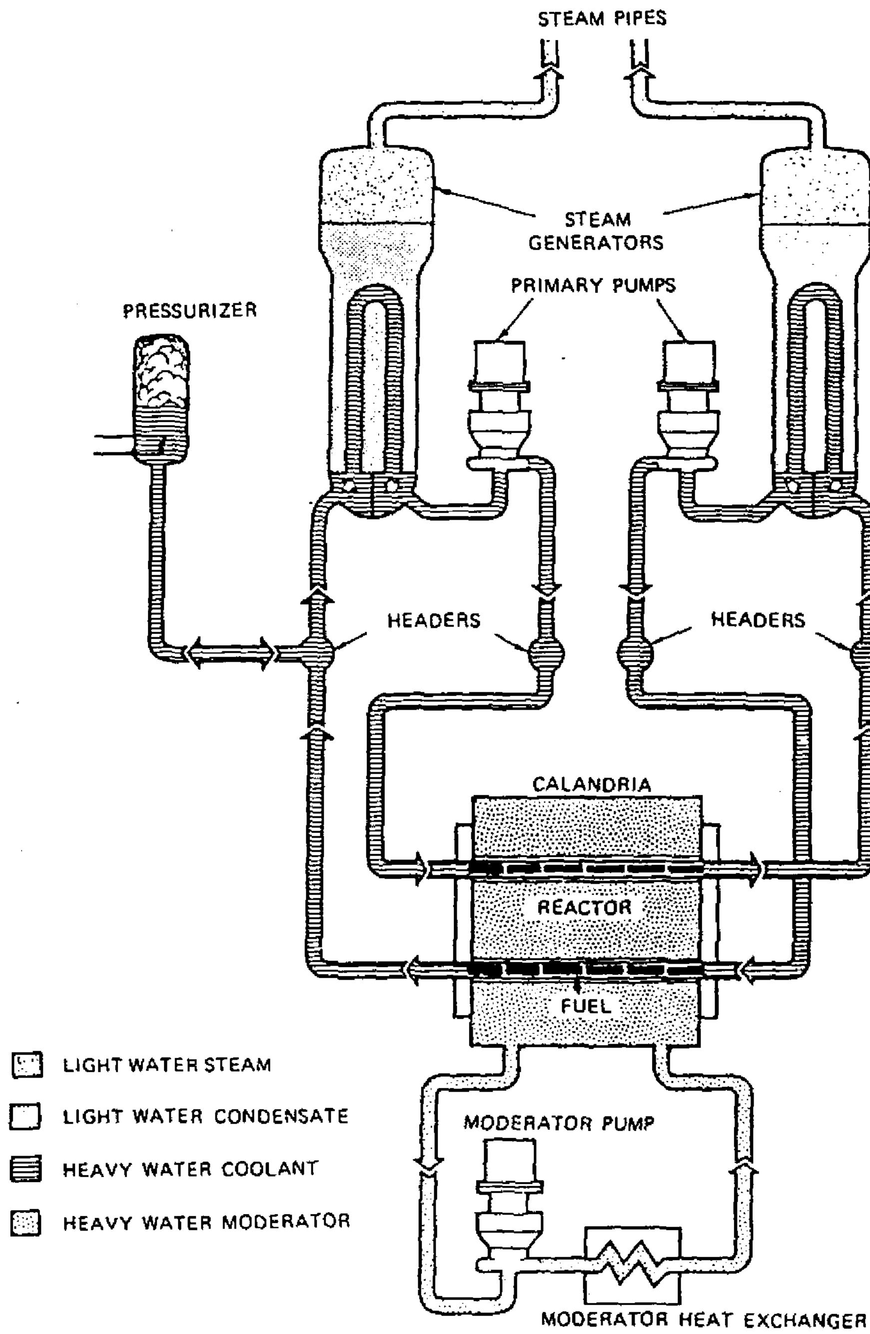


Figure 7-1. SCHEMATIC FLOW DIAGRAM FOR A CANDU POWER PLANT.
Present CANDU systems are essentially pressurized-water reactors. Individual fuel channels pass through a calandria, which contains heavy water moderator with its own circulation system. Heavy water coolant, on the other hand, flows through the fuel channels and raises steam (from ordinary water) in the steam generators. (Figure courtesy of Atomic Energy of Canada Ltd.)

The system uses slightly enriched uranium as its fuel. Britain has been developing the SGHWR as the basis of its nuclear power system. However, this choice is being reconsidered.

BASIC HWR SYSTEM

The important distinction, of course, between an LWR and an HWR is that the moderator of the latter is heavy water. In both of the reactor types cited, CANDU and SGHWR, a lattice of fuel channels is immersed in a pool of heavy-water moderator. The coolant passes through the channels and may be heavy water, light water, or some other fluid. In the case of the current CANDU, it is heavy water. A schematic diagram of the CANDU reactor and coolant system is given in Figure 7-1. Note that the fluid in the secondary loops, which drive the turbogenerators, is light water.

The fuel of a CANDU is similar to that of an LWR in that fuel pellets of uranium dioxide are sealed into Zircaloy-clad fuel pins, which are bound into bundles. A 600-MWe CANDU would have about 4500 bundles, containing about 100 tons (90 Mg) of uranium dioxide. However, in the case of the current CANDU, the uranium has only the natural concentration of ^{235}U, 0.7%. Moreover, the pins are arranged into bundles, shown in Figure 7-2, that are somewhat smaller and

