3

Boiling Water Reactors

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3.1 Introduction

3.1.1 Boiling Water Reactor Background

The boiling water reactors (BWRs) nuclear plant, like the pressurized water reactor (PWR), has its origins in the technology developed in the 1950s for the United States Navy nuclear submarine program. The first BWR nuclear plant to be built was the 5 MWe Vallecitos Plant (1957) near San Jose, California. The Vallecitos plant confirmed that BWR plants could successfully and safely produce electricity for a grid. The first large-scale BWR, Dresden I, followed in 1960, and since then the BWR design subsequently underwent a series of evolutionary changes with one purpose in mind: simplicity.

The major difference between the PWR and BWR is that the latter is a direct cycle nuclear system with heat generation occurring in the fuel region and water boiling in the envelope of the fuel bundles. This will be explored later.

There are approximately 92 operational BWRs in the world today and several advanced BWRs (ABWRs) currently under construction. This design comprises about 25% of the total number of units in operation globally. Current and former vendors are ASEA-Atom, Kraftwerken Union, Hitachi, Toshiba, and General Electric. Consolidation of the industrial supply base has led to continued partnerships in the nuclear supply chain.

The BWR design has been simplified in two key areas: reactor systems and containment design. Refer to Table 3.1 to see the evolution of simplification. The first BWR, Dresden I was, interestingly enough, not a true BWR. The design was based upon dual steam cycle, not the direct steam cycle that characterizes BWRs. Steam was generated in the reactor but then flowed to an elevated steam drum and a secondary steam generator before making its way to the turbine. The first step down the path of simplicity that led ultimately to the ABWR was elimination of the external steam drum by introducing two technical innovations: the internal steam separator and dryer.

General Electric selected the BWR as the most promising nuclear power concept because of its inherent advantages in control and design simplicity and established an atomic power equipment business in 1955 to offer it commercially. Aside from its heat source, the BWR generation cycle is substantially similar to that found in fossil-fueled power plants.

3.1.2 BWR-6 Product Line

The BWR-6 product line is capable of producing 20% more power from the same size pressure vessels as used in the BWR-5 product line without increasing the size of the respective buildings or support systems. Power output capabilities range from approximately 600–1400 MWe gross. Principal design features include

- Compact jet pumps with increased coolant circulation capability.
- Increased capacity from steam separators and dryers.
- More fuel bundles in standard pressure vessels and improvements in reactor internals arrangement.
Nuclear Engineering Handbook

3.1.3 ABWR

Development of the ABWR took place during the 1980s under the sponsorship of the Tokyo Electric Power Company (TEPCO) (Figures 3.1 and 3.2). The stated purpose of the development effort was to design a BWR plant that included a careful blend of (1) the best features of worldwide operating BWRs, (2) available new technologies, and (3) new modular construction techniques. Safety improvements were, as always, the top priority. Anticipating the economic challenges that lay ahead, special attention was paid to systematically reduce the capital cost and incorporate features into the plant design that would make maintenance significantly easier and more efficient.

Development of the ABWR occurred in a series of steps. Phase 1 was a conceptual design study that determined the feasibility of the ABWR concept. Phase 2, in which most of the development took place, included more detailed engineering and the testing of new technologies and design features. The purpose of Phase 3 was to put the finishing touches on the design and systematically reduce capital costs, which proved to be a highly

<table>
<thead>
<tr>
<th>Product Line</th>
<th>First Commercial Operation Date</th>
<th>Representative Plant/Characteristics</th>
</tr>
</thead>
<tbody>
<tr>
<td>BWR/1</td>
<td>1960</td>
<td>Dresden 1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Initial commercial-size BWR</td>
</tr>
<tr>
<td>BWR/2</td>
<td>1969</td>
<td>Oyster Creek</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Plants purchased solely on economics</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Large direct cycle</td>
</tr>
<tr>
<td>BWR/3</td>
<td>1971</td>
<td>Dresden 2</td>
</tr>
<tr>
<td></td>
<td></td>
<td>First jet pump application</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Improved ECCS: spray and flood capability</td>
</tr>
<tr>
<td>BWR/4</td>
<td>1972</td>
<td>Vermont Yankee</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Increased power density (20%)</td>
</tr>
<tr>
<td>BWR/5</td>
<td>1977</td>
<td>Tokai 2</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Improved ECCS</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Valve flow control</td>
</tr>
<tr>
<td>BWR/6</td>
<td>1978</td>
<td>Confrontes</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Compact control room</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Solid-state nuclear system protection system</td>
</tr>
<tr>
<td>ABWR</td>
<td>1996</td>
<td>Kashiwazaki-Kariwa 6</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Reactor internal pumps</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Fine-motion control rod drives</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Advanced control room, digital and fiber optic technology</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Improved ECCS: high/low pressure flooders</td>
</tr>
<tr>
<td>ESBWR</td>
<td>Under review</td>
<td>TBD</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Natural circulation</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Passive ECCS</td>
</tr>
</tbody>
</table>

- Smaller-diameter fuel rods, longer in active fuel length and arranged in $8 \times 8$ bundles within the same external outline as the previous $7 \times 7$ design. This lowers the kilowatt rating per length of fuel and permits increased heat output per bundle.
- Improved control and instrumentation systems incorporating the latest solid-state electronics technology.
- Improved operator–machine interface systems for better control of the plant.
FIGURE 3.1
ABWR major systems.
successful and, in hindsight, fortuitous endeavor. The development phases came to an end in 1988 when TEPCO announced that the next Kashiwazaki-Kariwa units to be constructed would be ABWRs.

With the selection of the ABWR for the K-6&7 projects, the detailed project engineering began. Licensing activities with the Japanese regulatory agency, the Ministry of International Trade and Industry (MITI), also started at this time and, interestingly, were conducted in parallel for some time with the review of the ABWR in the United States by the Nuclear Regulatory Commission (NRC). MITI and the NRC, in fact, held several meetings to discuss their respective reviews.

By 1991, the detailed design was essentially complete and MITI concluded its licensing review. An Establishment Permit, or license, was issued in May 1991. Excavation began later that year on September 17, bringing a decade of development work to a successful conclusion.

The key design objectives for the ABWR were established during the development program. The key goals, all of which were achieved, are as follows:

- Design life of 60 years
- Plant availability factor of 87%
- Less than one unplanned scram per year

**FIGURE 3.2**
ABWR reactor assembly.
• Refueling interval of 18–24 months
• Operating personnel radiation exposure limit <1 Sv/year
• Reduced calculated core damage frequency by at least a factor of 10 over previous BWRs (goal <10⁻⁶/year)
• Radwaste generation <100 m³/year
• Construction schedule of 48 months

3.1.4 Economic Simplified BWR

The economic simplified BWR (ESBWR) builds on the very successful ABWR technology and construction programs, as well as the simplified BWR (SBWR) development program (Figures 3.3 and 3.4). As of this publication production the ESBWR is being certified by the U.S. NRC. The key goals are

• Cost advantage over competing base load typical generating technologies
• Plant availability factor of 95%
• Design life of 60 years
• Less than one unplanned scram per year
• Refueling interval of 18–24 months
• Operating personnel radiation exposure limit <1 Sv/year
• Reduced calculated core damage frequency by at least a factor of 10 over previous BWRs (Goal <10⁻⁶/year)
• Radwaste generation less than that of the 10% best operating BWRs
• Construction schedule of 48 months
• 20% reduction in capital cost ($/kWh) versus previous 1100 MWe class BWRs typically complex safety systems

Table 3.2 is a comparison table for the key features of the described product lines.

3.1.5 Summary Description

The direct cycle BWR system (Figure 3.5) is a steam generation and steam utilization system consisting of a nuclear core located inside a reactor vessel and a conventional turbine generator and feedwater supply system. Associated with the nuclear core are auxiliary systems to accommodate the operational and safeguard requirements and necessary controls and instrumentation. Water is circulated through the reactor core, producing saturated steam, which is separated from recirculation water, dried in the top of the vessel, and directed to the steam turbine generator. The turbine employs a conventional regenerative cycle with condenser deaeration and condensate demineralization. The basic heat balance for a BWR system is shown in Figure 3.6.

The steam produced by the nuclear core is, of course, radioactive. The radioactivity is primarily N16, a very short-lived isotope (half-life of 7 s) so that the radioactivity of the steam exists from the reactor vessel only during power generation. Carryover of long-lived radioactive particles by the steam supply to the turbine and condensate system is virtually nonexistent.
FIGURE 3.3
ESBWR major systems.
Boiling Water Reactors

The nuclear core, the source of the heat, consists of fuel assemblies and control rods contained within the reactor vessel and cooled by the recirculating water system. A 1220-MWe BWR-6 core consists of 748 fuel assemblies and 177 control rod assemblies, forming a core array about 16 ft (4.9 m) in diameter and 14 ft (4.3 m) high. The power level is maintained or adjusted by positioning control rods up and down within the core. The BWR core power level is further adjustable by changing the recirculation flow rate through the core without changing control rod position. This unique BWR feature helps achieve the superior load-following capability of the BWR.

The BWR is the only light water reactor system that employs bottom-entry control rods. Bottom-entry and bottom-mounted control rod drives allow refueling without removal of control rods and drives and allow drive testing with an open vessel prior to initial fuel loading or at each refueling operation. The hydraulic control rod drive system, which incorporates

FIGURE 3.4
ESBWR reactor pressure vessel and internals.


### TABLE 3.2
Comparison of Key ESBWR Features to Previous BWRs

<table>
<thead>
<tr>
<th>Feature</th>
<th>BWR/6</th>
<th>ABWR</th>
<th>ESBWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Recirculation system inside RPV</td>
<td>Two external loop recirculation system with jet pumps</td>
<td>Vessel-mounted reactor internal pumps</td>
<td>Natural circulation</td>
</tr>
<tr>
<td>Control rod drives</td>
<td>Locking piston CRDs</td>
<td>Fine-motion CRDs</td>
<td>Fine-motion CRDs</td>
</tr>
<tr>
<td>ECCS</td>
<td>Two-division ECCS plus HPCS</td>
<td>Three-division ECCS</td>
<td>Four-division, passive, gravity-driven</td>
</tr>
<tr>
<td>Reactor vessel</td>
<td>Welded plate</td>
<td>Extensive use of forged rings</td>
<td>Extensive use of forged rings</td>
</tr>
<tr>
<td>Primary containment water</td>
<td>Mark III—large, low pressure, not inerted</td>
<td>Compact, inerted</td>
<td>Compact, inerted</td>
</tr>
<tr>
<td>Isolation makeup water</td>
<td>RCIC</td>
<td>RCIC</td>
<td>Isolation condensers, passive</td>
</tr>
<tr>
<td>Shutdown heat removal</td>
<td>Two-division RHR</td>
<td>Three-division RHR</td>
<td>Nonsafety system combined with RWCU</td>
</tr>
<tr>
<td>Containment heat removal</td>
<td>Two-division RHR</td>
<td>Three-division RHR</td>
<td>Passive</td>
</tr>
<tr>
<td>Emergency AC</td>
<td>Three safety-grade D/G</td>
<td>Three safety-grade D/G</td>
<td>Two nonsafety D/G</td>
</tr>
<tr>
<td>Alternate shutdown</td>
<td>Two SLC pumps</td>
<td>Two SLC pumps</td>
<td>Two SLE accumulators</td>
</tr>
<tr>
<td>Control and instrumentation</td>
<td>Analog, hardwired, single channel</td>
<td>Digital, multiplexed, fiber optics, multiple channel</td>
<td>Digital, multiplexed, fiber optics, multiple channel</td>
</tr>
<tr>
<td>In-core monitor calibration</td>
<td>TIP system</td>
<td>A-TIP system</td>
<td>Gamma thermometers</td>
</tr>
<tr>
<td>Control room</td>
<td>System-based</td>
<td>Operator task-based</td>
<td>Operator task-based</td>
</tr>
<tr>
<td>Severe accident mitigation</td>
<td>Not specifically addressed</td>
<td>Inerting, drywell flooding, containment venting</td>
<td>Inerting, drywell flooding, core catcher</td>
</tr>
</tbody>
</table>

![FIGURE 3.5](https://example.com/fig3_5.png)

Direct cycle reactor system.
mechanical locking of the rod at the selected position, provides positive driving and positioning of the control rods. Pressurized accumulators that provide a rod insertion force far greater than any gravity or mechanical system carry out rapid control rod insertion.

The core flow of a BWR is the sum of the feedwater flow and the recirculation flow (typical of any boiler). An important and unique feature of the BWR product line is the application of jet pumps inside the reactor vessel. These pumps generate about two-thirds of the recirculation flow within the reactor vessel. The jet pumps also contribute to the inherent safety of the BWR design under loss-of-coolant emergency conditions. Like most boilers, the BWR can deliver at least 10% power in a natural recirculation mode without operation of the recirculation pumps.

The BWR operates at constant pressure and maintains constant steam pressure similar to most fossil boilers. The integration of the turbine pressure regulator and control system in conjunction with the reactor water recirculation flow control system permits automated changes in steam flow to accommodate varying load demands on the turbine. Power changes of up to 25% of rated power can be accomplished automatically by recirculation

![FIGURE 3.6](image-url)
flow control alone, thus providing automatic load-following capability for the BWR without altering control rod settings.

