Nuclear Engineering Handbook

Boiling Water Reactor

3

Boiling Water Reactors (BWRs)

Kevin Theriault
GE-Hitachi Nuclear Energy

CONTENTS

3.1 Introduction

3.1.1 Boiling Water Reactor (BWR) Background

The BWR 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 Boiling Water Reactors (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 1 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

TABLE 3.1
Evolution of the GE BWR

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

respective buildings or support systems. Power output capabilities range from approximately 600 MWe to 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.
- Smaller-diameter fuel rods, longer in active fuel length and arranged in 8 by 8 bundles within the same external outline as the previous 7 by 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.

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

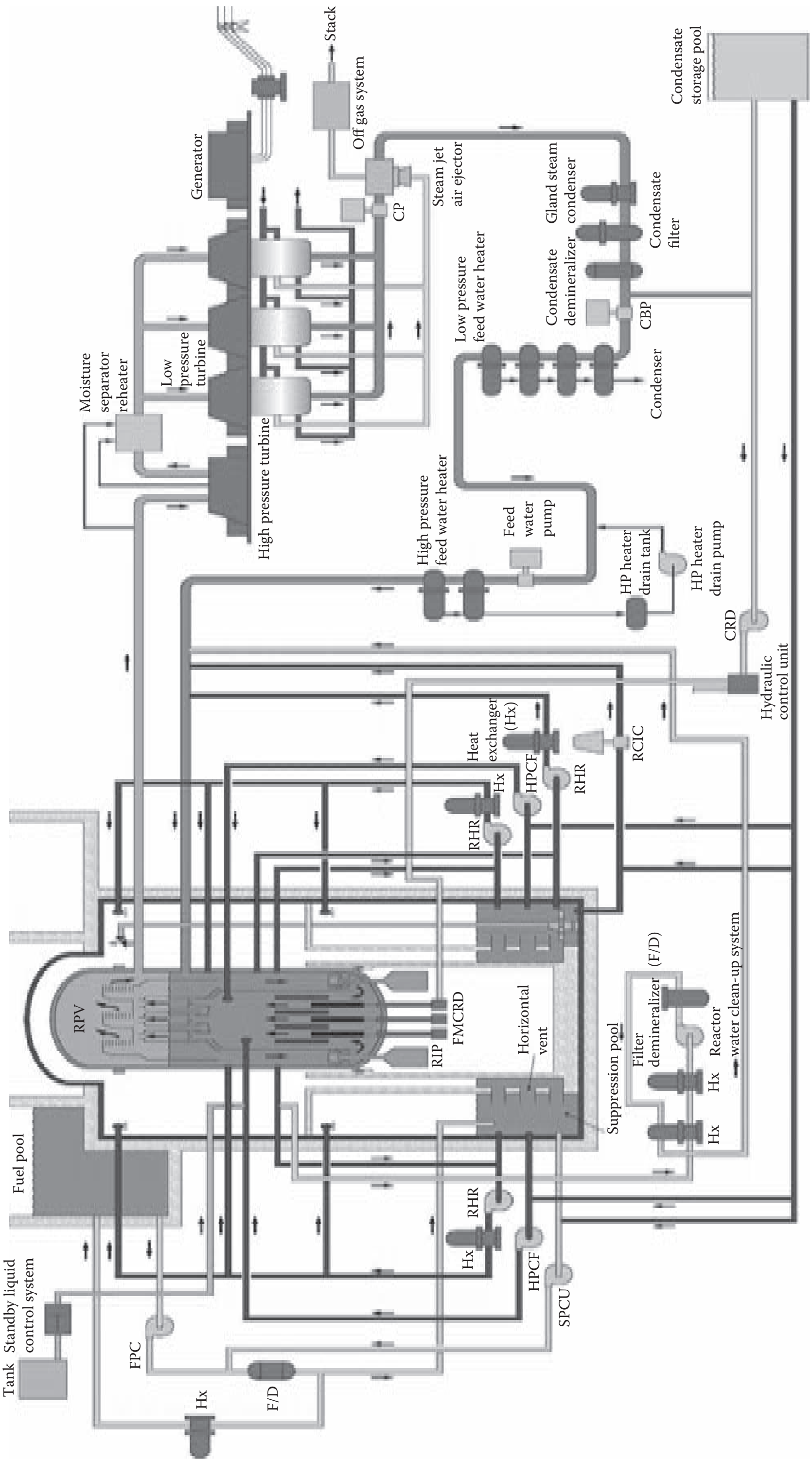
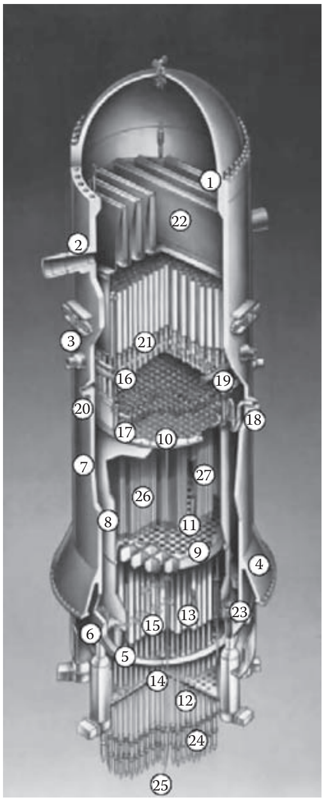