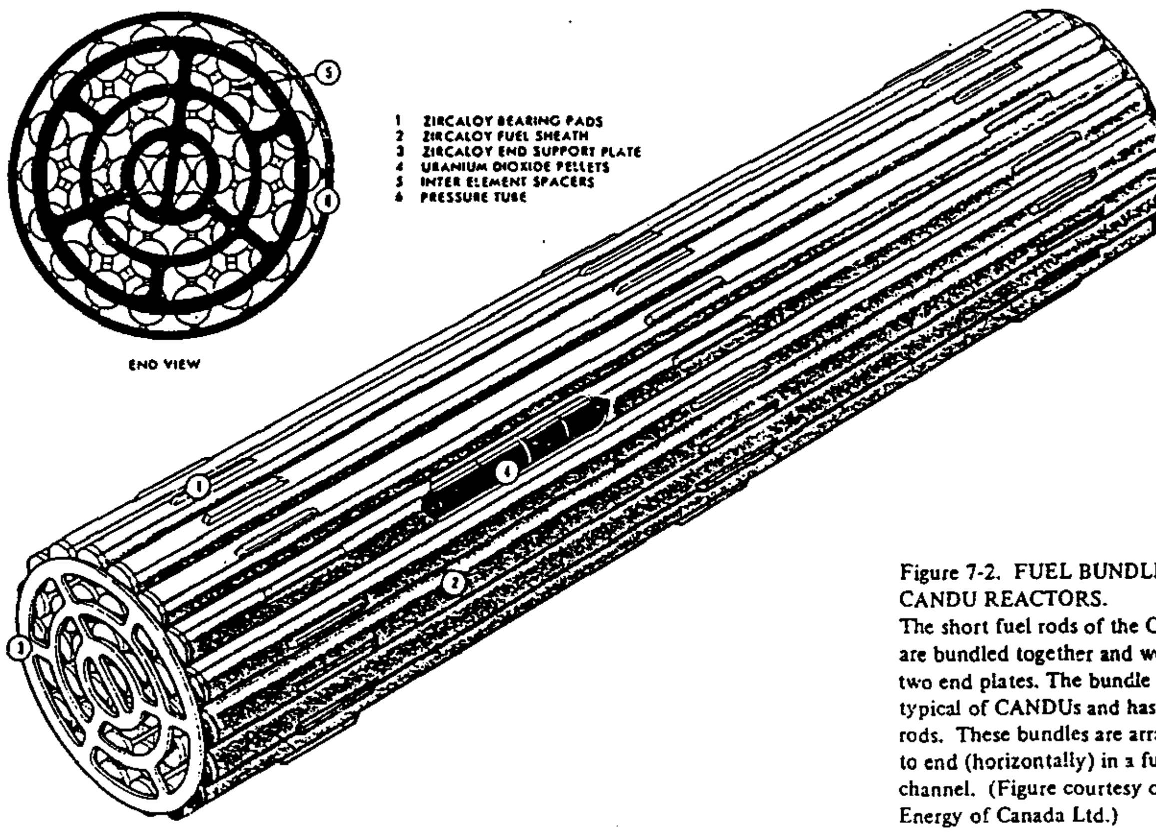
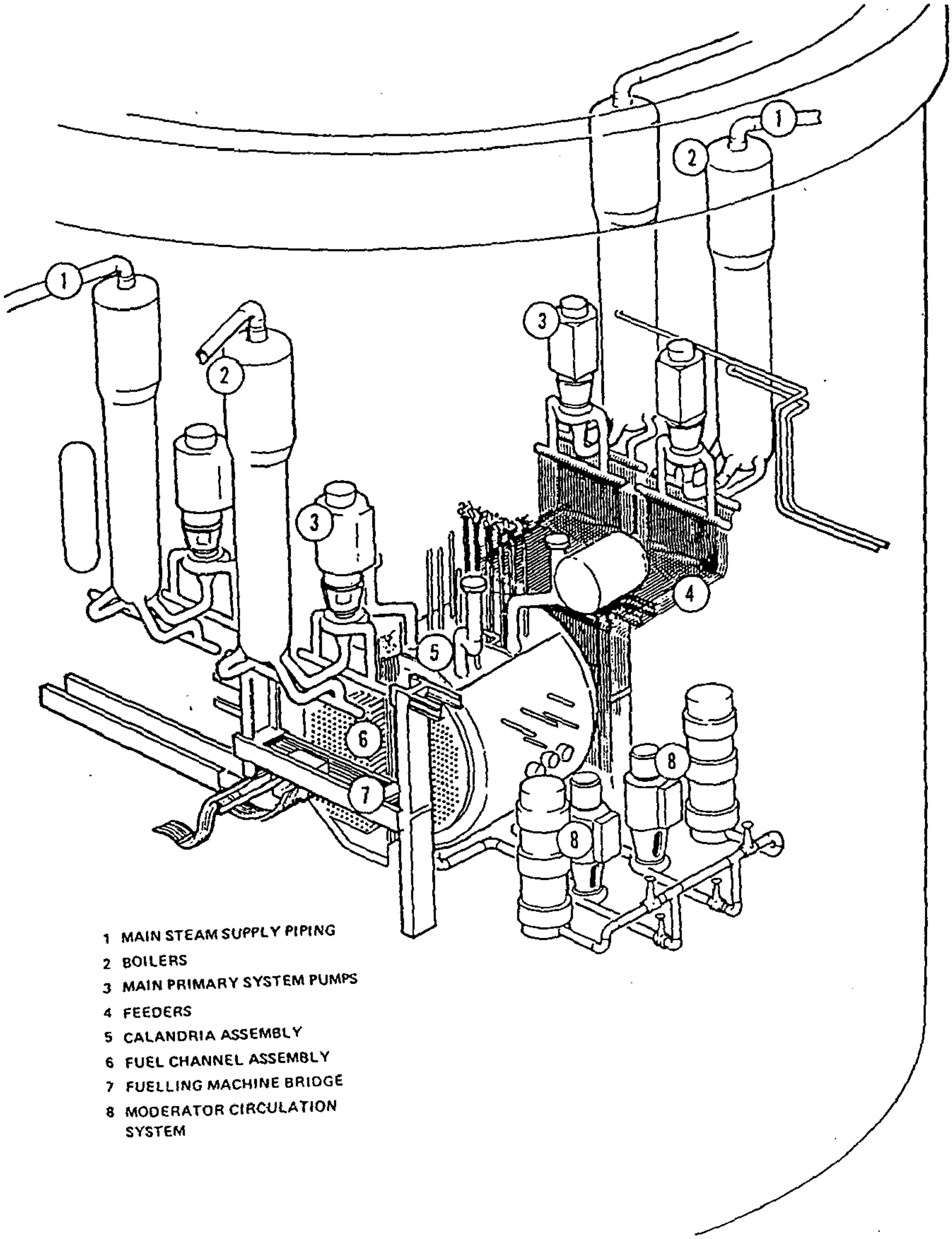


Figure 7-2. FUEL BUNDLE FOR CANDU REACTORS.
The short fuel rods of the CANDU are bundled together and welded to two end plates. The bundle shown is typical of CANDUs and has 37 fuel rods. These bundles are arranged end to end (horizontally) in a fuel channel. (Figure courtesy of Atomic Energy of Canada Ltd.)



1 MAIN STEAM SUPPLY PIPING
2 BOILERS
3 MAIN PRIMARY SYSTEM PUMPS
4 FEEDERS
5 CALANDRIA ASSEMBLY
6 FUEL CHANNEL ASSEMBLY
7 FUELLING MACHINE BRIDGE
8 MODERATOR CIRCULATION SYSTEM

simpler than those of LWRs. These bundles do not have hardware for maintaining the core configuration, a function that is performed by the fuel channels. Instead, the bundles and channels are designed for on-line refueling. On the average, about 15 bundles are replaced per day of operation, without shutting down the reactor. This has some advantage, perhaps, in that no refueling shutdown is necessary. However, its most important consequence from the point of view of reactor design is that relatively little neutron absorber is necessary during reactor operation, because there are no large swings in fissile content and fission product poisons during the fuel cycle. This leads to a higher conversion ratio and, under some conditions, to significantly improved resource utilization (see end of this chapter).

Figure 7-1 shows only two of the fuel channels. In an actual reactor, there are hundreds of channels, each with a row of fuel bundles arranged end to end. These fuel channels pass horizontally through a lattice of tubes which is part of a "calandria" which contains the moderator (see Figure 7-3). This moderator, heavy water, is maintained at near atmospheric pressure, so that this reactor system does not require fabrication of a large pressure vessel. The calandria is moderate in size, a cylinder about 25 feet (7.6 m) in diameter and 25 feet (7.6 m) long, made with stainless steel walls about 1 in. (2.5 cm) thick, and ends about 2 in. (5 cm) thick. The calandria tubes are made of Zircaloy. The moderator in the calandria has its own cooling system (including two pumps and two heat exchangers) which maintains moderator temperature at about 160 °F (70 °C). (See Table 7-1 for representative parameters.) During operation, the vault containing the calandria is filled with water.

The primary coolant system is similar to that of a PWR except that the pressure vessel is replaced by a lattice of hundreds of individual pressure tubes, each with a feeder at either end leading to headers at the pumps and steam generators. Individual pressure tubes may be opened during reactor operation for refueling. The tubes are fabricated from an alloy of zirconium and there is a gas space between the pressure tube and the surrounding calandria tube. The heavy-water coolant is maintained at a pressure of about 1500 psi (10 MPa) and, in passing through the pressure tubes, reaches a temperature of 590 °F (310 °C), below the boiling point at that pressure. The primary coolant flow pattern is relatively simple: coolant from a primary pump passes through a distribution header to the individual tubes, goes once through the reactor, through the header at the steam generator, and through the U-tube steam generator to the primary pump. The flow rate (600 MWe CANDU) is about 60 million pounds per hour (7.6 Mg/s). In the present CANDU (called a "pressurized heavy water reactor" for obvious reasons), there are four steam generators and pumps, paired to achieve the flow patterns shown in Figure

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Figure 7-3. PRIMARY SYSTEM FOR A CANDU REACTOR.
Numerous fuel channels pass through the CANDU calandria. Each is connected via its own pipes to the headers at a primary coolant pump and at a steam generator. There is, in addition, a circulation and cooling system for the moderator contained in the calandria. (Figure courtesy of Atomic Energy of Canada Ltd.)