The nuclear boiler system is supported by the specialized functions of its auxiliary system. Several auxiliary systems are used for normal plant operation:

- Reactor water cleanup (RWCU) system
- Shutdown cooling function of the residual heat removal (RHR) system
- Fuel building and containment pools cooling and filtering system
- Closed cooling water system for reactor service
- Radioactive waste treatment system

The following auxiliary systems are used as backup (standby) or emergency systems:

- Standby liquid control (SBLC) system
- Reactor core isolation cooling (RCIC) system
- RHR system
- Low-pressure coolant injection (LPCI)
- Steam condensing
- Containment spray
- Suppression pool cooling
- High-pressure core spray (HPCS) system
- Low-pressure core spray (LPCS) system
- Automatic depressurization

3.2 Nuclear Boiler Assembly

3.2.1 Introduction

The nuclear boiler assembly consists of the equipment and instrumentation necessary to produce, contain, and control the steam power required by the turbine generator. The principal components of the nuclear boiler are as follows:

- **Reactor vessel and internals**: Reactor pressure vessel, jet pumps for reactor water recirculation, steam separators and dryers, core spray, and feedwater spargers and core support structure
- **Reactor water recirculation system**: Pumps; control and equipment isolation valves; piping and its suspension devices, restraints, and suppressors; used in providing and controlling core flow
- **Main steam lines**: Safety/relief and containment isolation valves; piping up to and including outboard containment isolation valve, and its restraints, suppressors, and guides
- **Control rod drive system**: Control rods, control rod drive mechanisms, and hydraulic system for insertion and withdrawal of the control rods
3.2.2 Reactor Assembly

The reactor assembly (Figure 3.7) consists of the reactor vessel, its internal components of the core, the shroud, the top guide assembly, the core plate assembly, the steam separator and dryer assemblies, and the jet pumps. The reactor assembly also includes the control rods, control rod drive housings, and control rod drives. Each fuel assembly that makes up the core rests on an orificed fuel support mounted on top of the control rod guide tubes. Each guide tube, with its fuel support piece, bears the weight of four assemblies and is supported by a control rod drive penetration nozzle in the bottom head of the reactor vessel. The core plate provides lateral guidance at the top of each control rod guide tube. The top guide provides lateral support for the top of each fuel assembly.

Control rods occupy alternate spaces between fuel assemblies and may be withdrawn into the guide tubes below the core during plant operation. The rods are coupled to control rod drives mounted within housings, which are welded to the bottom head of the reactor vessel. The bottom-entry drives do not interfere with refueling operations. A flanged joint is used at the bottom of each housing for ease of removal and maintenance of the rod drive assembly.

Except for the Zircaloy in the reactor core, these reactor internals are stainless steel or other corrosion-resistant alloys. All major internal components of the reactor can be removed
except the jet pump diffusers, the core shroud, the jet pump, and high-pressure coolant injection inlet piping. The removal of the top guide assembly and the core plate assembly is a major task, and it is not expected that these components would require removal during the life of the plant. The removal of other components such as fuel assemblies, in-core assemblies, control rods, and fuel support pieces is performed on a routine basis.

### 3.2.2.1 Reactor Vessel

The reactor vessel is a pressure vessel with a single full-diameter removable head. The base material of the vessel is low alloy steel, which is clad on the interior except for nozzles with stainless steel weld overlay to provide the necessary resistance to corrosion. Since the vessel head is exposed to a saturated steam environment throughout its operating lifetime, stainless steel cladding is not used over its interior surfaces.

Fine-grained steels and advanced fabrication techniques are selected to maximize structural integrity of the vessel. BWR vessels have the lowest neutron exposure of any light water reactor, and the annulus space that carries recalculting water and feedwater downward between the core shroud and the vessel reduces radiation experienced by the vessel wall material. Vessel material surveillance samples are located within the vessel to enable periodic monitoring of exposure and material properties. Provisions are made for irradiating tensile and impact specimens for a program of monitoring and evaluating radiation-induced changes in vessel. Such programs have been conducted in most General Electric-designed power reactors, and considerable data have been accumulated on the performance of vessel materials after irradiation. The initial selection of high-quality materials, coupled with a continuing evaluation program, permits the vessel to meet the requirements of operability and safety throughout its design lifetime.

The vessel head closure seal consists of two concentric metal O-rings. This seal system has been demonstrated to perform without detectable leakage at all operating conditions. These conditions include cold hydrostatic testing, heating and cooling, and power operation. To monitor seal integrity, a leak detection system is used. Vessel supports, internal supports, their attachments, and adjacent shell sections are designed to take combined loads, including control rod drive reactions, earthquake loads, and jet reaction thrusts. The vessel is mounted on a supporting skirt, which is bolted to a concrete and steel cylindrical vessel pedestal, which is integrated with the reactor building foundation.

Many features have been incorporated in the design of the vessel and its associated piping to simplify the refueling operation. Steam outlet lines are welded to the vessel body, thereby eliminating the need to break flanged joints in the steam lines when removing the head for refueling. Another design feature is the seal between the vessel and the surrounding drywell, which permits flooding of the space (reactor well) above the vessel.

### 3.2.2.2 Core Shroud

The shroud is a cylindrical, stainless steel structure that surrounds the core and provides a barrier to separate the upward flow through the core from the downward flow to the annulus. The discharge plenum of the core shroud is formed by the following connections; a flange at the top of the shroud mates with a flange on the top guide, which in turn mates with a flange on the steam separator assembly. The jet pump discharge diffusers penetrate the peripheral shelf of the shroud support below the core elevation to introduce the coolant into the inlet plenum. The peripheral shelf of the shroud support is welded to the vessel wall to prevent the jet pump outlet flow from bypassing the core and to form a chamber.
around the core, which can be re-flooded in the event of a loss-of-coolant accident (LOCA). The shroud support carries the weight of the shroud, the steam separators, the jet pump system, and the seismic and pressure loads in normal and fault conditions of operation.

Two ring spargers, one for LPCS and the other for HPCS, are mounted inside the core shroud in the space between the top of the core and the steam separator base. The core spray ring spargers are provided with spray nozzles for the injection of cooling water. The core spray spargers and nozzles do not interfere with the installation or removal of fuel from the core. A nozzle for the injection of the neutron absorber (sodium pentaborate) solution is mounted below the core in the region of the recirculation inlet plenum.

The steam separator assembly consists of a domed base on top of which is welded an array of standpipes with a three-stage steam separator located at the top of each standpipe. The steam separator assembly rests on the top flange of the core shroud and forms the cover of the core discharge plenum region. The seal between the separator assembly and core shroud flanges is a metal-to-metal contact and does not require a gasket or other replacement sealing devices. The separator assembly is bolted to the core shroud flange by a long hold-down bolt for ease of removal and extension above the separators. During installation, the separator base is aligned on the core shroud flange with guide rods and finally positioned with locating pins. The objective of the long-bolt design is to provide direct access to the bolts during reactor refueling operations with minimum-depth underwater tool manipulation during the removal and installation of the assemblies, which makes it unnecessary to engage threads in making up the shroud head. A tee-bolt engages in the top guide and its nut is tightened to only nominal torque. Final loading is established through differential expansion of the bolt and compression sleeve. The fixed axial flow-type steam separators have no moving parts and are made of stainless steel. In each separator, the steam–water mixture rising through the standpipe impinges on vanes giving the mixture a spin to establish a vortex wherein the centrifugal forces separate the water from the steam in each of the three stages. The steam then leaves the separator at the top and passes into the wet steam plenum below the dryer. The separated water exits the lower end of each stage of the separator and enters the pool that surrounds the standpipes to join the down comer annulus flow.

### 3.2.2.3 Steam Dryer

The steam dryer assembly is mounted in the reactor vessel above the steam separator assembly and forms the top and sides of the wet steam plenum. Vertical guides on the inside of the vessel provide alignment for the dryer assembly during installation. Pads extending inward from the vessel wall support the dryer assembly, and it is held down in position during operation by the vessel head. Steam from the separators flows upward and outward through the drying vanes. These vanes are attached to top and bottom supporting members forming a rigid integral unit. Moisture is removed and carried by a system of troughs and drains to the pool surrounding the separators and then into the recirculation down comer annulus.

### 3.2.3 Reactor Water Recirculation System

The function of the reactor water recirculation system (Figure 3.8) is to circulate the required coolant through the reactor core. The system consists of two loops external to the reactor vessel, each containing a pump with a directly coupled water-cooled (air–water) motor, a flow control valve, and two shutoff valves.
High-performance jet pumps located within the reactor vessel are used in the recirculation system. The jet pumps, which have no moving parts, provide a continuous internal circulation path for a major portion of the core coolant flow.

The recirculation pumps take suction from the downward flow in the annulus between the core shroud and the vessel wall. Approximately one-third of the core flow is taken from the vessel through the two recirculation nozzles. There, it is pumped at a higher pressure, distributed through a manifold to which a number of riser pipes are connected, and returned to the vessel inlet nozzles. This flow is discharged from the jet pump nozzle into the initial stage of the jet pump throat where, due to a momentum exchange process, it induces surrounding water in the down comer region to be drawn into the jet pump throat where these two flows mix and then diffuse in the diffuser, to be finally discharged into the lower core plenum. The jet pump diffusers are welded into openings in the core shroud support shelf, which forms a barrier between the lower plenum and the suction side of the jet pump. The flow of water turns upward, where it flows between the control rod drive guide tubes and enters into the fuel support where the flow is individually directed to each fuel bundle through the nose-piece. Orifices in each fuel support piece provide the desired flow distribution among the fuel assemblies. The coolant water passes along the individual fuel rods inside the fuel channel where it is heated and becomes a two-phase, steam–water mixture. The steam–water mixture enters a plenum located directly above the core and bounded by the separator dome, which opens to the separator array of fixed steam separators. The steam is separated from the water and passes through a dryer where any remaining water is removed. The saturated steam leaves the vessel through steam line nozzles located near the top of the vessel body end and is piped to the turbine. Water collected in the support tray of the dryer is routed through drain lines, joins the water leaving the separators, and flows downward in the annulus between the
core shroud and the vessel wall. Feedwater is added to the system through spargers located above the annulus and joins the downward flow of water. A portion of this downward flow enters the jet pumps and the remainder exits from the vessel as recirculation flow.

### 3.2.3.1 Jet Pump Assembly

The jet pumps (Figure 3.9) are located in the annular region between the core shroud and the vessel inner wall. Each pair of jet pumps is supplied driving flow from a single riser pipe. The jet pump assembly is composed of two jet pumps and contains no moving parts.
Each jet pump consists of an inlet mixer, a nozzle assembly with five discharge ports, and a diffuser. The inlet mixer assembly, a replaceable component, is a constant-diameter section of pipe with a diffuser entrance section at the lower end and the drive nozzle at the upper end. The nozzle assembly can be removed by disconnecting the removable split flange. The jet pump diffuser is a gradual conical section terminating in a straight cylindrical section at the lower end that is welded into the shroud support. The overall length of the jet pumps is approximately 19 ft (5.8 m). Instrumentation monitors jet pump flow passages to ascertain their individual and collective flow rates under varying operating conditions.

3.2.3.2 Operating Principle of the Jet Pump

The driving flow enters the nozzle section at a high pressure and is accelerated to a high velocity because of the constriction at the nozzle outlet. The suction flow enters at a low pressure, which is further reduced as the flow is accelerated through the converging suction inlet nozzle. These two streams merge in the mixing section, where a pressure rise occurs because of the velocity profile rearrangement and the momentum transfer caused by the mixing. The rate of pressure rise decreases near the end of the mixing section because mixing is essentially completed. A diffuser is located downstream from the mixing section to slow the relatively high velocity mixed streams. This converts the dynamic head into static head. The jet pump system readily accommodates the full spectrum of flow rates required for load following.

3.2.3.3 Safety Feature of the Jet Pump

The safety feature of post-accident core flooding capability with a jet pump design allows flooding at no less than two-thirds of the core height. There is no recirculation line break that can prevent re-flooding of the core to the level at the top of the jet pump.

3.2.3.4 Pumps and Motors

The reactor recirculation pumps are vertically mounted, centrifugal, mechanical seal type and are constructed of stainless steel. The pumps operate at 25% of rated speed during startup and are powered from a low-frequency motor generator set. Following startup, the pumps operate at constant speed and are powered from auxiliary power.

The pump shaft seal assembly consists of multiple mechanical seals built into a cartridge or cartridges, which can be readily replaced with spare cartridges without removing the motor from the pump. Each seal carries an equal portion of the total pressure differential and is capable of sealing against maximum pump operating pressure. A throttle bushing located in the pump casing minimizes leakage in the unlikely event of a gross failure of all shaft seals. Cooling coils that circulate water from the closed cooling water system for reactor service controls the temperature of the seal cavity. The temperatures of the cavity and cooling water for each pump are recorded, and on high temperature activate an alarm in the control room.

The drive motor for each pump is a vertical water-cooled (air–water heat exchanger), totally enclosed, three-phase, squirrel-cage induction motor designed to operate at constant speed. Cooling water to the air–water cooler for motor windings cooling and through coils in the bearing oil reservoir for motor bearing cooling is provided from the closed cooling water system. Temperature recorders and high-temperature alarms are located in the control room for motor windings, bearing oil reservoirs, and cooling water.
3.2.3.5 Valves and Piping

The recirculation loop piping is of welded construction. The piping, associated valves, and pumps are hung using constant-support hangers, thereby minimizing resultant stresses at the point of attachment to the reactor vessel. All recirculation piping is restrained to prevent pipe whipping as a result of jet action forces that may arise if a pipe break were to occur. The shutoff and bypass valves are motor-operated gate valves and the flow control valve is a ball-type with electrohydraulic actuator.

The flow control valve is on the discharge side of the pump. One shutoff valve is on the suction side of the pump and the other is downstream of the flow control valve. This allows maintenance in parallel with the refueling operation. No special reactor pressure vessel water level considerations are necessary. The stainless steel valves have double sets of valve stem packing to provide a highly reliable seal.