FIGURE 3.1
ABWR major systems.



1 - Vessel flange and closure head
2 - Stem outlet flow restrictor
3 - Feedwater nozzle
4 - Vessel support skirt
5 - Vessel bottom head
6 - RIP penetrations
7 - Forged shell rings
8 - Core shroud
9 - Core plate
10 - Top guide
11 - Fuel supports
12 - Control rod drive housings
13 - Control rod guide tubes
14 - In-core housing
15 - In-core instrument guide tubes
16 - Feedwater sparger
17 - High pressure core flooder (HPCF) sparger
18 - HPCF coupling
19 - Low pressure flooder (LPFL)
20 - Shutdown cooling outlet
21 - Shroud head and steam separator assembly
22 - Steam dryer assembly
23 - Reactor internal pumps (RIP)
24 - Fine-motion control rod drives
25 - Local power range monitor
26 - Fuel assemblies
27 - Control rods

FIGURE 3.2
ABWR reactor assembly.

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 reducing the capital cost and incorporating 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 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
- 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 $<10^{-6}$/yr)
- Radwaste generation <100 m^3/year
- Construction schedule of 48 months

3.1.4 Economic Simplified Boiling Water Reactor (ESBWR)

The ESBWR builds on the very successful ABWR technology and construction programs, as well as the Simplified Boiling Water Reactor (SBWR) development program (Figures 3.3 and 3.4) As of this publication production the ESBWR is being certified by the U.S. Nuclear Regulatory Commission (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 $<l0^6$/yr)
- Radwaste generation less than that of the 10% best operating BWRs
- Construction schedule of 48 months
- 20% reduction in capital cost ($/kWh) vs. 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 Boiling Water Reactor System (Figure 3.5) is a steam generation and steam utilization system consisting of a nuclear core located inside a reactor vessel and a