TABLE 7-1.
Representative Characteristics of a CANDU Reactor[a]

Core thermal power	2,140 MWth
Plant efficiency	28%[a]
Plant electrical output	600 MWe
Core diameter	248 in (6.3 m)
Core length	234 in (5.9 m)
Core weight (fuel bundles)	240,000 lb (109 Mg)
Core power density	12 kW/liter (core average within calandria)
Cladding material	Zircaloy
Cladding diameter (OD)	0.515 in (1.31 cm)
Cladding thickness	0.016 in (0.04 cm)
Fuel material	UO_2
Pellet diameter	0.478 in (1.21 cm)
Fuel bundle array	37 rods, arranged in concentric circles
Array diameter (OD)	4 in (10 cm)
Total number of bundles	4,560
Total number of fuel rods	168,720
Total amount of fuel (UO_2)	210,000 lb (95 Mg)
Control rod types	Variable neutron absorbers (light-water compartments), adjustable absorbers (such as stainless steel); shutdown by absorbing rods or poison injection
Number of control rods or compartments	From 4 to 21 of each type of absorber
Coolant	Heavy water (liquid, plus some gas phase), >95% D_2O
Total coolant flow rate	60×10^6 lb/hr (7.6 Mg/s)
Coolant pressure (entrance to channel)	1,602 psi (11.1 MPa)
Coolant pressure (exit of channel)	1,493 psi (10.3 MPa)
Coolant temperature (entrance)	512 °F (267 °C)
Coolant temperature (exit)	594 °F (312 °C)
Average coolant exit quality	3%
Moderator	Heavy water, 99.75% D_2O (molecular ratio)
Moderator pressure	Approximately atmospheric
Moderator temperature (entrance)	110 °F (43 °C)
Moderator temperature (exit)	160 °F (71 °C)
Total heavy water inventory	1.02×10^6 lb (463 Mg)
Maximum fuel temperature	3,832 °F (2,110 °C)
Maximum clad temperature	684 °F (362 °C)
Axial peaking factor	1.5
Radial peaking factor	1.2
Fuel residence time	470 full-power days
Design fuel burnup	7,000 MWd/Te[a]
Fresh fuel assay	0.71% ^{235}U
Spent fuel assay	0.2% ^{235}U, 0.3% $^{239,241}Pu$
Refueling sequence	On-line, essentially continuous, refueling
Calandria outer diameter	25 ft (7.6 m)
Calandria length	25 ft (7.6 m)
Calandria wall thickness (stainless steel)	1-1/8 in (3 cm) thick walls, 2 in (5 cm) ends
Number of calandria tubes (Zircaloy)	380
Lattice array	Square with 11 in (28 cm) pitch

a. The detailed design varies from one reactor to another. In particular, newer models have slightly different dimensions, somewhat higher fuel burnup and efficiency.

SOURCE: Atomic Energy of Canada Ltd. specifications.

7-1. The system pressure is maintained by a single pressurizer, connected to the headers at two of the steam generators.

The secondary coolant fluid in a CANDU is light water. As in any steam power plant, this steam drives a turbine, is condensed, then returned to the boilers (steam generators) as feedwater. The overall thermal efficiency of a CANDU system is about 29%, significantly lower than that of most commercial nuclear power plants.

Reactivity control is achieved by several systems, including (light) water zone control absorbers, solid absorber rods, and poisons for addition to the moderator. (In some older models, control has been via highly enriched fuel rods, whose withdrawal reduces the reactivity.) In current CANDUs, routine on-line control is accomplished by the zone absorbers, which consist of compartments in the core into which light water, a neutron absorber, can be introduced. In addition, several mechanical control rods (containing cadmium) supplement this control and can be dropped under gravity for quick power reduction. Two banks of about 14 cadmium control rods are available specifically for reactor shutdown. Long-term reactivity control and startup reactivity control, respectively, are provided by neutron absorbing compounds of boron and gadolinium in the moderator. Finally, core power shaping is achieved by stainless steel adjuster rods. In addition, the power distribution can be effectively controlled by the refueling sequence, since only one pressure tube is serviced at a time.

AUXILIARY SYSTEMS

Systems are available for performing important service functions for the main system, including chemistry and volume control and shutdown cooling. These are similar to those for a PWR except for the differences required by the separate moderator and coolant systems.

The moderator cleanup system controls impurities and includes the capability for removing boron and gadolinium neutron poisons. The coolant purification system takes flow from a primary pump outlet and returns it to the pump inlet; the system uses filtering and ion exchange for removing impurities. The coolant volume control system is closely linked with the pressurizer and has enough capacity to handle all changes in coolant volume associated with alterations in power level. Because of the expense of heavy water (about $100/kg), the reactor building contains systems for the collection, purification, and upgrading of heavy water, in order to minimize inventory losses.

Two shutdown cooling systems connect to the reactor inlet and outlet headers, essentially in parallel with the primary pumps and steam generators. As the reactor cools down, these systems, each with a pump and heat exchanger, gradually take over decay cooling. Initially, pumping force through the heat exchangers is provided by the primary pumps, but, as the coolant temperature decreases, shutdown pumps assume this function and the primary pumps and steam generators are isolated.

SAFETY SYSTEMS

Under abnormal conditions, the first action is to shut down the reactor. This is accomplished by gravity drop of the shutdown control rods. For cases where these rods could not be inserted, earlier CANDUs had provision for dumping the moderator out of the calandria into a large tank. In current versions this capability is replaced by a fast-acting system for injecting gadolinium into the moderator.

The CANDU has an emergency core cooling system for controlling loss-of-coolant accidents. Should a reactor coolant system rupture, valves close to isolate the intact system, and light water from a storage tank (dousing tank) built into the roof of the containment system is injected into the ruptured system. Heat is initially rejected through the steam generators. As the dousing tank is emptied, water is recovered from the bottom of the reactor building, passed through a heat exchanger, and reinjected into the ruptured system. The moderator in the calandria provides some independent heat capacity, with heat removal provided by the heat exchangers in the moderator circulation system.