The ABWR design enhancements include elimination of the external recirculation loops and pumps and installation of reactor internal pumps (Figure 3.10). The ESBWR transitions

FIGURE 3.10
Cross section of reactor internal pump.
from a forced recirculation mode to a natural circulation mode and eliminates the need for reactor recirculation pumps.

3.2.4 Main Steam Lines

Steam exits from the vessel several feet below the reactor vessel flange through four nozzles. Carbon steel steam lines are welded to the vessel nozzles and run parallel to the vertical axis of the vessel, downward to the elevation where they emerge horizontally from the containment. Two air-operated isolation valves are installed on each steam line, one inboard and one outboard of the containment penetration. The safety/relief valves are flange-connected to the main steam line for ease of removal for test and maintenance. A flow-restricting nozzle is included in each steam line as an additional engineered safeguard to protect against a rapid uncovering of the core in case of a break of a main steam line.

3.2.4.1 Safety/Relief Valves

The safety/relief valves are dual-function valves discharging directly to the pressure suppression pool. The safety function provides protection against overpressure of the reactor primary system. The relief function provides power-actuated valve opening to depressurize the reactor primary system. The valves are sized to accommodate the most severe of the following two pressurization transient cases determined by analysis:

- Turbine trip from turbine design power, failure of direct scram on turbine stop valve closure, failure of the steam bypass system, and reactor scrams from an indirect scram
- Closure of all main steam line isolation valves, failure of direct scram based on valve position switches, and reactor scrams from an indirect scram

For the safety function, valves open at spring set point pressure and close when inlet pressure falls to 96% of spring set point pressure.

For the pressure-relief function, the valves are power-actuated manually from the control room or power-actuated automatically upon high pressure. Separate power circuits supply each valve. Valves that are power-actuated automatically upon high pressure close when pressure falls to a preset closing pressure. The pressure-relief function set point is below that for operation of valves for the safety (spring-actuated) function. By operating at the lower set point for the pressure-relief function, the re-closing set point of the valves provides a higher differential shutoff pressure than a spring reset valve, assuring leak tightness of the valves. The pressure-relief function may be used (in the event the main condenser is not available as a heat sink after reactor shutdown) to release steam generated by core decay heat until the RHR system steam condensing function is initiated.

To limit the cycling of relief valve to one valve subsequent to their initial actuation during a main steam line isolation event, two valves (one a backup to the other) have the feature of automatically changing normal set pressures (opening and closing) following their initial actuation at normal set pressures to a lower level, thereby limiting the pressure cycles to a level where the other relief valves will not reopen. In conjunction with these two valves, the set pressure for the closing of the other valves is changed automatically, which allows them to stay open longer before closing to accommodate pressure swings. Manual valve operation and resetting of valve set pressure to their normal levels after the transient is by the control room operator.
Selected safety/relief valves are associated with the automatic depressurization of the primary system under assumed LOCA conditions. These valves have two independent logic channels powered from different power sources, either of which can initiate depressurization. Valves open automatically and remain open until the pressure falls to a preset closure pressure. These valves open automatically upon signals of high drywell pressure and low reactor water level and confirmation of one LPCI function of the RHR system or LPCS system running. Initiation signals need not be simultaneous. The valves remain open until the primary system pressure is reduced to a point where the LPCI function of the RHR system and/or the LPCS system can adequately cool the core. The initiation of automatic depressurization is delayed from 90 to 120 s to allow the operator to terminate the initiation should the HPCS system initiation and acceptable reactor vessel level have been confirmed.

The valves used for automatic depressurization can be manually power actuated to open at any pressure. The signal for manual power actuation is from redundant control room switches from different power sources.

In the unlikely event that the RHR shutdown suction line is unavailable during reactor shutdown to cool reactor water and during the period when the LPCI function of the RHR system and/or the LPCS system pumps are injecting water into the reactor vessel, safety/relief valves used for automatic depressurization can be used to pass water from the reactor vessel to the suppression pool via valve discharge lines. For this to occur, the reactor vessel floods to a level above the vessel main steam line nozzles, selected safety/relief valves are opened from the control room to pass reactor water to the suppression pool.

Each steam line has two containment isolation valves, one inside and one outside the containment barrier. The isolation valves are spring-loaded pneumatic piston-operated globe valves designed to fail closed on loss of pneumatic pressure or loss of power to the pilot valves. Each valve has an air accumulator to assist in the closure of the valve upon loss of the air supply, electrical power to the pilot valves, and failure of the loaded spring. Each valve has an independent position switch initiating a signal into the reactor protection system scram trip circuit when the valve closes.

The isolation valves close upon (1) low water level in the reactor vessel, (2) high radiation from the steam line, (3) high flow rate in the main steam line, (4) low pressure at inlet to the turbine, (5) high ambient and differential steam line tunnel temperature (outside the containment), (6) low condenser vacuum (unless procedurally bypassed), and (7) high turbine building temperature. The signal for closure comes from two independent channels; each channel has two independent tripping sensors for each measured variable. Once isolation is initiated, valves continue to close and cannot be opened except by manual means. Independent remote manual switches located in the control room may each operate the valves. Lights in the control room indicate the position of the valve.

A shutoff valve is used in each steam line outboard of the external containment isolation valve and functions as a backup to the isolation valve. The shutoff valve is part of leakage control system to prevent possible release of nuclear steam which could leak through the main steam containment isolation valves following a LOCA. Independent containment inboard and containment outboard divisions are used to establish a pressurized barrier between the containment barrier and the environs. Out leakage is effectively eliminated and in leakage is directed into the containment from the pressurized volume. Both divisions are powered from auxiliary and standby AC power. While either of the two divisions is sufficient to establish the necessary pressure barrier, both are initiated in the control
room by a remote manual switch after it has been determined that a LOCA has occurred. The system will not actually initiate unless the pressure levels of the air supply and the reactor vessel are within the permissive interlock set points.

The main steam line isolation valves remain closed if the steam line pressure is greater than the air pressure interlock set point. When the interlock is cleared, air is admitted to raise the pressure in the main steam lines to a predetermined level to establish the containment pressure boundary.

### 3.2.5 Control Rod Drive System

Positive core reactivity control is maintained by the use of movable control rods interspersed throughout the core. These control rods thus control the overall reactor power level and provide the principal means of quickly and safely shutting down the reactor. The rods are moved vertically by hydraulically actuated, locking piston-type drive mechanisms. The drive mechanisms perform a positioning and latching function and a scram function with the latter overriding any other signal. The drive mechanisms are bottom-entry, upward-scramming drives which are mounted on a flanged housing on the reactor vessel bottom head. Here, they cause no interference during refueling and yet they are readily accessible for inspection and servicing. Hydraulic connections to the drive mechanism are made at ports in the face of the housing flange.

The control rod drive system consists of several locking-piston control rod drive mechanisms, a hydraulic control unit for each drive mechanism, a hydraulic power supply for the entire system, and instrumentation and controls with necessary interconnections. The locking-piston-type control rod drive mechanism is a double-acting hydraulic piston that uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion. The drive mechanism can position the rods at intermediate increments over the entire core length. The control rod can be uncoupled from below the vessel without removing the reactor vessel head, or with the vessel head removed for refueling, without removing the drive mechanism. Some of the advantages of the bottom-mounted drive arrangement are detailed below.

- The drives do not interfere with refueling and are operative even when the head is removed from the reactor. Furthermore, this location makes them more accessible for inspection and servicing. Such an arrangement makes maximum use of the water in the reactor as a neutron shield, while yielding the least possible neutron exposure to drive components.
- The locking piston drive provides the highest scram forces and operating force margins of all known types of drive mechanisms. This provides high operational reliability, particularly in the scram function.
- The use of water of reactor quality as the operating fluid eliminates the need for special hydraulic systems, with their inherent leakage and maintenance problems.
- The continuous inflow of high-purity water through the drives minimizes the contamination deposits within the drives from foreign material that may be in the reactor vessel.
By using high-purity water as the operating fluid, the drives can use simple internal piston seals, which allow leakage into the reactor vessel. Dynamic shaft or push rod seals and their attendant systems and wear problems are eliminated.

Control rod entry from below the core provides the best axial flux shaping and resultant fuel economy for the BWR.

The number of drives supplied with a reactor is selected to give the optimum power distribution in the core and to give the operator the maximum degree of control flexibility during startup, maneuvering, and flux shaping.

Enhancement of the ABWR and ESBWR converts from the complex mechanical-hydraulic control rod drive system to a simple fine motion control rod drive (Figure 3.11).

FIGURE 3.11
Fine-motion control rod drive cross section.
### 3.3 Reactor Core Design

#### 3.3.1 Introduction

The design of the BWR core and fuel is based on the proper combination of many design variables and operating experience. These factors contribute to the achievement of high reliability, excellent performance, and improved fuel cycle economy.

Discussed in this section are such design parameters as moderator-to-fuel volume ratio, core power density, thermal-hydraulic characteristics, fuel exposure level, nuclear characteristics of the core and fuel, heat transfer, flow distribution, void content, cladding stress, heat flux, and the operating pressure. Design analyses and calculations employed in this scope design have been verified by comparison with data from operating plants. General Electric continually evaluates this combination of design variables to be certain that changing conditions, which may significantly affect fuel cycle economics, are properly considered, and that the resulting final core design represents an optimum combination of the variables. The basic core configuration is shown in Figure 3.12, and a single fuel module cell is shown in Figure 3.13.

Several important features of the BWR core design are summarized below.

- The BWR core mechanical design is based on conservative application of stress limits, operating experience, and experimental test results. The moderate pressure level characteristics of a direct cycle reactor (approximately 1000 psia [6900 kPa]) reduce cladding temperatures and stress levels.
- The low coolant saturation temperature, high heat transfer coefficients, and neutral water chemistry of the BWR are significant, advantageous factors in minimizing Zircaloy temperature and associated temperature-dependent corrosion and hydride buildup. This results in improved cladding performance at long

![Figure 3.12](image)

**FIGURE 3.12**
ABWR core configuration.
exposures. The relatively uniform fuel cladding temperatures throughout the BWR core minimize migration of the hydrides to cold cladding zones and reduce thermal stresses.

- The basic thermal and mechanical criteria applied in the BWR design have been proven by irradiation of statistically significant quantities of fuel. The design heat fluxes and linear thermal outputs (approximate maximum of 13.4 kW/ft [44 kW/m]) are similar to values proven in fuel assembly irradiation.
- The design power distribution used in sizing the core represents a worst expected state of operation. Provisions for nonoptimal operation allow operational flexibility and reliability.
- The reactor is designed so that the peak bundle power at rated conditions is significantly less than the critical power limit.
- Because of the large negative moderator density (void) coefficient of reactivity, the BWR has a number of inherent advantages. These are the use of coolant flow as opposed to control rods for load following, the inherent self-flattening of the radial power distribution, the ease of control, the spatial xenon stability, and the ability to override xenon to follow load.

### 3.3.2 Core Configuration

The reactor core of the BWR is arranged as an upright cylinder containing many fuel assemblies and located within the reactor vessel. The coolant flows upward through the core. Important components of this arrangement are described in the following sections.

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**FIGURE 3.13**

Fuel module (cell).
3.3.3 Description of Fuel Assembly

The BWR core comprises essentially only two components: fuel assemblies (Figure 3.14) and control rods (Figure 3.15).

3.3.3.1 Fuel Rod

A fuel rod consists of UO$_2$ pellets and a Zircaloy 2 cladding tube. UO$_2$ pellets are manufactured by compacting and sintering UO$_2$ powder in cylindrical pellets and grinding to size. The immersion density of the pellets is approximately 95% of theoretical UO$_2$ density.

A fuel rod is made by stacking pellets into a Zircaloy 2 cladding tube, which is evacuated, back-filled with helium under a pressure of three atmospheres, and sealed by welding Zircaloy end plugs in each end of the tube. The Zircaloy tube is 0.483 in. (12.3 mm) in diameter, 160-1/4 in. (4.07 m) long, with a 32-mil (0.81 mm) wall thickness. The pellets are stacked to an active height of 150 in. (3.8 m), with the top 9.5 in. (241 mm) of tube available as a fission gas plenum. A plenum spring is located in the plenum space to exert a downward force on the pellets; this plenum spring keeps the pellets in place during the pre-irradiation handling of the fuel bundle. The selected dimensions result in a 9-mil (0.23 mm) diameter gap between the pellet and the tube.
3.3.3.2 Design Basis of Fuel Rods

The BWR fuel rod is designed as a pressure vessel. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, is used as a guide in the mechanical design and stress analysis of the fuel rod.

The rod is designed to withstand the applied loads, both external and internal. The fuel pellet is sized to provide sufficient volume within the fuel tube to accommodate differential expansion between fuel and cladding. Overall fuel rod design is conservative in its accommodation of the mechanisms affecting fuel in a BWR environment.

3.3.3.3 Fuel Bundle

Each fuel bundle contains 100 rods, which are spaced and supported in a square (10 × 10) array by lower and upper tie plates. The lower tie plate has a nosepiece, which fits into the
fuel support piece and distributes coolant flow to the fuel rods. The upper tie plate has a handle for transferring the fuel bundle. Both tie plates are fabricated from stainless steel and are designed to satisfy flow and mechanical strength considerations. Mechanically, these parts are designed to stay within the yield strength of the material during normal handling operations.