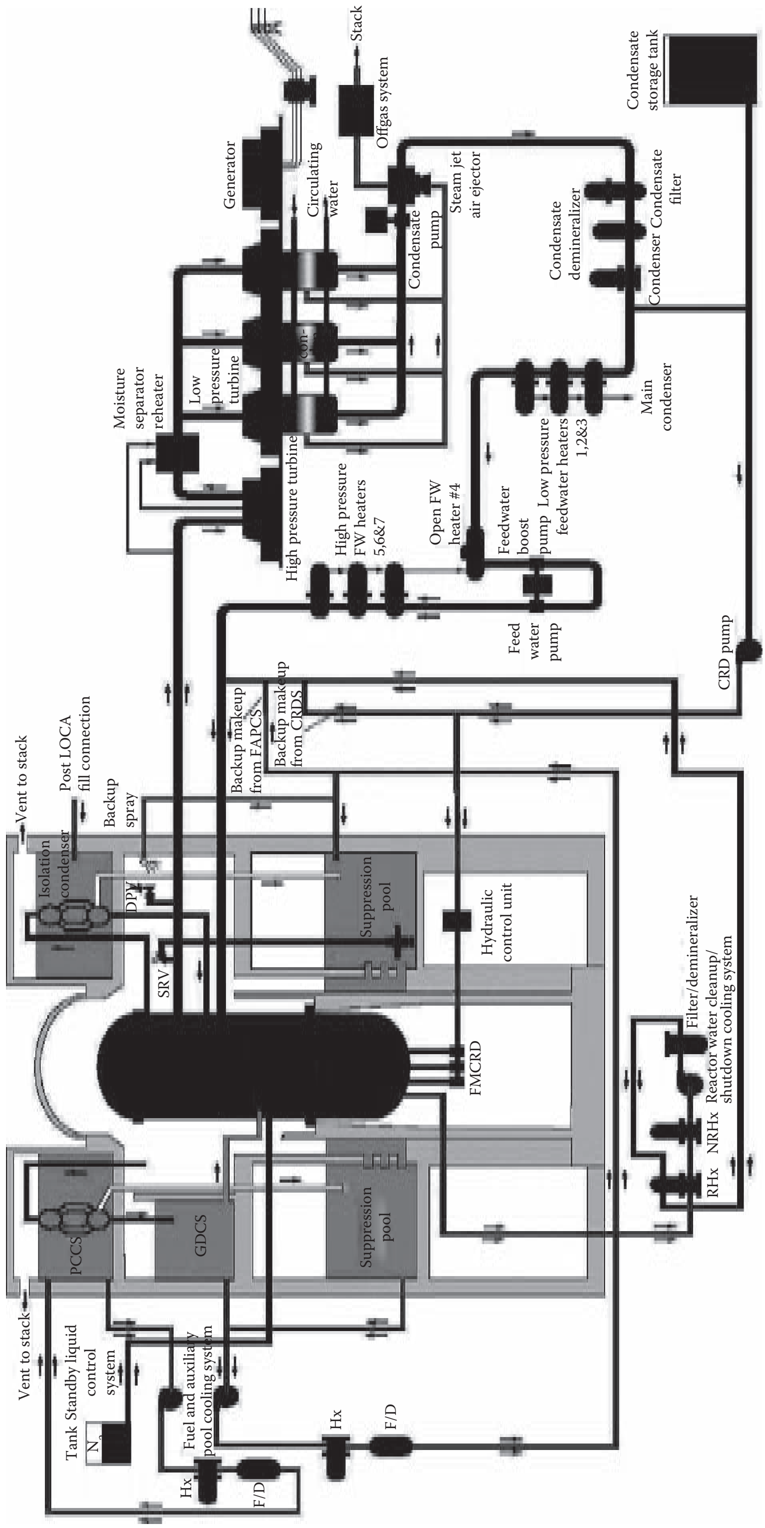


FIGURE 3.3
ESBWR major systems.

FIGURE 3.4
ESBWR reactor pressure vessel and internals.

conventional turbine-generator and feed water 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 seconds) 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.

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 feet (4.9 m) in diameter and 14-feet (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