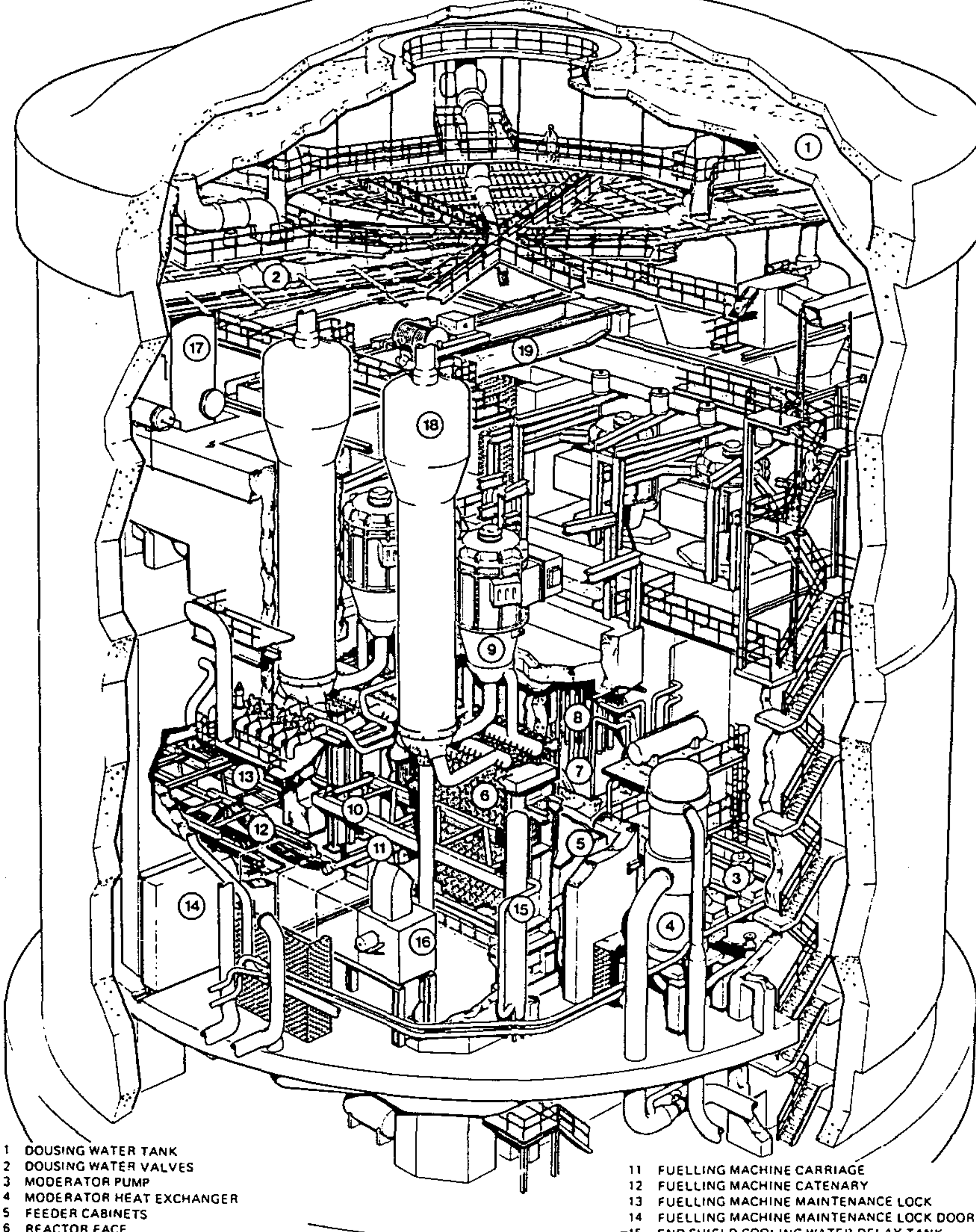
A design with many pressure tubes has an advantage in that gross failure of the pressure vessel is not possible. On the other hand, a large LOCA can still occur; for example, one of the headers could be ruptured. However, the other independent coolant loop would presumably still be intact. Furthermore, in the extreme case where all the coolant was lost and the ECC system failed, although the fuel and pressure tubes would be severely damaged, the moderator could carry off enough heat to prevent gross melting.

The containment structure (Figure 7-4) is a prestressed concrete building with a plastic liner. Its subsystems include a spray system and air coolers for reducing the building pressure. In some designs, the containment atmosphere is ordinarily at negative pressure with respect to the external environment.

NEUTRONICS, FUEL UTILIZATION, AND REACTOR OPERATION

Heavy-water reactors have an advantage over LWRs in that relatively few neutrons are lost to absorption by the moderator. CANDUs in particular have the advantage of on-line refueling. These two effects are the most significant factors in permitting design of a reactor with a conversion ratio (CR) that approaches 0.75 to 0.80. The fact that (1 - CR) is only 0.20 to 0.25 means that operation of a CANDU requires significantly less resource depletion than an LWR, for which (1 - CR) is approximately 0.4. However, this advantage is fully realized only if fissile material in the spent fuel is recovered. If not, the resource utilization of a CANDU is comparable to, or somewhat poorer than, that of an LWR with fissile recycle. The

Figure 7-4. CANDU REACTOR BUILDING.
A reactor building contains the entire primary system of a CANDU, as well as various safety-related systems. The building itself is concrete with a plastic liner. (Figure courtesy of Atomic Energy of Canada Ltd.)



1 DOUSING WATER TANK
2 DOUSING WATER VALVES
3 MODERATOR PUMP
4 MODERATOR HEAT EXCHANGER
5 FEEDER CABINETS
6 REACTOR FACE
7 REACTOR
8 REACTIVITY MECHANISM
9 PRIMARY HEAT TRANSPORT SYSTEM PUMP
10 FUELLING MACHINE BRIDGE
11 FUELLING MACHINE CARRIAGE
12 FUELLING MACHINE CATENARY
13 FUELLING MACHINE MAINTENANCE LOCK
14 FUELLING MACHINE MAINTENANCE LOCK DOOR
15 END SHIELD COOLING WATER DELAY TANK
16 VAULT COOLER
17 PRESSURIZER
18 STEAM GENERATOR
19 60-TON CRANE

TABLE 7-2.
Approximate CANDU Neutronics (equilibrium cycle)

Approximately 2.1 *fast neutrons* are produced following the absorption of 1 neutron by fissile material and have the following fate:

0.79[a]	Captured by fertile material, leading to fissile production
1	Absorbed by fissile material (of which 0.8 result in fission)
0.02	Absorbed by heavy water
0.22	Absorbed by structural and fission products
0.06	Absorbed by other materials, including control poisons
0.04	Lost by leakage
2.13	

a. The conversion ratio is therefore 0.79 for this system. However, this high a ratio has not yet been achieved for the CANDU; 0.70 to 0.75 is typical.

full potential of a CANDU is realized only if it is operated near break-even on a thorium cycle (see Chapter 14).

To indicate the manner in which neutrons are used in a CANDU, Table 7-2 summarizes the neutrons produced as the result of one thermal neutron absorption in fissile material. As in an LWR, about two fast neutrons ultimately result, and their final disposition differs from that in an LWR (Table 5-2) in subtle, but important, ways. Note that the conversion ratio, the ratio of fissile material produced to fissile material destroyed, is 0.79. This is possible largely because, of the 2.1 neutrons resulting from absorption by fissile, less than 0.1 are lost to absorption by moderator and control. (This contrasts with 0.3 for LWRs, as noted in Chapter 5.)

The fissile content of fresh fuel in a CANDU is only 0.7%. Not surprisingly, the design burnup is much less than in LWRs – about 8000 MWd/Te. It is interesting to note, too, that the fissile content of the discharged fuel is about 0.5%, slightly more than half of which is fissile plutonium. Whereas the lifetime uranium commitment to a CANDU (1000 MWe) would be about 4200 tons of U_3O_8[1] on a throwaway fuel cycle (see Table 10-1), this would be reduced by about half were the plutonium to be recycled. However, so much more material must be reprocessed and fabricated that, from an economic point of view, there is much less incentive to recycle plutonium in a CANDU than in an LWR.

The fact that CANDUs are continuously refueled offers a clear advantage in fuel management. The utility is never faced with the decision whether to refuel on schedule even when the fuel has not reached design burnup. Fueling can take place as needed, so that the maximum energy may be extracted from the fuel. In a way, the refueling machine acts as a reactivity control, increasing the fissile content precisely when it is required. The on-line refueling may also reduce outage time, but the extent of such reduction is not clear since, during refueling shutdowns, other types of power plants are also serviced in other ways. A disadvantage of on-line

[1] This assumes a burnup of 9600 MWd/Te, a goal that has not yet been achieved.

refueling is that inspection to monitor diversion of nuclear materials (see Chapter 12) becomes more difficult.