Three types of rods are used in a fuel bundle: tie rods, water rods, and standard fuel rods. The eight tie rods in each bundle have threaded end plugs which screw into the lower tie plate casting and extend through the upper tie plate casting. A stainless steel hexagonal nut and locking tab is installed on the upper end plug to hold the assembly together. These tie rods support the weight of the assembly only during fuel handling operations when the assembly hangs by the handle; during operation, the lower tie plate supports the fuel rods. Two rods in the interior foursome within each bundle (diagonally opposite the control blade) are water rods, that is, tubes of Zr-2 cladding without UO₂ fuel. Small holes are located at the lower and upper ends, allowing water to be driven through the rod, thus introducing moderating material within the bundle interior. One water rod also serves as the spacer-positioning rod, being mechanically locked to each of the seven spacer assemblies, thereby fixing the axial position of each spacer. The fuel rod spacers are equipped with Inconel springs to maintain rod-to-rod spacing. The remaining 54 fuel rods in a bundle are standard rods having a single tube of fuel pellets the same length as the tie rods. The end plugs of the spacer-capture rod and the standard rods have pins, which fit into anchor holes in the tie plates. An Inconel expansion spring located over the top end plug pin of each fuel rod keeps the fuel rods seated in the lower tie plate while allowing them to expand axially by sliding within the holes in the upper tie plate to accommodate differential axial thermal expansion.

The initial core has an average enrichment ranging from approximately 1.7 wt% U-235 to approximately 2.0 wt% U-235 depending on initial cycle requirements. Individual fuel bundle enrichments in the initial core are of three or four average enrichments, ranging from that of natural uranium, 0.711 wt% U-235, to a maximum of approximately 2.2 wt% U-235. The design of the initial core achieves an optimal balance of fuel economy, operating margins, and ease of transition to equilibrium cycle refueling.

Different U-235 enrichments are used in fuel bundles to reduce local power peaking. Low enrichment uranium rods are used in corner rods and in the rods nearer the water gaps; higher enrichment uranium is used in the central part of the fuel bundle. Selected rods in each bundle are blended with gadolinium burnable poison. The fuel rods are designed with the characteristics described below.

Poison: A material that absorbs neutrons unproductively and hence removes them from the fission chain reaction in a reactor, thereby decreasing its reactivity.

Mechanical end fittings, one for each of the enrichments: End fittings are designed so that it is not mechanically possible to complete assembly of a fuel bundle with any high-enrichment rods in positions specified to receive a low enrichment.

3.3.3.4 Bundle Features

The design has two important features:

- The bundle design places minimum external forces on a fuel rod; each fuel rod is free to expand in the axial direction.
- The unique structural design permits the removal and replacement, if required, of individual fuel rods.
3.3.3.5 Fuel Channel

A fuel channel encloses the fuel bundle; the combination of a fuel bundle and a fuel channel is called a fuel assembly. The channel is a square-shaped tube fabricated from Zircaloy 4; its outer dimensions are 5.455 in. (138.6 mm) × 5.455 in. (138.8 mm) × 166.9 in. (4.2 m) long. The reusable channel makes a sliding seal fit on the lower tie plate surface. The channel fastener assembly (consisting of a spring and a guard) attaches the reusable channel to the upper tie plate and a capscrew secured by a lock washer. Spacer buttons are located on two sides of the channel to properly space four assemblies within a core cell. The fuel channels direct the core coolant flow through each fuel bundle and also serve to guide the control rod assemblies.

The use of the individual fuel channel greatly increases operating flexibility because the fuel bundle can be separately orificed and thus the reload fuel design can be changed to meet the newest requirements and technology. The channels also permit fast in-core sampling of the bundles to locate possible fuel leaks.

3.3.4 Core Design

The reactor core is designed to operate at rated power with sufficient design margin to accommodate changes in reactor operations and reactor transients without damage to the core. In order to accomplish this objective the core is designed, under the most limiting operating conditions and at 100% of rated power, to meet the bases detailed below.

- The maximum linear heat generation rate, in any part of the core, is <13.4 kW/ft (44 kW/m).
- Less than 0.1% of the core experiences transition boiling during the worst expected transient.

3.3.4.1 Power Distribution

The design power distribution is divided into several components for convenience. The relative assembly power peaking factor is the maximum fuel assembly average power divided by the reactor core average assembly power. The axial power peaking factor is the maximum heat flux of a fuel assembly divided by the average heat flux in that assembly. The local power peaking factor is the maximum fuel rod heat flux at a horizontal plane in an assembly divided by the average fuel rod heat flux at that plane. Peaking factors vary throughout an operating cycle, even at steady-state full power operation, because they are affected by withdrawal of control rods to compensate for fuel burnup.

3.3.4.2 Axial Distribution

Because of the presence of steam voids in the upper part of the core, there is a natural characteristic for a BWR to have the axial power peak in the lower part of the core. During the early part of an operating cycle, bottom-entry control rods permit a reduction of this axial peaking by locating a larger fraction of the control rods in the lower part of the core. At the end of an operating cycle, the higher accumulated exposure and greater depletion of the fuel in the lower part of the core reduces the axial peaking. The operating procedure is to locate control rods so that the reactor operates with approximately the same axial power shape throughout an operating cycle.
3.3.4.3 *Relative Assembly Power Distribution*

The maximum-to-average fuel bundle peaking or radial distribution is reduced in a BWR core because of greater steam voids in the center bundles of the core. A control rod operating procedure is also used to maintain approximately the same radial power shape throughout an operating cycle.

3.3.4.4 *Local Power Distribution*

The local power distribution is reduced by the use of several uranium enrichments in a fuel bundle. Lower uranium enrichments are located near the water gaps, and higher enrichments are located in the center of a fuel bundle.

3.3.4.5 *Core Thermal Hydraulics*

*Central UO₂ temperature:* The maximum UO₂ temperature will occur in new fuel operating at the maximum linear heat generation rate of 13.4 kW/ft (44 kW/m). Based on published conductivity data, the maximum temperature is approximately 3400°F (1871°C).

*Core orificing:* Fixed orifices accomplish control of core flow distribution among the fuel assemblies. These orifices are located in the fuel support pieces and are not affected by fuel assembly removal and replacement.

The core is divided into two orifice zones. The outer zone of fuel assemblies, located near the core periphery, has more restrictive orifices than the inner zone, so flow to the higher power fuel assemblies is increased. The orificing of all fuel assemblies increases the flow stability margin.

Three types of boiling heat transfer must be considered in defining thermal limits: nucleate boiling, transition boiling, and film boiling. Nucleate boiling is the extremely efficient mode of heat transfer in which the BWR is designed to operate. Transition boiling is manifested by an unstable fuel cladding surface temperature which rises suddenly as steam blanketing of the heat transfer occurs, then drops to the nucleate boiling temperature as the steam blanket is swept away by the coolant flow, then rises again. At still higher bundle powers, film boiling occurs, which results in higher fuel cladding temperatures. The cladding temperature in film boiling (and possibly the temperature peaks in transition boiling) may reach values that could cause weakening of the cladding and accelerated corrosion. Overheating is conservatively defined as the onset of the transition from nucleate boiling. The core and fuel design basis has been defined, accommodating uncertainties such that margin is maintained between the most limiting operating condition and the transition boiling condition at all times in core life.

3.3.4.6 *Thermal-Hydraulic Analysis*

A computer program is used to analyze the thermal and hydraulic characteristics of the reactor core as a whole. The geometric, hydraulic, and thermal characteristics of the core design are represented, including number of fuel assemblies in each orifice zone of the reactor core, fuel assembly dimensions, friction factors and flow restrictions, and the flow characteristics of the fuel orifices, inlet plenum region of the reactor, along with bypass and leakage flow paths around the fuel assembly channels. Individual cases are analyzed by providing reactor power, flow, inlet enthalpy, and appropriate power distribution factors as input to the above computer program. The output of the
program includes the calculated flow distribution among the several channel types and a detailed analysis of the heat fluxes, steam quality, void fraction, and maximum critical power ratio (MCPR) at as many as 24 axial nodes for the average and peak power fuel assemblies in each orifice zone.

Comparisons of the analytical models used with fuel assembly design details such as fuel-rod-to-fuel-rod and fuel-rod-to-fuel-assembly-channel clearances and spacer configurations have been made to ensure that the computer programs adequately represent the actual core and fuel design, and that design correlations are applicable.

In addition, fuel thermal design calculations, including calculation of fuel rod temperature, UO₂ pellet thermal expansion characteristics, and rate of UO₂ swelling due to irradiation have been performed. Thermal effects of irradiation, including reduction in local power peaking factor due to U-235 depletion, buildup of plutonium near the surface of the pellet, and effect of gap width and gas composition on gap conductance, have been considered in confirming that the thermal-hydraulic performance objectives will be met.

3.3.4.7 Core Nuclear Characteristics

Nuclear calculations are based on nuclear data selected from the best current sources of information throughout the nuclear industry and on mathematical computer codes developed by General Electric for the BWR.

3.3.4.7.1 Reactivity Coefficients

In a BWR, two reactivity coefficients are of primary importance: the fuel Doppler coefficient and the moderator density reactivity coefficient. The moderator density reactivity coefficient may be broken into two components: that due to temperature and that due to steam voids.

3.3.4.7.2 Fuel Doppler Reactivity Coefficient

As in all light water-moderated and low-enrichment reactors, the fuel Doppler reactivity coefficient is negative and prompt in its effect, opposing reactor power transients. When reactor power increases, UO₂ temperature increases with minimum time delay, resulting in higher neutron absorption by resonance capture in the U-238.

3.3.4.7.3 Moderator Density Reactivity Coefficient

During normal plant operations, the steam void component of the moderator density reactivity coefficients is of prime importance. The steam void component is large and negative at all power levels. At full rated power, the steam voids are equivalent to approximately 3% reactivity.

- The fuel assembly design is such that the moderator density reactivity coefficient of the water within the fuel channel is negative for all conditions of operation. The in-channel moderator coefficient is smallest at the cold, zero power condition.
- The large and negative moderator density coefficient at operating power levels is due to the steam void effect. This steam void effect results in the operating advantages listed below.
- **Xenon override capability:** Because the steam void reactivity effect is large compared with xenon reactivity, the BWR core has the excellent capability of overriding the xenon effect, thereby increasing power after a power decrease.
• **Xenon stability:** The steam void reactivity is the primary factor in providing the high xenon stability characteristic.
• **Load changing by flow control:** Because the fuel Doppler reactivity opposes a change in load, the void effect must be and is larger than the fuel Doppler effect to provide load changing capability by flow (or moderator density) control.
• **Thermal-hydraulic stability:** The negative void effect is an important contributor to reactor thermal-hydraulic stability.

### 3.3.4.8 Reactivity Control

The movable boron-carbide (B₄C) control rods are sufficient to provide reactivity control from the cold shutdown condition to the full load condition. Additional reactivity control in the form of solid burnable poison is used only to provide reactivity compensation for fuel burnup or depletion effects.

The movable control rod system is capable of maintaining the reactor in a subcritical condition when the reactor is at ambient temperature (cold), zero power, zero xenon, and with the strongest control rod fully withdrawn from the core, in order to provide greater assurance that this condition can be met in the operating reactor, the core design is based on calculating a reactivity less than 0.99%, or a 1% margin on the stuck rod condition.

Supplementary solid burnable poisons are used to assist in providing reactivity compensation for fuel burnup. For all operating cycles, the supplementary control is provided by gadolinium mixed into a portion of the UO₂ reload fuel rods.

### 3.3.4.9 Margin between Operating Limits and Damage Limits

Two mechanisms that could result in fuel damage (i.e., perforation of the cladding) are

• Severe overheating of the fuel cladding
• Fracture of the fuel cladding due to excessive strain resulting from UO₂ thermal expansion

Although significant weakening of the fuel cladding due to overheating is not expected to occur until well into the film boiling region, fuel damage is conservatively defined as the onset of transition boiling. This, by definition, corresponds to MCPR = 1.00. In addition to this limit, a statistical margin of approximately 6% is made to allow for the various uncertainties in predicting and detecting the actual boiling state. So, during the worst expected transient, the MCPR is not permitted to go below a value of approximately 1.06. An additional margin for the effects of the worst transient produces the normal operating limit. A typical value for this operating limit is MCPR = 1.23. During full power operation, the fuel will typically operate with an MCPR >1.30. The difference between the actual operating value of MCPR and the operating limit is termed the *operating margin*.

A value of 1% plastic strain of Zircaloy cladding is conservatively defined as the limit below which fuel damage is not expected to occur. Available data indicate that the threshold for damage in irradiated Zircaloy cladding is in excess of this value. The linear heat generation rate required to cause this amount of cladding strain is approximately 25 kW/ft (82 kW/m). The linear heat generation rate for the worst expected transient is approximately 16 kW/ft (52.5 kW/m). During normal full power operation, the maximum linear heat generation rate will not exceed 13.4 kW/ft (44 kW/m).
3.3.4.10 Strain Localization

It was determined for $7 \times 7$ BWR fuel that strain localization due to pellet-to-cladding interaction at pellet interfaces (ridging) and pellet cracks can cause a small but statistically significant number of fuel rod perforations during normal reactor operation. The fuel design improvements listed below have been made for $8 \times 8$ BWR/6 fuel to reduce pellet-to-cladding localized strain.

- The fuel pellet length-to-diameter ratio is decreased from 2:1 to 1:1, which reduces ridging.
- The fuel pellet is chamfered, which reduces ridging.
- The maximum linear heat generation is decreased from 18.5 kW/ft (60.7 kW/m) to 13.4 kW/ft (44 kW/m), which reduces thermal distortion and ridging.
- The cladding heat treatment procedure is improved to reduce the variability of the cladding ductility.