TABLE 3.2

Comparison of Key ESBWR Features to Previous BWRs

Feature	**BWR/6**	**ABWR**	**ESBWR**
Recirculation system inside RPV	Two external loop recirculation system with jet pumps	Vessel-mounted reactor internal pumps	Natural circulation
Control rod drives	Locking piston CRDs	Fine-motion CRDs	Fine-motion CRDs
ECCS	2-division ECCS plus HPCS	3-division ECCS	4-division, passive, gravity-driven
Reactor vessel	Welded plate	Extensive use of forged rings	Extensive use of forged rings
Primary containment	Mark III – large, low pressure, not inerted	Compact, inerted	Compact, inerted
Isolation makeup water	RCIC	RCIC	Isolation condensers, passive
Shutdown heat removal	2-division RHR	3-division RHR	Nonsafety system combined with RWCU
Containment heat removal	2-division RHR	3-division RHR	Passive
Emergency AC	Three safety-grade D/G	Three safety-grade D/G	Two nonsafety D/G
Alternate shutdown	Two SLC pumps	Two SLC pumps	Two SLE accumulators
Control & instrumentation	Analog, hardwired, single channel	Digital, multiplexed, fiber optics, multiple channel	Digital, multiplexed, fiber optics, multiple channel
In-core monitor calibration	TIP system	A-TIP system	Gamma thermometers
Control room	System-based	Operator task-based	Operator task-based
Severe accident mitigation	Not specifically addressed	Inerting, drywell flooding, containment venting	Inerting, drywell flooding, core catcher

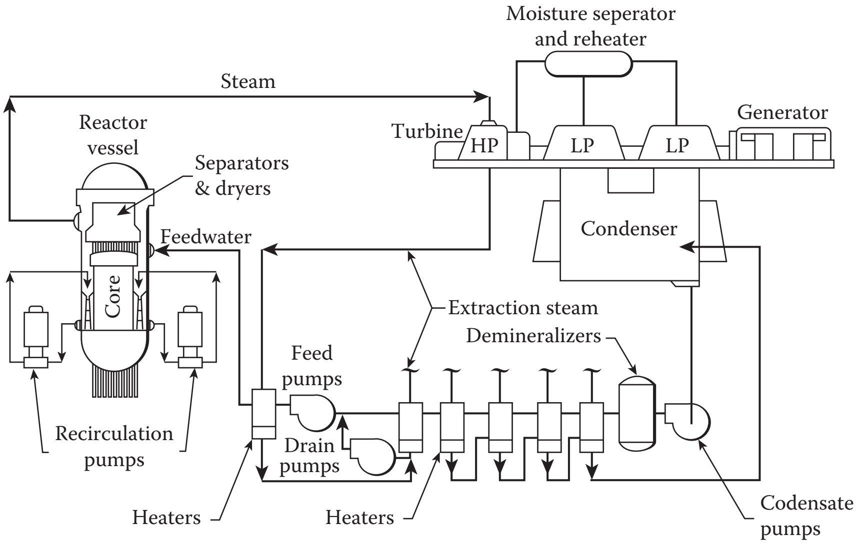


FIGURE 3.5
Direct Cycle Reactor System.

the core without changing control rod position. This unique BWR feature helps achieve the superior load-following capability of the BWR.

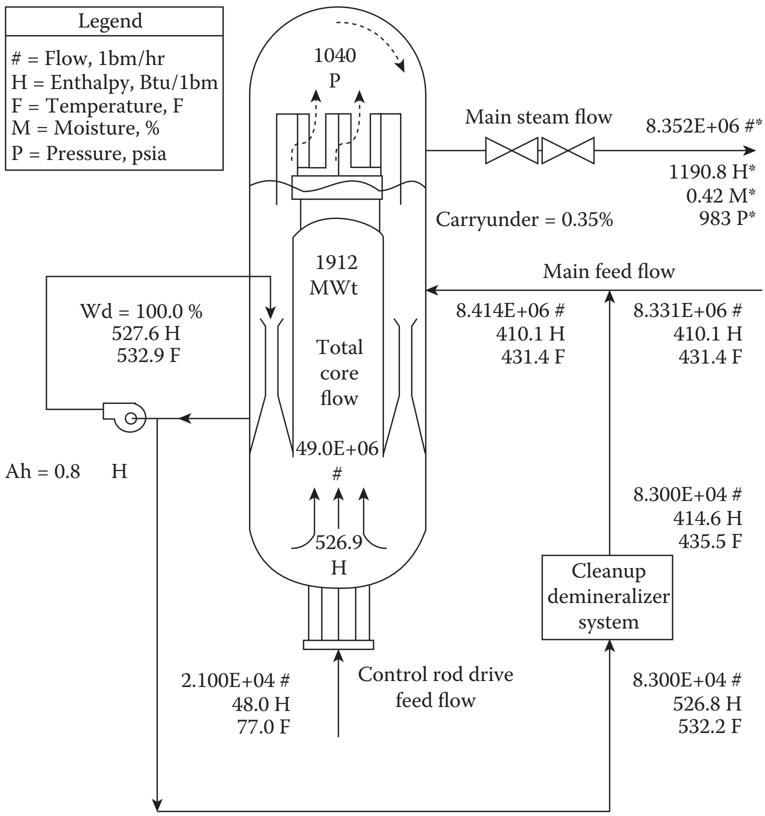
The BWR is the only light water reactor system that employs bottom-entry control rods. Bottom-entry and bottom-mounted control rod drives allow refueling without removal of control rods and drives, and allow drive testing with an open vessel prior to initial fuel loading or at each refueling operation.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