Having mentioned economics above, we might go on to note two other such factors. The fact that CANDUs do not require enriched uranium significantly reduces CANDU fuel cycle costs relative to those of LWRs. However, the need for a million pound heavy water (actually 0.4 Mg/MWe) inventory, mostly at the start of the operation, substantially raises the initial cost of the power plant, so that these two characteristics of the CANDU tend to balance one another.

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CHAPTER EIGHT

Gas-Cooled Thermal Reactors

AN IMPORTANT ALTERNATIVE to hydrogen as the moderator in a thermal reactor is carbon. As noted in Appendix B, carbon, with atomic mass 12, requires more collisions to slow down neutrons than does water (see "slowing down power" on Table B-1), but it also absorbs a smaller proportion of neutrons. Because the "moderating ratio," a measure of the slowing down power relative to the absorption, is even better for carbon than light water, designing a reactor with a relatively large mass of carbon can be very effective neutronically. This approach has been taken in numerous reactor systems, including the earliest reactors, which utilized natural uranium as the fuel. In most cases, the coolant in a carbon-moderated reactor is a gas, such as helium or carbon dioxide, but this is by no means necessary: in many Russian carbon-moderated reactors, the coolant is water confined to pressure tubes; the molten salt breeder reactor (Chapter 14) immerses carbon in a liquid fuel salt.

Several gas-cooled carbon-moderated commercial nuclear power plants have been designed. In Great Britain, a number of carbon dioxide cooled reactors have actually been built; this "advanced gas reactor" (AGR) is sometimes considered as an alternative to the SGHWR (Chapter 7). In the United States, the General Atomic Company has built one 330-MWe gas-cooled reactor, but the larger commercial versions were withdrawn from the market in 1976. A similar reactor, but with a "pebble bed," is being developed in Germany (Chapter 14). Interest in these reactors survives, largely because a high-temperature gas coolant offers the potential for high thermal efficiency, particularly in a direct cycle with a gas turbine, and for industrial process heat production. Moreover, the level of interest has risen in connection with the search for more proliferation resistant nuclear systems (see Chapters 12 and 14).

The reactor offered by General Atomic affords a good opportunity to examine the features of gas-cooled reactors. This reactor, called a "high-temperature gas-cooled reactor" (HTGR), uses helium coolant and a core consisting of stacked carbon blocks with small uranium-thorium fuel regions. The basic heat transfer diagram of this reactor (see Figure 8-1) is similar to that of a PWR, except that the primary system of an HTGR contains helium, not water, and the core consists of stacked carbon blocks, not metal fuel rods. Details are given in the rest of this chapter.

BASIC HTGR SYSTEM

The HTGR differs in two major respects from the reactors described in previous chapters. The fuel/moderator system is radically different, since the fuel consists of uranium and thorium pellets contained in fuel regions of carbon moderator blocks. The primary coolant system is distinctive, both because the coolant is a gas, helium, and because the entire primary coolant system is contained in a large prestressed concrete reactor vessel (PCRV), as indicated by the dashed line on Figure 8-1. The general appearance of the core and the physical layout of the primary system are shown in Figure 8-2.

The HTGR core consists of a massive pile of hexagonal graphite blocks, each containing fueled regions, as well as holes for passage of the pressurized helium gas. The fuel itself consists of highly enriched uranium as the fissile material and thorium as the fertile. These fuels, in the form of the dioxide or carbide, both ceramics, are present as small fuel kernels with ceramic coatings. The two types of pellet, shown in Figure 8-3, have different coatings in order to facilitate separation at reprocessing: the fissile pellets, with uranium enriched to 93% ^{235}U, or with recycled ^{233}U, are coated with pyrolitic carbons and silicon carbide; the fertile

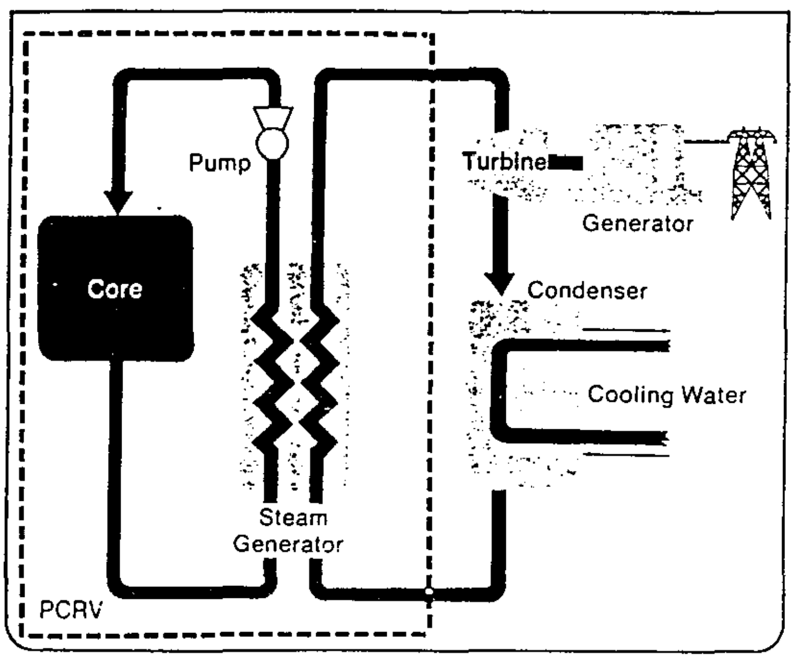


Figure 8-1. SCHEMATIC OF HIGH-TEMPERATURE GAS-COOLED REACTOR POWER PLANT.
The core of an HTGR is mostly carbon, with uranium and thorium fueled regions. Heat from the core is carried off by helium coolant to steam generators. The core, steam generators, helium circulators, and other equipment are contained in a prestressed concrete reactor vessel (PCRV). (Figure reproduced from ERDA-76-107.)