3.3.4.11 Reactor Stability

The large fuel time constant and inherent negative moderator feedback are major contributors to the stability of the BWR. The Doppler reactivity feedback appears simultaneously with a change in fuel temperature and opposes the power change that caused it, while heat conduction to water and the subsequent formation of steam voids must always transfer heat through the fuel material. Because the Doppler reactivity opposes load changes, it is desirable to maintain a large ratio of moderator-to-Doppler coefficient for optimum load following capability. The BWR takes advantage of its inherently large moderator-to-Doppler coefficient ratio by permitting a variation of coolant flow for load following.

Xenon instability is an oscillatory phenomenon of xenon concentration throughout the reactor that is theoretically possible in any type of reactor. If such a condition should occur, it can restrict load following performance, cause increased local power peaking in the core, and possibly reduce the fuel economics of the core. The BWR as designed by General Electric has characteristics that provide a large margin of damping to such oscillations. This is primarily brought about by the high negative power coefficient characteristic of the core. In addition, the use of in-core ion chambers for local monitoring of core conditions and for local reactivity adjustment brought about by the control rods and local steam void control provide complete knowledge of core conditions and adequate control capability. Xenon oscillations are local phenomena within the core. They are not evident when looking at core averaged values and, unless in-core instrumentation is provided, the presence of such oscillations may not be known until they have caused power peaking with possible core damage.

If the magnitude of the power coefficient of reactivity becomes too small, spatial xenon oscillations will occur and restrict reactor load following and performance. Even in the stable region, it is important to have well-damped power distributions and to select reactor load following variables which do not tend to encourage spatial xenon oscillations. Current BWR designs result in power coefficients well beyond the range of instability of xenon. This advantage of the BWR is of major importance for large, loosely coupled nuclear cores. Flow control further aids spatial xenon stability by providing a power shape, which remains relatively constant at varying reactor power levels.

The water-to-fuel volume ratio is determined from consideration of the reactivity coefficient for safe and stable operation. This ratio is selected to provide cold lattice coefficients,
which preclude detrimental startup transients. In parallel to the allowance of considerable margin in design for good load following and spatial xenon stability, the water-to-fuel volume ratio selected is close to the optimum for minimum fuel cycle costs.

3.3.5 Reactor Reactivity Control

3.3.5.1 Control Rods

Control rods (Figure 3.15) using $\mathrm{B}_4\mathrm{C}$ compacted in stainless steel tubes were introduced into service in 1961. Since then, this design has been the standard reference control element in all General Electric BWRs, and has replaced the 2% boron-steel rods previously used. Over the years, $\mathrm{B}_4\mathrm{C}$ control rods have been produced routinely in quantity by tested manufacturing procedures. During the years since the first rods were placed in service, they have demonstrated excellent mechanical and nuclear performance.

The control rods perform dual functions of power distribution shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of selected patterns of rods. The rods, which enter from the bottom of the near-cylindrical reactor, are positioned in such a manner to counterbalance steam voids in the top of the core and affect significant power flattening. These groups of control elements, used for power flattening, experience a somewhat higher duty cycle and neutron exposure than the other rods.

The reactivity control function requires that all rods be available for reactor *scram* (prompt shutdown) or reactivity regulation. Control elements are therefore mechanically designed to withstand the dynamic forces resulting from a scram. They are connected to bottom-mounted, hydraulically actuated drive mechanisms that allow axial positioning for reactivity regulation or rapid scram insertion. The design of the rod-to-drive connection permits each blade to be attached or detached from its drive during refueling without disturbing the remainder of the control functions. The bottom-mounted drives permit the entire control function to be left intact and operable for tests with the reactor vessel open.

Control rods are cooled by the core leakage (bypass) flow. The core leakage flow is made up of recirculation flow that leaks through several leakage flow paths:

- Four holes in fuel assembly nosepiece (lower tie plate)
- The area between fuel channel and fuel assembly nosepiece
- The area between fuel assembly nosepiece tie and fuel support piece
- The area between fuel support piece and core plate
- The area between core plate and shroud
- Holes in the core plate for bypass flow control

3.3.5.2 Control Rod Nuclear Characteristics

The control rod system is designed so that adequate shutdown capability is available at all times. To permit a margin for credible reactivity changes, the control system has the capability to shut down and maintain the core continuously subcritical with the maximum worth control rod fully withdrawn. This capacity is experimentally demonstrated when reactivity alterations are made to the reactor core. The use of mechanical control permits periodic tests on the core reactivity during refueling. Control rods are withdrawn adjacent to an inserted fresh fuel assembly to verify subcriticality and predicted excess reactivity
of the fuel. The control rod insertion rates on scram are sufficient to protect the reactor against damage in all transients, which are expected to occur during the life of the plant.

Control rods are used primarily for power distribution shaping and for shim control of long-term reactivity changes, which occur as a result of fuel irradiation. The flow control function, which is used to follow rapid load changes, reduces requirements on speed of control rod response and thus improves plant safety. Every 2–3 months, the control rod patterns are altered to provide more uniform fuel and control rod burnup. In normal daily operation, little control rod movement is required for depletion of reactivity. The resulting low frequency of control rod changes reduces the possibility of operator error.

With the normal control rod patterns required to maintain an acceptable power distribution in the operating core, an average control rod will be worth about 0.005 d/k effective. The maximum worth of a rod in a typical power operation pattern will be about 0.01 d/k effective. The notch increment dimensions and spacing of the rods are set to limit the reactivity insertion to about 0.0003 d/k for any notch increment of control withdrawn. Preplanned withdrawal patterns and procedural patterns and procedural controls are used to prevent abnormal configurations yielding excessively high rod worth.

The velocity limiter is a mechanical device which is an integral part of the control rod assembly and protects against the low probability of a rod drop accident. It is designed to limit the free fall velocity and reactivity insertion rate of a control rod so that fuel damage would not occur. It is a one-way device in that control rod scram time (or fast insertion) is not significantly affected.

### 3.3.5.3 Supplementary Reactivity Control

The control requirements of the initial core are designed to be considerably in excess of the equilibrium core requirements because all of the fuel is fresh in the initial core. The initial core control requirements are met by use of the combined effects of the movable control rods and a supplemental burnable poison.

Only a few materials have nuclear cross sections suitable for burnable poisons. An ideal burnable poison must deplete completely in one operating cycle so that no poison residue exists to penalize initial U-235 enrichment requirements. It is also desirable that the positive reactivity from poison burnup matches the almost linear decrease in fuel reactivity from fission product buildup and U-235 depletion. A self-shielded burnable poison consisting of Gd₂O₃ dispersed in a few selected fuel rods in each fuel assembly provides the desired characteristics. Gd₂O₃ depletes as a cylinder with decreasing radius to provide a linear increase in reactivity. The concentration is selected so that the poison essentially depletes in the operating cycle. It is possible to improve power distributions by spatial distribution of the burnable poison.

### 3.4 Reactor Auxiliary Systems

#### 3.4.1 Introduction

Because the reactor is basically a water boiler, process systems are required which clean and control the chemistry of the water in the reactor vessel as well as protect the reactor core. Called the reactor auxiliary systems, these systems may be divided into two general
categories: systems necessary for normal nuclear boiler operations, including startup and shutdown; and systems that accommodate or provide backup in case of an abnormal condition.

Auxiliary systems used during normal plant operation include

- The RWCU system
- The fuel building and containment pools cooling and filtering system
- The closed cooling water system for reactor services
- The shutdown cooling function of the RHR system
- Radioactive waste treatment system
- Off-gas treatment system

Backup auxiliary systems used during abnormal plant operation include the RCIC system, the SBLC system, the steam condensing function of the RHR system (hot standby), and the suppression pool cooling function of the RHR system. Other process systems, commonly referred to as emergency core cooling systems (ECCS), are designed as safety systems to mitigate the consequences of postulated emergency situations that could otherwise lead to core damage and release of fission products to the environs. ECCS consists of the LPCI function of the RHR system, the HPCS system, the LPCS system, and automatic depressurization (blow down). The essential service water system and the area cooling systems (which service the areas where ECCS equipment is located) are also required during abnormal plant operation.

3.4.2 RWCU System

The purpose of the RWCU system is to maintain high reactor water quality by removing fission products, corrosion products, and other soluble and insoluble impurities. In addition, the system provides a means for water removal from the primary system during periods of increasing water volume.

The cleanup system is sized to process the water volume of the reactor system in approximately 3–3.5 h. The system can be operated during startup, shutdown, and refueling operation, as well as during normal plant operations. Water is removed from the reactor through the reactor recirculation pump suction line and returned through the feedwater line. Under normal operation, the water is removed at reactor temperature and pressure and pumped through regenerative and nonregenerative heat exchangers (NRHXs) where it is cooled, and then through the filter-demineralizer units. The flow continues through the shell side of the regenerative heat exchanger (RHX), where it is heated before returning to the reactor.

During startup and other times of increasing water volume, excess water is normally removed from the reactor by blowdown through the cleanup system to the main condenser, or alternately to the waste collector tank, or waste surge tank. During this operation, the return flow to the RHX is reduced, thereby reducing the cooling capability of this exchanger and correspondingly increasing the duty of the NRHX. The NRHX is designed to cool reactor water flow to the filter-demineralizer units to approximately 120°F during normal operation and reactor vessel blowdown. Cooling water is supplied to the NRHX by the closed cooling water system for reactor service.

The operation of the RWCS is controlled from the main control room. Filter resin backwashing and precoating operations are controlled from a local panel. The cleanup system
is isolated from the reactor automatically by closure of motor-operated isolation valves on any of the following signals:

- High temperature after the NRHX
- Low reactor water level
- SBLC solution injection
- High ambient temperature in the cleanup system equipment area
- High flow rate differential between system inlet and outlet
- High differential temperature across the system’s ventilation system

3.4.3 Fuel Building and Containment Pools Cooling and Cleanup System

The fuel building and containment pools cooling and cleanup system accommodates the spent fuel cooling heat load as well as drywell heat transferred to the upper containment pool. The equipment for the cooling and cleanup systems consists of circulating pumps, heat exchangers; filter-de-mineralizers, and the required piping, valves, and instrumentation. Pumping loops circulate pool water through the heat exchangers and fuel pool filters and returns the flow by discharging it through diffusers mounted in the fuel storage pool and in the containment pool. The suction for the circulating pumps is taken from the skimmer surge tank. The skimmer surge tank is fed from skimmers located at the top of these pools.

The upper containment pool has a shield wall with a removable gate between the reactor well, the fuel holding pool, and the fuel transfer pool. With the gate inserted in the slot, the upper containment pool can be drained for work at the pressure vessel flange level. With the pools full of water, the gates are removed during refueling operations to permit the transfer of fuel and equipment between pools.

The RHR system heat exchangers are also available to supplement the fuel pool cooling heat exchangers. RHR system heat exchangers are not normally required but may be needed if more than the normal number of spent fuel assemblies is stored in the pool. The system pumps and heat exchangers are located in the fuel building below the normal fuel pool water level. Heat exchangers are cooled by essential plant service water.

Essential service water may become radioactive because it collects corrosion products normally located in the fuel building. Pool water is usually filtered continually.

Because there are no drain connections at the bottom of the fuel storage pool, the spent fuel assemblies can never be exposed by an accidental valve opening or pipe break. Fuel is not stored in the upper containment pool during normal operation. A portable underwater vacuum system, similar to that used in swimming pools, can be used to clean pool walls, floors, and internals removed from the reactor vessel. Deposition at the water line of the pool walls is minimized by several surface skimmers. These devices remove a portion of the surface water and recycle it to the pool.

3.4.4 Closed Cooling Water System for Reactor Service

The closed cooling water system consists of a separate, forced circulation loop. This system uses water piped from the site service water source to provide a heat sink for selected nuclear system equipment. Its purpose is to provide a second barrier between the primary systems containing radioactive products and the service water system that is the final heat
sink, thereby eliminating the possibility of radioactive discharge into plant effluents that could result from heat exchanger leaks. The plant service water pumps provide coolant to the closed cooling water system for reactor service which in turn generally service the following equipment:

- Reactor recirculation pump seal coolers
- Reactor recirculation pump motor coolers
- Nonregenerative cleanup heat exchanger
- Clean sump coolers
- Sample coolers
- Drywell coolers
- Cleanup recirculation pump coolers
- Off-gas system glycol coolers
- Control rod drive supply pumps
- Radwaste concentrator condensers
- Radwaste concentrated waste tank
- Control rod drive supply pumps

Any possible radioactive leakage from the foregoing reactor equipment would be to, and would be confined in, the closed loop cooling water system, which is monitored continuously for radioactivity. A surge tank is used to accommodate system volume swell and shrinkage and to provide a means for adding makeup water and inhibitors.

The closed cooling water system design temperature depends on the maximum temperature of service water intake. The closed cooling water system satisfies the plant’s full power load requirements. Extra cooling capability, with all spares operating, is adequate to handle plant startup duty.

### 3.4.5 Emergency Equipment Cooling System

The emergency equipment cooling system (EECS) services certain equipment required for normal and emergency shutdown of the plant. The system provides cooling water for the RHR system pump motor and pump seal cooler and the HPCS and LPCS systems pump motors and pump seal coolers. Upon loss of normal ventilation, such as may occur upon loss of external AC power, the EECS provides ventilation and cooling for the HPCS and LPCS systems, the RHR system, and the reactor core ICS equipment as required preventing overheating. On failure of any single component, the EECS will service at least two RHR system pumps or one RHR system pump and the LPCS system pump, the HPCS system pump, and any standby core cooling system equipment being serviced by the EECS.

### 3.4.6 SBLC System

The SBLC system is a redundant control system capable of shutting the reactor down from rated power operation to the cold condition in the postulated situation that the control rods cannot be inserted. The operation of this system is manually initiated from the reactor control room.
The equipment for the SBLC system is located in the reactor building and consists of a stainless steel storage tank; a pair of full capacity positive displacement pumps and injection valves; a test tank; and the necessary piping, valves, and instrumentation.