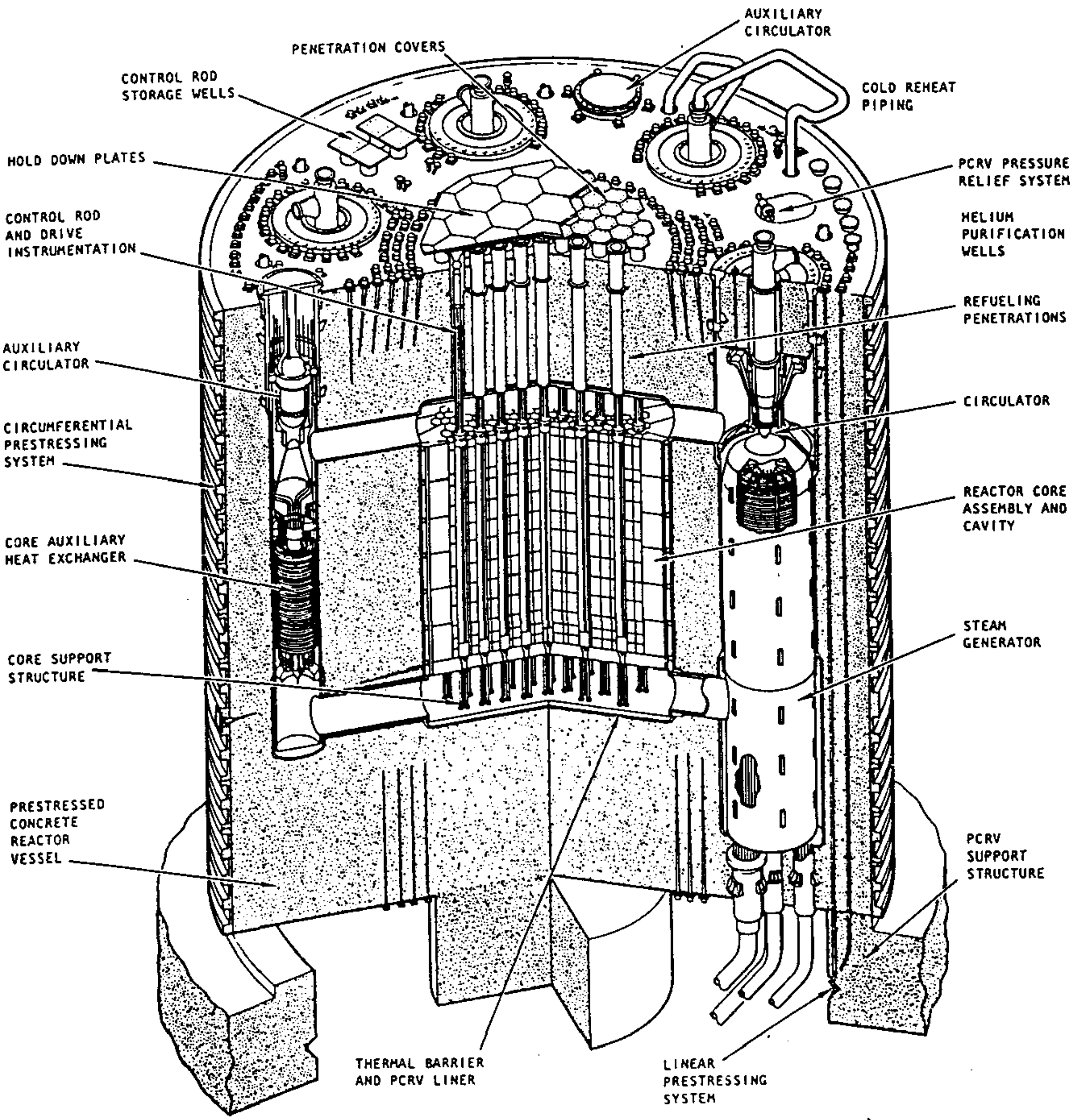


Figure 8-2. HTGR PRESTRESSED CONCRETE REACTOR VESSEL ARRANGEMENT. The primary system components are contained in a large cylinder of prestressed concrete. Penetrations exist for refueling, as well as for servicing (and even replacing) various pieces of equipment. Several primary coolant loops, as well as secondary cooling loops, are contained in the vessel. (Figure courtesy of General Atomic Co.)

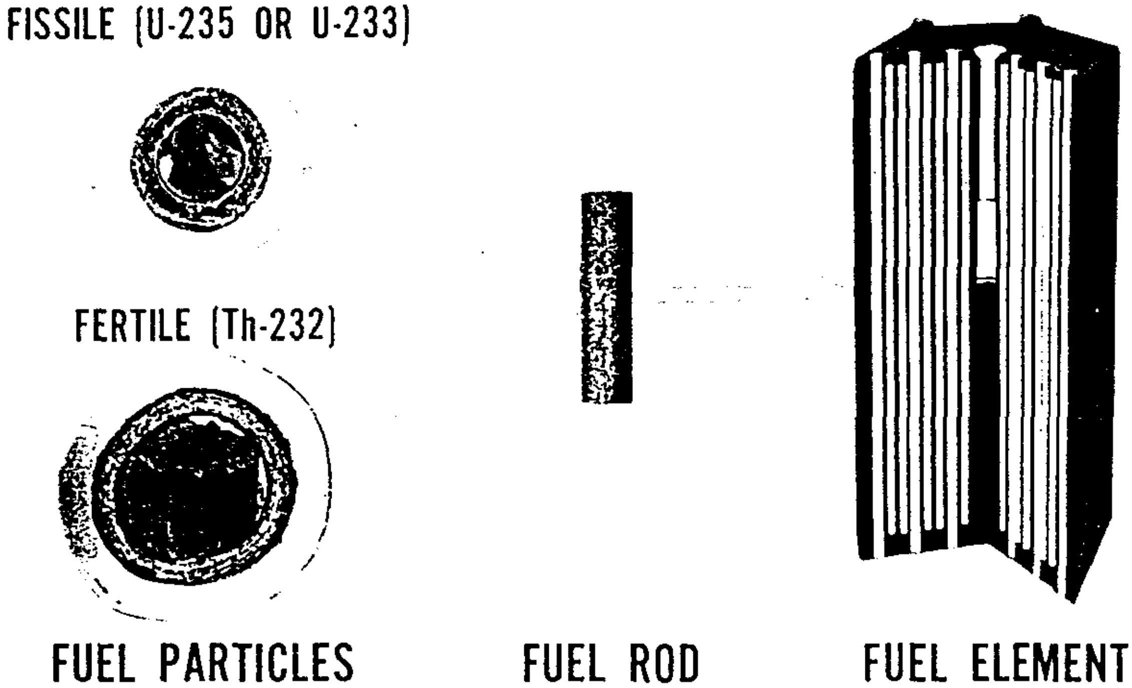
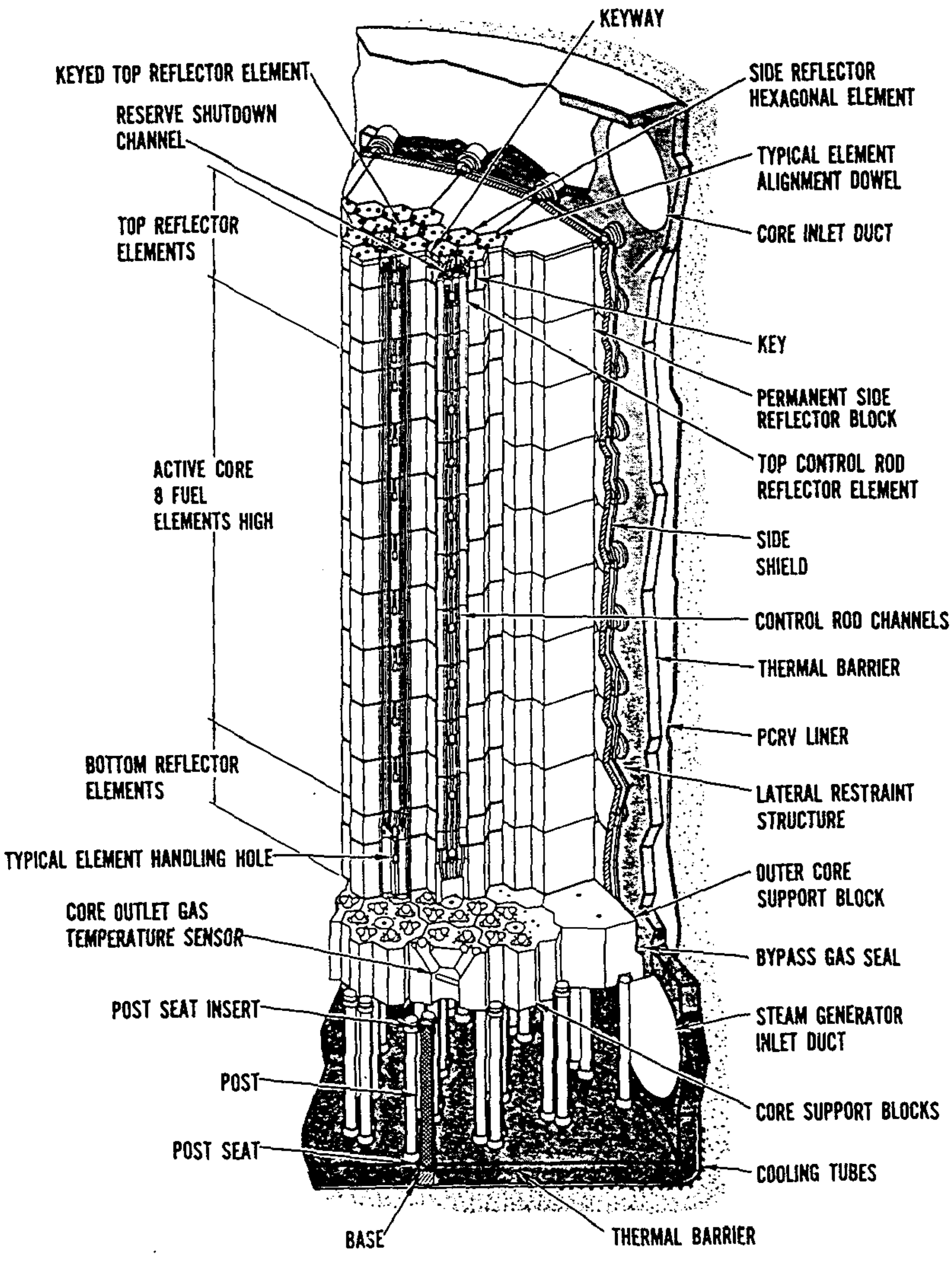