The SBLC system is adequate to bring the reactor from the hot operating condition to cold shutdown and to hold the reactor shutdown with an adequate margin when considering temperature, voids, Doppler effect, equilibrium, xenon, and shutdown margin. It is assumed that the core is operating at normal xenon level when injection of liquid control chemical is needed.

The liquid control chemical used is boron in the form of sodium pentaborate solution. It is injected into the bottom of the core where it mixes with the reactor coolant. The sodium pentaborate is stored in solution in the SBLC tank. Electric heaters automatically keep the solution above the saturation temperature. The system temperature and liquid level in the storage tank are monitored, and abnormal conditions are annunciated in the control room.

3.4.7 RCIC System

The RCIC system maintains sufficient water in the reactor pressure vessel to cool the core and then maintains the nuclear boiler in the standby condition in the event the vessel becomes isolated from the turbine steam condenser and from feedwater makeup flow. The system also allows for complete plant shutdown under conditions of loss of the normal feedwater system by maintaining the necessary reactor water inventory until the reactor vessel is depressurized, allowing the operation of the shutdown cooling function of the RHR system. The system delivers rated flow within 30 s after initiation. A water leg pump keeps the piping between the pump and the discharge shutoff valve full of water to ensure quick response and to eliminate potential hydraulic damage on system initiation.

Following a reactor scram during normal plant operation, steam generation continues at a reduced rate due to the core fission product decay heat. The turbine bypass system directs the steam to the main condenser, and the feedwater system provides makeup water required to maintain the reactor vessel inventory.

In the event the reactor vessel becomes isolated from the main condenser, the relief valves automatically (or by operator action from the control room) maintain vessel pressure within desirable limits. In the event feedwater becomes unavailable, the water level in the reactor vessel drops due to continued steam generation by decay heat. Upon reaching a predetermined low level, utilizing one-out-of-two-twice logic, the RCIC system is initiated automatically to maintain safe standby conditions of the isolated primary system. The turbine-driven pump supplies makeup water from one of the following sources capable of being isolated from other systems: the condensate storage tank (first source), the steam condensed in the RHR heat exchangers (second source), or the suppression pool (an emergency source). The turbine is driven with a portion of the decay heat steam from the reactor vessel and exhausts to the suppression pool. The makeup water is pumped into the reactor vessel through a connection to the reactor feedwater line.

A design flow functional test of the RCIC system may be performed during normal plant operation by drawing suction from the condensate storage tank and discharging through a full flow test return line to the condensate storage tank. The discharge valve to the reactor feedwater line remains closed during the test and reactor operation remains undisturbed. If the system requires initiation while in the test mode, the control system automatically returns to the operating modes. Cooling water for pump and turbine operations and for the lube oil cooler and the gland seal condenser is supplied from the discharge of the pump.
The RCIC system operates independently of auxiliary AC power, plant service air, or external cooling water systems. System valves and auxiliary pumps are designed to operate by DC power from the station batteries.

Two turbine control systems include a speed governor limiting the speed to its maximum operating level and a control governor with automatic set-point adjustment, which is positioned by demand signal from a flow controller. Manual operation of the control governor is possible when in the test mode but automatically repositioned by the demand signal from the controller if system initiation is required. The operator can select manual control of the governor, and adjust power and flow to match decay heat steam generation.

The turbine and pump automatically shut down upon:

- Turbine overspeed
- High water level in the reactor vessel
- Low pump suction pressure
- High turbine exhaust pressure
- Automatic isolation signal

The steam supply system to the turbine is automatically isolated upon:

- Large pressure drop across two pipe elbows in the steam supply line
- High area temperature
- Low reactor pressure (two-out-of-two logic)
- High pressure between the turbine exhaust rupture diaphragms (two-out-of-two logic)

3.4.8 ECCS

The ECCS comprises the LPCI function of the RHR system, HPCS and LPCS systems, and automatic depressurization of the primary system. The ECCS is designed to perform the following functions:

- Prevent fuel cladding fragmentation for any mechanical failure of the nuclear boiler system up to, and including, a break equivalent to the largest nuclear boiler system pipe
- Provide this protection by at least two independent, automatically actuated cooling systems
- Function with or without external (off-site) power sources
- Permit testing of all the ECCSs by acceptable methods, including, wherever practical, testing during power plant operations
- Provide this protection for long time periods and from secure sources of cooling water with the capability of dissipating the rejected heat for a minimum of 30 days

The aggregate of the ECCS is designed to protect the reactor core against fuel cladding damage (fragmentation) across the entire spectrum of line break accidents. The operational capability of the various ECCSs to meet functional requirements and performance objectives is outlined in the following paragraphs.
The operation of the ECCS network is automatically activated by the reactor protection system upon redundant signals that are indicating low reactor vessel water level or high drywell pressure or a combination of indicators showing low reactor vessel water level and high drywell pressure.

During the first 10 min following initiation of operation of the ECCS, any one of the following three combinations satisfies the functional requirements of the system objectives:

- The operation of the automatic depressurization function, the HPCS system, and two LPCI loops of the RHR system (failure of division 1)
- The operation of the automatic depressurization function, the HPCS system, the LPCS system, and one LPCI loop of the RHR system (failure of division 2)
- The operation of the automatic depressurization function, three LPCI loops of the RHR system, and the LPCS system (failure of division 3)

In the event of a break in a pipe that is part of the ECCS, any one of the following four combinations satisfies the functional requirement:

- The operation of the automatic depressurization function and two LPCI loops of the RHR system
- The operation of the automatic depressurization function, one LPCI loop of the RHR system, and the LPCS system
- The operation of the automatic depressurization function, the HPCS system, and one LPCI loop of the RHR system
- The operation of the automatic depressurization function, the HPCS system, and the LPCS system

A combination of the HPCS system or the LPCS system plus any two other ECCS pumps provides two phenomenological cooling methods (flooding and spraying).

After the first 10 min following the initiation of operation of the ECCS and in the event of an active or passive failure in the ECCS or its essential support system, one of the following two combinations satisfies the performance objectives and the requirement for removal of decay heat from the containment.

- Two LPCI loops of the RHR system with at least one heat exchanger and 100% service water flow
- Either the HPCS system or the LPCS system, one LPCI loop of the RHR system with one heat exchanger, and 100% service water flow

The separation of redundant equipment of the various systems that make up the ECCS is maintained to assure maximum operational availability. Electrical equipment and wiring for the engineered safeguard features of the ECCS are broken into segregated divisions, further assuring a high degree of redundancy.

The power for operation of the ECCS is from regular AC power sources. Upon loss of regular power, operation is from on-site standby AC power sources, and the standby diesel-generator set is capable of accommodating full capacity of the LPCI and spray function. The HPCS system is completely independent of external power sources, having its own diesel generator as shown.
The operation of ECCS pumps is also possible from a local key lock hand switch and from the control room but automatic signals pre-empt all others. In the event of normal power failure while the system is operating or in the process of going into operation, the system will restart from the standby sources. All system alarms annunciate in the control room.

Although the feedwater system is not considered a part of the ECCS, under some circumstances it could refill the vessel or at least maintain a water level, depending upon the location of the postulated break, for a given spectrum of break sizes. In the case of turbine-driven feedwater pumps, this additional coolant source would still be available from the electrically driven condensate pumps.

3.4.8.1 HPCS System

The purpose of the HPCS system is to depressurize the nuclear boiler system and to provide makeup water in the event of a loss of reactor coolant inventory. In addition, the HPCS system prevents fuel cladding damage (fragmentation) in the event the core becomes uncovered due to loss of coolant inventory by directing this makeup water down into the area of the fuel assemblies. The makeup water is jetted as a spray over the area of the fuel assemblies from nozzles mounted in a sparger ring located inside the reactor vessel above the fuel assemblies. The HPCS system is an integral part of the total design for ECC, which provides for adequate core cooling and depressurization for all rates of coolant loss from the nuclear boiler.

The HPCS system includes a sparger ring with spray nozzles located inside the reactor pressure vessel, a motor-driven pump, diesel generator, valves, piping, and instrumentation necessary to provide an operating system with the capability of being tested during plant operation.

Cooling water for the operation and testing of the HPCS system is from the condensate storage tank. Upon depletion of this supply, the system automatically transfers water from the containment suppression pool. Water inventory lost from the nuclear boiler system drains to the drywell to weir wall level and then into the suppression pool, thereby providing an inexhaustible supply of cooling water allowing continued operation of the HPCS system until the operator from the control room manually stops it. System piping and equipment are maintained full of condensate water at all times to avoid time delays in filling the lines and to avoid hydraulic hammer.

The HPCS system can operate independently of normal auxiliary AC power, plant service air, or the emergency cooling water system. Operation of the system is automatically initiated from independent redundant signals indicating low reactor vessel water level or high pressure in the primary containment. The system also provides for remote-manual startup, operation, and shutdown. A testable check valve in the discharge line prevents backflow from the reactor pressure vessel when the reactor vessel pressure exceeds the HPCS system pressure such as may occur during initial activation of the system. A low flow bypass system is placed into operation until pump head exceeds the nuclear system pressure and permits flow into the reactor vessel.

The HPCS system can be tested during normal plant operation or when the plant is shut down. During normal plant operation, pump suction is from the condensate storage tank with a full flow return line to the condensate storage tank. During plant shutdown, pump suction is from the primary containment pressure suppression pool with a full flow return line to the suppression pool. The control system provides for the automatic transfer to the service mode upon the presence of ECC demand signal.
The integrity of the piping internal to the reactor vessel is determined by comparing the difference in pressure between spray sparger and the bottom of the core area with the pressure drop across the core. An increase in this comparison initiates an alarm in the control room.

3.4.8.2 LPCS System

The function of the LPCS system is to prevent fuel cladding damage (fragmentation) in the event the core is uncovered by the loss of coolant. The cooling effect is accomplished by directing jets of water down into the fuel assemblies from spray nozzles mounted in a sparger ring located above the reactor core. The system is an integral part of the total design for ECC, which provides for adequate core cooling for all rates of coolant loss from the nuclear boiler. The system goes into operation once the reactor vessel pressure has been reduced and the operation of the other systems of the ECCS prove inadequate to maintain the necessary water level in the reactor vessel at the reduced vessel pressure.

The LPCS system includes a sparger ring with spray nozzles located in the reactor vessel above the core, a motor-driven pump, motor-operated valves, piping, valves, and instrumentation necessary to provide a system for required operation with the capability of being tested. The system is connected to the containment suppression pool for its supply of water for cooling and connectable to the RHR system for testing and flushing. The elevation of the pump, with respect to the minimum water level of the suppression chamber, ensures adequate net positive suction head. The system pump is protected from overheating during operation against high reactor vessel pressure or closed injection or test valves by a low-flow bypass line to the suppression pool. A water leg pump keeps the piping between the pump and the injection valve full of water to ensure quick response and to eliminate potential hydraulic damage on system initiation. In the event of complete loss of normal electrical power, the spray system may be operated (automatically or manually) from the standby diesel generator.

The operation of the LPCS system pump is initiated from independent, redundant signals indicating low-reactor-water level and/or high pressure in the drywell, both using a one-out-of-two-twice logic. The same signals initiate starting of the standby diesel generators. The motor-operated valve in the discharge line opens automatically upon activation of the pump and a permissive pressure differential across the valve. As the reactor vessel pressure decreases, the flow rate of water to the reactor vessel increases. A testable check valve in the discharge line located inside the containment precludes backflow from the reactor vessel when the vessel pressure is greater than the pump discharge pressure. The operation of the system can be initiated from the main control room.

Water lost from the reactor vessel collects in the drywell to the level of the weir wall and then flows into the suppression chamber. This establishes a closed loop allowing the spray system to continue to operate until the operator manually stops it.

A bypass line to the suppression pool capable of rated core spray flow permits testing while the power plant is in service. A motor-operated valve controls bypass flow and is operated by a key locked switch in the control room. The position of the valve (as is true for all air- or motor-operated valves) is indicated in the control room. The valve receives a signal to close, which pre-empts all others, in the event that operation of the LPCS system is required.

To allow for system testing during plant shutdown, reactor water, via a temporary connection (removable spool piece) to the RHR system, is discharged into the reactor vessel through the core spray sparger. The spool piece is removed prior to plant startup and the open pipe capped.
3.4.8.3 Automatic Depressurization Function

Blowdown, through selected safety/relief valves, in conjunction with the operation of the LPCI function of the RHR system and/or the LPCS system functions as an alternate to the operation of the HPCS system for protection against fuel cladding damage (fragmentation) upon loss of coolant over a given range of steam or liquid line breaks. The blowdown depressurizes the reactor vessel, permitting the operation of the LPCI function and/or the LPCS system. Blowdown is activated automatically upon coincident signals of low water level in the reactor vessel and high drywell pressure. A time delay of approximately 2 min after receipt of the coincident signals allows the operator time to bypass the automatic blowdown if the signals are erroneous or the condition has corrected itself. The operator can initiate blowdown from the control room at any time.

3.4.9 RHR System

The RHR system removes residual heat generated by the core under normal (including hot standby) and abnormal shutdown conditions. The LPCI function of the RHR system is an integral part of the ECCS. The design objectives of the system follow:

- To restore and maintain, if necessary, the water level in the reactor vessel after a LOCA so that the core is sufficiently cooled to prevent fuel cladding damage (fragmentation)
- To limit suppression pool water temperature
- To remove decay heat and sensible heat from the nuclear boiler system while the reactor is shut down for refueling and servicing
- To condense reactor steam so that decay and residual heat may be removed if the main condenser is unavailable (hot standby)
- To supplement the fuel and containment pools cooling and cleanup system capacity when necessary to provide additional cooling capability

The RHR system is made up of various subsystems with the following operational functions to satisfy these objectives.