Figure 8-3. HTGR COATED FUEL PARTICLES, ROD, AND ELEMENT.
The HTGR uses two particle types: fissile material is coated with layers of carbon and silicon carbide, fertile material only with carbon. The particles are incorporated into a carbon binder to form a fuel rod, and these are put into fuel elements. (Figure courtesy of General Atomic Co.)

pellets (^{232}Th) are coated with only the carbon. As the reactor runs, fissile ^{233}U builds up in the latter particles. The silicon carbide, because it does not burn, aids in separating the two particle types at reprocessing, where the carbon is burned away.

The fuel particles are incorporated into fuel rods, with graphite as the binder, which are incorporated into the basic block or element (Figure 8-3). These elements are stacked as indicated in Figure 8-4. A basic refueling region consists of a central stack, which has two vertical control rod penetrations, and the adjacent six stacks, without such channels. The PCRV penetration above the central stacks (see Figure 8-2) serves both for refueling and, during operation, for the control drive mechanism. The central stacks also have an additional channel into which boron carbide balls can be poured as a reserve shutdown system. All the fuel elements have holes through which the coolant flows.

The core and other components of the nuclear steam supply system are contained in various cavities of the PCRV (Figure 8-2). Each of the cavities is steel lined to provide a seal and protect the concrete vessel. For detail of the core cavity


KEYWAY
KEYED TOP REFLECTOR ELEMENT
RESERVE SHUTDOWN CHANNEL
SIDE REFLECTOR HEXAGONAL ELEMENT
TYPICAL ELEMENT ALIGNMENT DOWEL
TOP REFLECTOR ELEMENTS
CORE INLET DUCT
KEY
PERMANENT SIDE REFLECTOR BLOCK
ACTIVE CORE 8 FUEL ELEMENTS HIGH
TOP CONTROL ROD REFLECTOR ELEMENT
SIDE SHIELD
CONTROL ROD CHANNELS
THERMAL BARRIER
PCRV LINER
BOTTOM REFLECTOR ELEMENTS
LATERAL RESTRAINT STRUCTURE
TYPICAL ELEMENT HANDLING HOLE
OUTER CORE SUPPORT BLOCK
CORE OUTLET GAS TEMPERATURE SENSOR
BYPASS GAS SEAL
POST SEAT INSERT
STEAM GENERATOR INLET DUCT
POST
CORE SUPPORT BLOCKS
POST SEAT
COOLING TUBES
BASE
THERMAL BARRIER

lining, see Figure 8-4. The PCRV has penetrations for refueling and control, as noted above, and for piping. In addition, there are removal plugs for servicing, and even replacement, of steam generators, helium circulators, etc. (It has been found, however, at the 330 MWe Fort St. Vrain (Colorado) HTGR, that imperfections in the core lining are difficult to repair.) The vessel is prestressed with vertical steel tendons and wrapped with circumferential cables. The PCRV and its contents are extremely massive, about 100 million pounds (45,000 Mg); and indeed the core itself is more massive, by about an order of magnitude, than the core of an LWR. See Table 8-1 for HTGR parameters.

The primary coolant system consists of the core and four to six primary coolant loops, each with its own circulator and steam generator. Helium gas, at a pressure of 700 psi (5 MPa), is pumped downward through the core and exits with a temperature of about 1370 °F (743 °C), considerably higher than for water-cooled reactors. The gas then passes into one of the pipes leading to a steam generator, where steam is raised for driving the turbogenerators. Above each steam generator is mounted a circulator which pumps the helium into the core.

The high reactor operating temperature is permitted by the gaseous form of the coolant and the good high-temperature characteristics of the core (there is no metal cladding that is sensitive to high temperature). This high temperature yields steam that can be converted to electrical energy with an efficiency of 39%, unusually high among thermal reactors. Moreover, the potential arises, with helium-driven turbogenerators, to improve even this high efficiency.

AUXILIARY SYSTEMS

The most noticeable auxiliary systems, shown in Figure 8-2, are the two or three auxiliary cooling loops. They are also contained in the PCRV and, in the event of failure of the main loops, can serve to remove the decay heat after reactor shutdown. However, the main cooling system is ordinarily the primary residual heat removal system following any shutdown.

Two identical systems are available for purifying the helium coolant. Each system uses filtration, adsorption, and a hydrogen getter to remove particulates and contaminant gases. One system operates while the other is shut down for decay and regeneration. The radioactive waste gas system is devoted largely to processing of gases released during regeneration of the purification system. These gases are separated into a radioactive component, which is ordinarily returned to the PCRV, and a stable component, which is released to the atmosphere. Liquid wastes arise only from decontamination operations, and the principal solid wastes are the tritium contaminated getters from the helium purification systems.

Figure 8-4. HTGR FUEL ELEMENT ARRANGEMENT.
HTGR fuel elements are arranged into stacks, which themselves are arranged in groups of seven; the central stack of each group has control rod channels. Note that the prestressed concrete vessel is lined with steel and protected with a thermal barrier. In addition, neutron reflector blocks surround the active core. (Figure courtesy of General Atomic Co.)

Representative Characteristics of High-Temperature Gas-Cooled Reactors

Core thermal power	2,900 MWth
Plant efficiency	39%
Plant electrical output	1,160 MWe
Core diameter	27.8 ft (8.5 m)
Core active height	20.8 ft (6.3 m)
Core power density	8.4 kW/liter
Number of core stacks (columns)	493
Number of fuel elements per column	8
Number of fuel elements	3,944
Element geometry	Hexagonal shape, 31 in high, 14 in across flats
Control rod type	Pairs of control rods in central stack of each refueling region (set of seven stacks)
Number of control rods	73 pairs
Reserve shutdown system	Spheres of boron carbide in carbon
Form of fuel	Fissile and fertile materials in different fuel particles, ^{235}U as UC_2, thorium+ bred ^{233}U in other particle type. Types have different coatings to facilitate separation.
Maximum fuel temperature	2,750 °F (1510 °C)
Average fuel temperature	1,450 °F (788 °C)
Average moderator temperature	1,320 °F (716 °C)
Coolant	Helium gas
Coolant flow rate	10.4×10^6 lb/hr (1.3 Mg/s)
Coolant pressure	700 psi (4.8 MPa)
Coolant temperature (inlet)	636 °F (336 °C)
Coolant temperature (outlet)	1,366 °F (741 °C)
Fuel exposure	98,000 MWd/Te
Fresh fuel assay (fissile particles)	93% ^{235}U (in initial loading)
Spent fuel assay (fissile particles)	30% ^{235}U (from initial loading)
Refueling sequence	One-fourth of the fuel per year
Weight of core and innards	6×10^6 lb (3×10^3 Mg)
Weight of PCRV (empty)	90×10^6 lb (4×10^4 Mg)

Source: General Atomic Co. specifications.