3.4.9.1 LPCI

The LPCI function in conjunction with the LPCS system, the HPCS system, and/or automatic depressurization of the nuclear boiler system (depending upon operability of the HPCS system or level of depletion of reactor vessel water) will restore and maintain the desired water level in the reactor vessel required for cooling after a LOCA.

In conjunction with the LPCS system, redundancy of capability for core cooling is achieved by sizing the RHR pumps so that the required flow is maintained with one pump not operating. Using a split bus arrangement for pump power supply (essential power system), two RHR pumps are connected to one bus and the third RHR pump and an LPCS pump are connected to the second bus to obtain the desired cooling capability. The pumps deliver full flow inside the core shroud when the differential pressure between the reactor vessel and the containment approaches 20 psi (138 Pa). The availability of the LPCI function is not required during normal nuclear system startup or cool down when the reactor vessel gage pressure is <135 psi (931 kPa). The operability of the pumps can be tested at any time during normal plant operation by bypassing the reactor vessel and pumping the flow back to the pressure suppression pool.
3.5 Instrumentation and Controls

3.5.1 Introduction

The instrumentation of the BWR is generally associated with the control of the reactor, the prevention of the operation of the plant under unsafe or potentially unsafe conditions, the monitoring of process fluids and gases, and for monitoring of the performance of the plant. The control of the plant is from the control room. Instrumentation for monitoring the performance of the plant is located in the control room and locally.

Power output from the BWR is controlled by changes in reactor water recirculation flow rate or by the moving of control rods. As the reactor power output changes, the turbine initial pressure regulator adjusts the turbine admission valve to maintain nearly constant reactor pressure, admitting the new steam flow to produce the desired change in the turbine generator power output. The BWR is operated at constant reactor pressure because pressure changes caused by turbine throttle operation in response to load changes tend to bring about reactor power changes opposite to the desired change. However, small controlled pressure changes are used to improve load response.

3.5.2 Plant Startup

Startup of the plant from a cold standby condition to a power producing condition requires the

- Startup of the reactor water recirculation pumps
- Pumps brought to rated speed
- Manipulation of the reactor water recirculation flow control valves to provide the required flow
- Movement of control rods to attain the desired power level
- Monitoring of the reactor to record and monitor reactor behavior

The operator manually controls the startup of the plant from a cold standby condition.

3.5.2.1 Reactor Startup and Operation

The operational sequence for the startup of the plant from a cold standby condition is as follows:

- The flow control valves are set at the minimum position, which corresponds to approximately 25% of rated flow.
- The reactor water recirculation pumps are started. Because the low-frequency motor generator sets cannot start the recirculation pump motors, the pump motors are started from auxiliary power and transferred to the low frequency motor generator sets when the pump motor nears full speed and after the starting current has dropped.
- Control rods are manually withdrawn according to a predetermined schedule to achieve criticality of the reactor. They are further withdrawn to approximately 32% of rated power with the reactor water flow control valves fully open and the recirculation pumps operating at low speed (25%). The rate at which power level is raised is usually limited by conditions of thermal expansion of the reactor vessel.
- At approximately 32% of rated power, the reactor water flow control valves are closed and the recirculation pump transferred to auxiliary power and operated at rated speed.
- From approximately 30% to approximately 40% of rated power, the control of power level is by manual control of recirculation flow by changes in control valve position from minimum position.
- Above approximately 65% of rated core flow, the recirculation flow control is automatic.
- Between approximately 38% and approximately 75% of rated power, control rods are normally used to change power level.
- Above approximately 75% of rated power, change in reactor water recirculation flow is normally used to change power level.
- Neutron monitoring channels monitor the nuclear behavior of the reactor. Counting channels are used in the subcritical range up through criticality. The neutron counting channels and/or intermediate range monitoring channels monitor the intermediate range, from criticality to the power range. The power range neutron monitors are used throughout the power range usually above 21% of rated power.

During initial power operation, an operating curve is established relating reactor power to recirculation flow. The first point of the curve is full flow and rated power. When a rod pattern is established for this point, recirculation flow is reduced in steps at the same rod pattern, and the relationship of flow to power is plotted. Other curves are established at lower power ratings and other rod patterns as desired. During operation, flow control adjustment, rod positioning, or a combination of the two may change reactor power, while adhering to established operating curves. A rod withdrawal interlock is used to prevent unscheduled rod withdrawal, which would result in an excessive power-to-flow ratio. The operating curves are evaluated periodically, usually during startups, to compensate for changing reactivity coefficients. Although control rod movement is not required when the load is changed by recirculation flow adjustment, long-term transient reactivity effects are normally compensated for by control rod adjustment.

### 3.5.2.2 Turbine Startup

While the reactor temperature is being increased, the turning gear rotates the turbine. When reactor steam is available, the shaft seal steam is applied and the mechanical vacuum pump is started. After a partial vacuum is established in the main condenser, heating of the turbine and steam flow from the reactor are accomplished by first establishing a flow of steam to the condenser through the bypass valves. This flow is gradually transferred to the turbine until rated speed is achieved after which the unit is synchronized with the system. The initial pressure regulator controls the bypass flow during this initial period and the governor controls the turbine. The initial pressure regulator assumes normal control of the turbine admission valve after the unit is synchronized and a small amount of load is applied.

### 3.5.3 Power Operation

After the generator is synchronized to the electrical system and is producing a substantial output, the power output is adjusted to meet the system requirements by manual adjustment of control rods, manual or automatic adjustment of reactor recirculation flow, or a combination of these two methods.
3.5.3.1 Control Rod Adjustment

Withdrawing a control rod reduces the neutron absorption and increases core reactivity. Reactor power then increases until the increased steam formation just balances the change in reactivity caused by the rod withdrawal. The increase in boiling rate ends to raise reactor pressure, causing the initial pressure regulator to open the turbine admission valves sufficiently to maintain a constant pressure. When a control rod is inserted, the converse effect occurs.

The rate of power increase is limited to the rate at which control rods can be withdrawn. Control rods can be operated one at a time, or in groups of four rods in a symmetrical pattern. Single rods or rod groups can be withdrawn continuously or in incremental steps (notch steps). Continuous movement is usually limited to subcritical and heatup conditions. Control rod movement is the normal method of making large changes in reactor power, such as daily or weekly load shifts requiring reduction and increases of more than 25% of rated power.

3.5.3.2 Recirculation Flow Control

The BWR is unique in that reactor power output can be varied over a power range of approximately 25% of the operating power level by adjustment of the reactor recirculation flow without any movement of control rods. This is the normal method used for load following and maneuvering the reactor and allows for load following at rates of up to 1% of rated power per second.

Reactor power change is accomplished by using the negative power coefficient. An increase in recirculation flow temporarily reduces the volume of steam in the core by removing the steam voids at a faster rate. This increases the reactivity of the core, which causes the reactor power level to increase. The increased steam generation rate increases the steam volume in the core with a consequent negative reactivity effect, and a new constant power level is established. When recirculation flow is reduced, the power level is reduced in a similar manner.

The adjustment of the flow control valve changes the recirculation flow rate. To change reactor power, a demand signal from the operator or a load speed error signal from the speed governing mechanism is supplied to the master controller. A signal from the master controller adjusts the position setting of the controller for each valve. This signal is compared with the actual position of the valve associated with each controller. The resulting error signal causes adjustment of the valve position to reduce the error signal to zero, and the reactor power change resulting from the change in recirculation flow causes the initial pressure regulator to reposition the turbine control valves.

Automatic load control is accomplished by supplying a speed-load error signal from the turbine governor to the master controller. The energy storage capability of the water in the reactor system is used to increase the speed of response of the automatic load control system. An automatic, temporary change in the set point of the pressure regulator is produced when there is a demand for a change in turbine output. If an increase in load is demanded, the pressure set point is lowered and water in the reactor system flashes to produce extra steam flow to the turbine. If a decrease in load is demanded, the pressure set point is raised which causes the turbine control valve to move toward the closed position.

3.5.3.3 System Control

Control signals between the reactor and the turbine provide two functions required for normal operation. A signal from the initial pressure regulator is provided to the turbine
admission valves to maintain a nearly constant reactor pressure. A signal from the speed-load governing mechanism to the master flow controller establishes the necessary reactor recirculation flow required to meet the system power requirements.

If, while under normal load, the turbine speed decreases or the speed-load changer setting is increased, a positive speed-load signal is transmitted to the initial pressure regulator and the master flow controller. The increase in signal causes a momentary decrease in the pressure setting of the initial pressure regulator and causes the master controller to increase the flow demand to the recirculation system flow valve controller.

Decreasing the pressure setting of the initial pressure regulator causes a signal to be sent to the turbine admission valves, instructing them to open rapidly by an amount and for a length of time, which is a function of the speed-load error. This gives a limited rapid initial response to a speed-load error by increasing the steam flow from the reactor vessel. The allowable duration of this transient increased steam flow is limited by the fact that increased steam flow tends to reduce the reactor pressure and power level.

The increased flow demand to the recirculation system flow valve controller causes the flow control valve to open wider, causing an increase in reactor recirculation flow. The increased flow increases the reactor power output by sweeping out steam bubbles from the core faster, thus raising the effective density of the moderator. The increased steaming rate causes a slight increase in reactor pressure. The increase of pressure is sensed by the initial pressure regulator, which sends a signal instructing the turbine admission valves to open sufficiently to increase the turbine output to a level that will cancel the speed-load error.

**Daily load following**: Essentially any practical daily load following profile can be followed. There are no restrictions due to spatial xenon oscillations. (Early ascent above 95% of rated power would be subject to xenon override considerations.) Power levels can be readily reduced to any level during daily load following, including the power level where the turbine generator is supplying only house loads. Automatic load following provides the capability to accept large changes in load demand at operating power levels. The change in load demand may be initiated at any power level or reactor water recirculation flow combination in the automatic flow control range. This region lies between 28% and 75% of rated power and core flow rate ranging from approximately 65% to approximately 68% (constant flow control valve setting) and between 40% and rated power at rated core flow. For load reduction demands that exceed the range of the automatic flow control system, the main steam bypass system provides additional capability up to the bypass system capacity. The reactor operator would then establish a new control rod configuration to match the new power demand. For load increase demands that exceed the range of the automatic flow control system (assuming reduced flow initially), the power level will rise to that level corresponding to rated core flow and remain there until the control rods can be adjusted to increase power up to the desired level. Step demands for up to 25% of the power at rated core flow is accommodated by automatic reactor water recirculation flow control.

**Automatic dispatch operation**: Automatic reactor water recirculation flow control in combination with ganged control rods allows full participation in an automatic dispatch system with the combined purpose of meeting tie line regulation, spinning reserve, grid load rejection, and daily load following requirements. During such operation, automatic reactor water recirculation flow control meets the rapid changes in load demand required by tie line regulation, while simultaneously providing margins for spinning reserve or grid load rejection. The unit operator would adjust control rods to preserve the desired automatic margins during the slower changes in base power level required by daily load following.
3.5.3.4 Turbine Bypass Valve

A fast response, modulating-type valve, controlled by the steam bypass pressure regulator system, is used to perform three basic functions. The primary function is to reduce the rate of rise of reactor pressure when the turbine admission valves are moved rapidly in the closing direction. To perform this function, the bypass valve needs about the same speed of response as the turbine admission valves to prevent a pressure-induced reactor scram from high neutron flux when the turbine load is suddenly reduced by partial or complete closure of the turbine admission valves.

The second function of the bypass valve is to control reactor pressure during startup of the turbine. This allows the reactor power level to be held constant while the turbine steam flow is varied as the turbine is brought up to speed under the control of its speed governor.

The third function of the bypass valve is to help control reactor pressure after the turbine has been tripped. It is used to discharge the decay heat to the condenser and to control the rate of cooling of the reactor system.

3.5.3.5 Pressure Relief Function

A pressure relief function is used to control large pressure transients. This system will operate safety/relief valves following closure of the main steam isolation valves or the sudden closure of the turbine admission or stop valves and failure of the turbine bypass system to relieve the excess pressure. For this function, the safety/relief valves discharge steam from the steam lines inside the drywell to the suppression chamber. Each safety/relief valve is operated from its own overpressure signal for the relief function, and by direct spring action for the safety function.

To limit the cycling of safety/relief valves to one valve subsequent to their initial actuation during a main steam line isolation event, two valves (one a backup to the other) have the feature of automatically changing normal set pressures (opening and closing) following their initial actuation at normal set pressures to a lower level, thereby limiting the pressure cycles to a level where the other relief valves will not reopen. In conjunction with these two valves, the set pressure for the closing of the other valves is changed automatically which allows for them to stay open longer before closing to accommodate pressure swings. Manual valve operation and resetting of valve set pressure to their normal levels following the transient is by the control room operator.

3.5.3.6 Reactor Feedwater Control System

The reactor feedwater control system automatically controls the flow of reactor feedwater into the reactor vessel to maintain the water in the vessel within predetermined levels during all modes of plant operation. The control system utilizes signals from reactor vessel water level, steam flow, and feedwater flow.

The reactor feedwater control system provides the signal for the reduction of reactor water recirculation flow to accommodate reduced feedwater flow caused by failure of a single feedwater pump.

3.5.4 Plant Shutdown

For normal plant shutdown, reactor power and plant output are reduced by manual insertion of control rods. After turbine load is reduced to a minimum value, steam flow is maintained through the bypass valve and the generator is disconnected from the system.
Reactor power is further reduced to a low level, and the decay heat is rejected to the condenser through the turbine bypass valve. If the reactor is to be kept in the hot standby or steam condensing condition, criticality is maintained but fission power is reduced to a low level (about 0.01% of rated power is sufficient to maintain operating temperature). If refueling or other functions requiring access to the vessel are planned, all control rods are inserted and the reactor is cooled down by release of steam to the main condenser. The rate of cool down is normally controlled by periodically lowering the setting of the initial pressure regulator. After vessel gage pressure has been reduced sufficiently (1135 psi (930 kPa), the heat sink can be switched from the main condenser to the RHR system heat exchangers to get the reactor to the cold shutdown condition.

Reactor power is monitored from the source range up through the power operating range by suitable neutron monitoring channels, with all detectors inside the reactor core. This location of detectors provides maximum sensitivity to control rod movement during the startup period and provides optimum monitoring in the intermediate and power ranges. Three types of neutron monitoring are used: source range counting, intermediate range; voltage variance method, local power range; and DC ion chambers. A traversing in-core probe system provides for periodic calibration of the neutron detectors.

3.5.4.1 Source Range Monitor

In the source range, the neutron flux is monitored by fission counters, which are inserted to about the mid-plane of the core by the drive mechanisms, which move each chamber into the core through inverted thimbles. A range from below the source level to $10^9$ nv is covered.

As startup progresses and the count rate approach the top of the meter range (about $10^6$ cps), the counters are withdrawn downward to give a drop in apparent count rate. Criticality normally occurs before movement of the counters is necessary. The counters can be motor driven to any position within their limits of travel; however, two or three selected positions will provide the necessary range to achieve criticality and provide overlap with the intermediate range monitors.

When the reactor reaches the power range, the counters are moved to a position approximately 2 ft (0.61 m) below the core. This places the counters in a low neutron flux so that burnup and activation of the counters are minimized.

3.5.4.2 Intermediate Range Monitors

The intermediate range is from about $10^8$ to $1.5 \times 10^{13}$ nv. In this range, the neutron flux is monitored by a system using a voltage variance method (also known as mean square value or Campbell method). This method makes use of the AC component of voltage, which is due to the random nature of neutron pulses generated in a detection chamber. With small chambers located in the high temperature ambient of the reactor core, the AC component is used to measure neutron flux at lower power levels because cable leakage and gamma radiation have relatively little effect on the signal.

These fission chambers are also withdrawn during full power operation to maintain their expected life and to reduce activation. They are positioned with drive mechanisms similar to those used for the source range fission counters.
3.5.4.3 Local Power Range Monitor

In the power range, neutron flux is monitored by fixed in-core ion chambers, which are arranged in a uniform pattern throughout the core. These chambers cover a range of about 1%–125% of rated power on a linear scale. When a control rod or group of control rods is selected for movement, the readings from the adjacent detectors are displayed on the operator's control bench board together with a display of the position of the rod or group of rods.

Detector assemblies each contain four fission chambers and a calibration guide tube for a traversing ion chamber. The chambers are uniformly spaced in an axial direction and lie in four horizontal planes. Each ion chamber is connected to a DC amplifier with a linear output. Internal controls permit adjustment of the amplifier gain to compensate for the reduction of chamber sensitivity caused by burnup of its fissionable material. These detectors are individually replaced through the bottom of the reactor vessel.

The calibration guide tube included in each fixed in-core assembly permits the insertion of a traversing ion chamber to obtain vertical flux profiles and to calibrate the chambers. Each calibration guide tube extends nearly to the top of the active portion of the core and is sealed at the upper end. The tubes pass through the nozzles and seals beneath the reactor vessel and connect to an indexing mechanism located inside the containment. The indexing mechanism permits the traversing ion chamber to be directed to many different detector assemblies. Fully inserting the traversing ion chamber into one of the calibration guide tubes and then taking data as the chamber is withdrawn obtains flux readings along the axial length of the core. The data goes directly to the computer. One traversing chamber and its associated drive mechanism is used for each group of seven to nine fixed in-core assemblies (depending on reactor size).

3.5.4.4 Average Power Range Monitor

Four average power range monitors (APRMs) measure the average power level. Each monitor measures bulk power in the core by averaging signals from as many as 24 detectors distributed throughout the core. The output signals from these monitors are displayed and are also used to operate trips in the reactor protection system.

3.5.5 Nuclear System Protection System

The nuclear system protection system is a four-channel electrical alarm and actuating system that monitors the operation of the reactor, which, upon sensing an abnormal condition, initiates action to prevent an unsafe or potentially unsafe condition. The system integrates the following functions:

- **Reactor trip**: Monitors reactor operation and shuts down the reactor when certain limits are exceeded
- **Nuclear system isolation**: Isolates the reactor vessel and all connections of the primary pressure boundary that penetrate the containment barrier
- **Engineered safety feature actuation**: Actuates engineered safety feature systems

The nuclear system protection system uses *solid-state* electronic technology from sensor output to actuation device inputs, which include sensors, signal conditioning, and combinational logic and actuator logic. The system provides for the analog indication of major variables, separation of divisions, and online testability.
Logic bases for the nuclear system protection system functions are as follows:

- Reactor trip initiation for automatic control and reactor shutdown is based on a two-out-of-four logic.
- Nuclear system isolation by isolation valve closure in process lines penetrating the containment barrier is based on two-out-of-four logic for main steam isolation valves and a one-out-of-two taken twice logic for remainder of nuclear system isolation function.
- Engineered safety feature systems initiation is based on a one-out-of-two taken twice logic.

Sensors can be analog (such as process control transmitters) or digital (such as pressure switches or limit switches). Analog inputs for important variables drive indicators, which allow the operator not only to see the absolute value but also to compare readings in different channels. Both analog and digital signals are modified, if necessary, in signal conditioners to signals that are compatible with the solid-state logic. After conditioning, the digital signals go directly to the decision logic. Each conditioned analog signal is compared with the output of a set point generator in a bistable trip unit. When the preset level is exceeded, the bistable puts out a signal to the decision logic.

The decision logic is made up of solid-state circuitry that compares with various inputs. When a combination of inputs requires action, the logic circuitry provides a signal that seals into turn on a solid-state power gate that operates activation devices, such as contactors, circuit breakers, and solenoid pilot valves. Actuation devices in turn control power to the motors that operate valves and drive pumps, or control the air supply to pneumatically operated valves.

Simultaneous open and close manual switch conflicts are prevented by exclusive OR logic. Manual inputs may be momentary or maintained. The identity of the most recent momentary input is retained. When a maintained manual input is removed, the input channel reverts to automatic status.

Upon loss of AC power functions, which are normally energized (such as the reactor trip function) will provide fail-safe trip action. For such functions, loss of power to a sensor, its channel, or associated logic automatically produces a trip output. For normally de-energized functions (such as emergency core cooling) loss of power to a sensor, its channel, or associated logic leaves the state of the actuated equipment unchanged. Subsequent restoration of power will not introduce transients that could cause a change of state in the actuated equipment.

### 3.5.5.1 Reactor Trip Function

The nuclear system protection system initiates the rapid insertion of the control rods to shut down the reactor. The system is of the fail-safe design where it will trip on loss of electrical power but will not trip and cause a scram on the loss of a single power source. The four trip channels are physically separated from each other and from other equipment precluding the possibility of interactions that could cause possible false scrams or failure to scram. The logic requires a manual reset by the operator, which is automatically inhibited for 10 s. One reset switch is used for each trip channel. Failure of a single trip channel, division logic, or a system component will not prevent the normal protective action of the nuclear system protection system.
3.5.5.2 Nuclear System Isolation Function

The nuclear system protection system provides for the closing of valves to isolate the containment, thereby preventing the release of steam and process fluids. The logic and equipment required for those valves, which are required to be open during ECC are part of each of the separate ECCSs.

The lines, which penetrate the containment and are required, to be isolated during emergency core cooling consist of three groups:

- **Reactor coolant pressure boundary isolation**: These are lines that connect directly to the reactor vessel and penetrate the drywell and containment barrier.
- **Containment isolation**: These are lines that do not connect to the reactor vessel but penetrate the drywell and containment atmosphere.
- **Closed system isolation**: These are the lines that penetrate the containment. However, they are neither part of the reactor coolant pressure boundary nor are they connected directly to the containment atmosphere.

All isolation valves except nontestable check valves are capable of remote manual control from the control room. Automatic closure signals override manual control signals. Once isolation has been initiated, valves close fully and will not reopen automatically when the signal clears. Valve position (except nontestable check valves) is indicated in the control room.

Power and control systems associated with containment isolation are multichannel, failsafe. Failure of a single sensor circuit or system component does not prevent normal protective action. Separate routes from different, reliable power sources feed two valves in the same line. Control power and motive power for an electrically operated valve are supplied from the same source.

3.5.5.3 Engineered Safety Features Actuation Function

The engineered safety features include the ECCS (HPCS system, LPCS system, LPCI function of the RHR system, and the automatic depressurization function of the nuclear boiler system) and the RCIC systems.

3.5.5.4 Divisional Separation

Four divisional separations are used for reactor trip, isolation, and ECC inputs and outputs, both physically and electrically. Physical separation divisions are established by their relationship to the reactor vessel, which is divided into four quadrants. The sensors, logic, and output of the various systems are allocated to divisions.

Connections between divisions are isolated optically at the output of the originating cabinet or panel and buffered electrically at the receiving cabinet or panel. Connections to external devices, such as annunciators, indicators, and the computer, are similarly isolated and buffered.

3.5.5.5 Power Distribution

AC and DC power are required for the nuclear system protection system. Power distribution is divided into four divisions.
An inverter supplied from either the AC emergency bus or the DC battery bus provides AC power for the scram solenoid pilot valves and the main steam isolation valves.

### 3.5.5.6 Reset and Annunciation

A momentary trip of any channel is annunciated and causes that channel to lock out until manually reset. Sufficient annunciation trip signals are used so that the operator can determine the particular sensor or sensors, which caused the channel trip. The computer also prints out the identification of sensors, which have caused scram and, if several variables are involved, it prints out the sequence of events in which they occurred.

### 3.5.5.7 Backup Protection

Two three-way normally de-energized solenoid valves are used to remove the main instrument air supply from all scram valves. If any of the scram pilot valves failed to operate properly during scram, then the associated control rods would be scrambled by the loss of air supply due to operation of the back-up scram valve.

Conditions monitored and inputs that activate the nuclear system protection system are as follows:

- **High pressure in the drywell**: Abnormal drywell pressure trips the reactor, initiates the automatic depressurization function, the HPCS system, the LPCS system, and the RHR system.
- **Low water level in the reactor vessel**: A low water level in the reactor vessel trips the reactor, causes nuclear system isolation, activates the automatic depressurization function, initiates the HPCS and LPCS systems, and initiates the RCIC system.
- **High pressure in the reactor vessel**: High pressure in the reactor vessel will trip the reactor and initiate automatic depressurization function.
- **High neutron flux** will cause a reactor trip.
- **High water level in the scram discharge volume**: High water level in the control rod drive scram discharge volume will cause a reactor trip.
- **Turbine stop valve closure and turbine control valve fast closure**: Turbine stop valve closure and fast closure of the turbine control valve will cause a reactor trip.
- **Main steam line isolation**: The closure of the main steam line isolation valves will cause a reactor trip.
- **High radiation activity near main steam line**: High radiation levels near the main steam lines will cause a reactor trip and nuclear system isolation.
- **Leak detection**: Excessive leakage will cause nuclear system isolation.
- **Low pressure at the turbine inlet**: Low pressure at the turbine inlet will cause nuclear system isolation.

### 3.5.5.8 Bypass and Interlocks

An operation mode switch on the reactor control panel controls the interlocking and bypassing of the protection system for the various operational modes. Following are the modes and interlocks provided.
**Shutdown**: This mode is for use when the reactor is to be shut down and maintenance work performed. All rods must be fully inserted and none can be withdrawn.

**Refuel**: This mode is used during refueling operations. It allows a single control rod to be withdrawn for test purposes.

**Startup and standby**: This mode is for starting up the reactor and bringing it to a maximum of about 5% rated power. It also permits keeping the reactor critical while the turbine and associated equipment are being serviced with the main steam line isolation valves closed.

**Run**: This mode is for normal operation. The intermediate range flux scram is bypassed and all other function bypasses are removed. However, bypassing of some individual instruments for maintenance may be accomplished where permitted by operating procedures.

Interlocks are used on the intermediate range neutron monitors to ensure that all units are operating properly and on the proper range. Control rod withdrawal is blocked if the ratio of reactor power to recirculation flow exceeds a predetermined value.

### 3.5.6 Rod Control and Information

The primary purpose of the rod control and information (RC&I) function is to effect control rod motion as requested by the operator. It displays all information, which is relevant to the movement of rods. In addition to enabling the operator to move rods, this function also enforces adherence to operating restrictions, which limit the consequences of a potential rod drop accident. At higher power levels, it limits rod movement so that rods cannot be withdrawn to the point of generating excessive heat flux in the fuel. Unit conditions are considered in determining which restrictions are applied to a given rod movement request.

Rod position is sensed by a series of sealed glass reed switches contained within a tube inside the drive piston. Two switches are spaced every 3 in. (76 mm) with each of the dual switches feeding a separate channel. These signals are multiplexed inside the containment and transmitted to the control room. The rod position information function decodes these data and makes them available to other parts of the RC&I function, to the process computer and to the operator. The detection of an invalid input caused by a failed reed switch is indicated. The status of the scram valves and accumulators on the hydraulic control unit is monitored, and these data are available to the operator and the computer.

The speed and capacity of the RC&I function permit the control of more than one rod at a time. Up to four rods can be operated simultaneously. The position of each rod in a gang is monitored.