A steam generator isolation system is designed to prevent leakage of water or steam into the primary coolant. If the presence of water is detected, the defective cooling loop is isolated while the reactor is shut down, and the remaining loops provide cooling.

SAFETY SYSTEMS

The safety requirements of an HTGR are substantially different from those of water-cooled reactors. For one thing, the core provides a massive enough heat sink to lengthen by a large factor the time required for damage to occur to the fuel. Whereas decay heat can melt light-water fuel cladding within a minute or two of loss of cooling, HTGR fuel particles, with their ceramic coatings, can survive for as much as an hour. Moreover, the core's structural strength is provided by graphite,

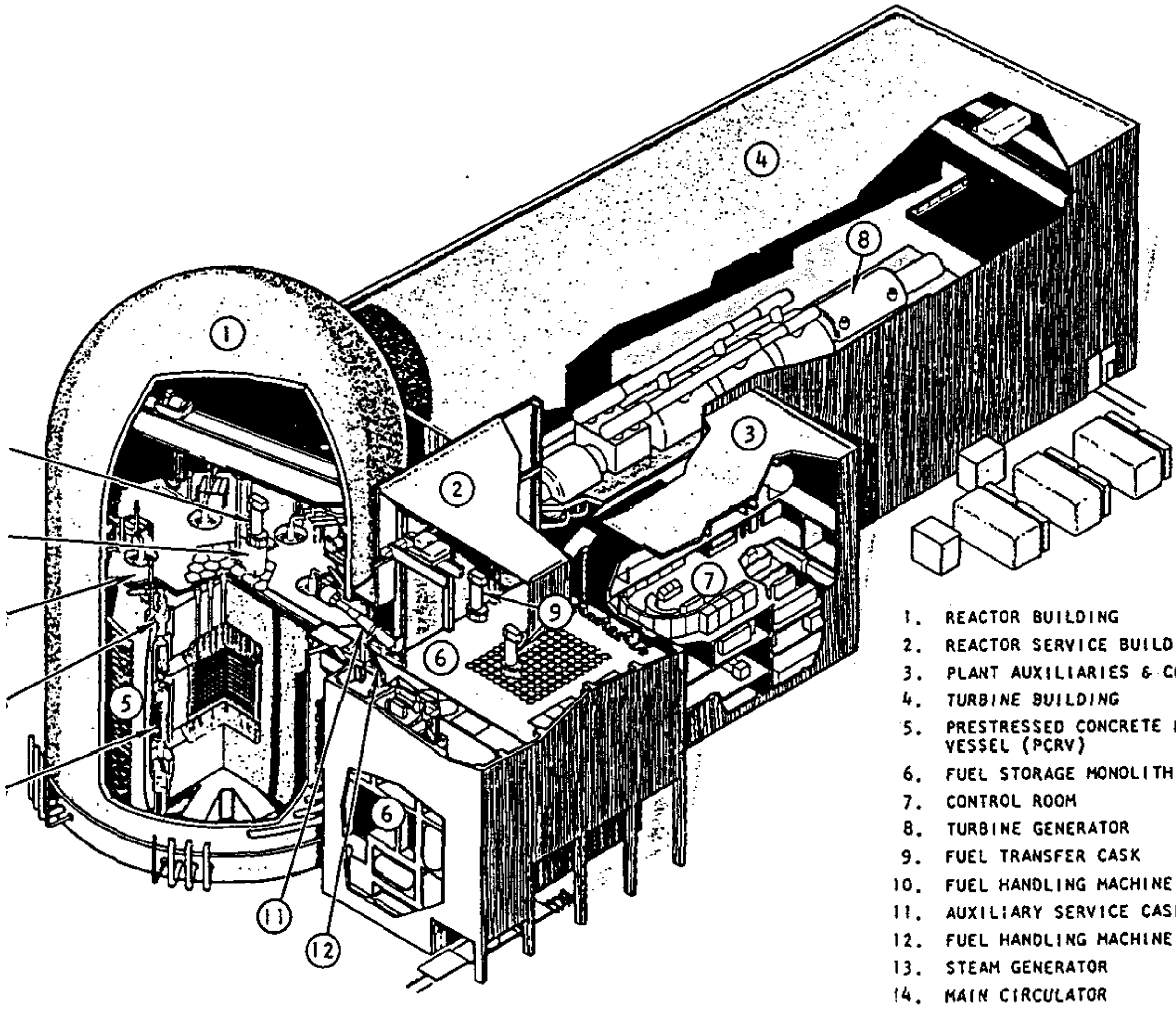


Figure 8-5. HTGR PLANT ARRANGEMENT.
An HTGR plant has a reactor containment building, as well as buildings for fuel storage and for the turbogenerator system. (Figure courtesy of General Atomic Co.)

whose strength increases as the temperature rises. On the other hand, the helium coolant does not provide much cooling capacity unless it continues to be pumped at high pressure. To make a complete loss of coolant extremely improbable, flow restrictors are incorporated around PCRV penetrations to reduce helium loss should the vessel integrity be violated there. As a result, helium is always presumed to be in the system. Should all the primary cooling loops become unavailable (this is unlikely since they are largely independent), the auxiliary cooling loops can be activated and are sized to handle the decay heat. It is also worth noting that since the coolant, helium, can be only in one phase and is nonreactive, certain complications that may arise during accidents involving water coolant are eliminated.

HTGRs include a secondary containment structure (see Figure 8-5) as in other reactor plant types. The containment isolation and radioactive cleanup systems are similar to those of PWRs. In principle, though, it might be possible for less expensive containment systems to be used, considering the integrally contained nature of the primary coolant system in an HTGR.

NEUTRONICS, FUEL UTILIZATION, AND REACTOR OPERATION

Gas-cooled, carbon-moderated reactors have basic physics characteristics that are substantially different from those of water-cooled reactors. Use of carbon as a moderator implies that fission neutrons have to travel a much larger distance to reach thermal energies. The fuel distribution of an HTGR implies that the neutron energy distribution within the fuel pellets is not greatly different than it is in the moderator. As a result, HTGR fuel is subjected to more neutrons of intermediate energy than LWR fuel, and this can lead to a greater absorption of neutrons by fertile material, in this case ^{232}Th. This fact can be used to design a reactor with a relatively high conversion ratio.

However, the HTGRs offered commercially in the early 1970s generally had conversion ratios slightly less than 0.7, only slightly higher than that of LWRs. But the uranium utilization was also improved because the HTGR has a higher thermal efficiency (39%) than LWRs (33%). These factors led to a lifetime uranium requirement, assuming uranium recycle, of about 3000 tons of U_3O_8 as compared with more than 4000 tons for LWRs (with recycle). Gas-cooled reactors, including the HTGR, can be designed with significantly higher conversion ratios, as discussed in Chapter 10 (see especially Table 10-1) and Chapter 14 (Table 14-2).

The basic HTGR is designed with an average fuel burnup of 96,000 MWd/Te, about three times that of an LWR. This assumes replacement of a quarter of the fuel annually. Even though the thermal efficiency of an HTGR is high, the fuel burnup is higher than that of an LWR because the annual loading of fuel (both fertile and fissile) for an HTGR is about one-third the weight of that for an LWR. An HTGR designed for a higher conversion ratio typically includes a larger mass of thorium, and irradiates the fuel to a lower burnup (see Table 10-1).

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