Core thermal power	1912.0
Pump heating	5.1
Cleanup losses	–2.7
Other system losses	–1.1
Turbine cycle use	1913.3 MWt

FIGURE 3.6
Typical heat balance diagram.

- 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:

- Reactor Vessel and Internals: Reactor pressure vessel, jet pumps for reactor water recirculation, steam separators and dryers, core spray, and feed-water 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

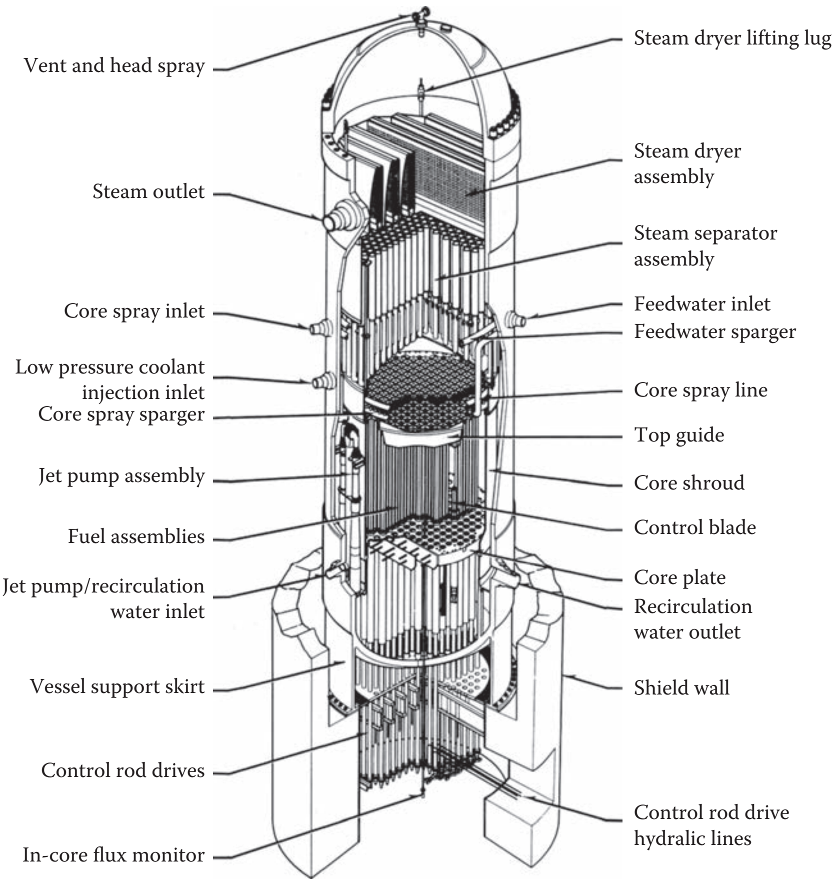


FIGURE 3.7
Reactor Assembly.

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 recalculating water and feed water 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 long hold down bolt which, 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 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 a 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 nosepiece. 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

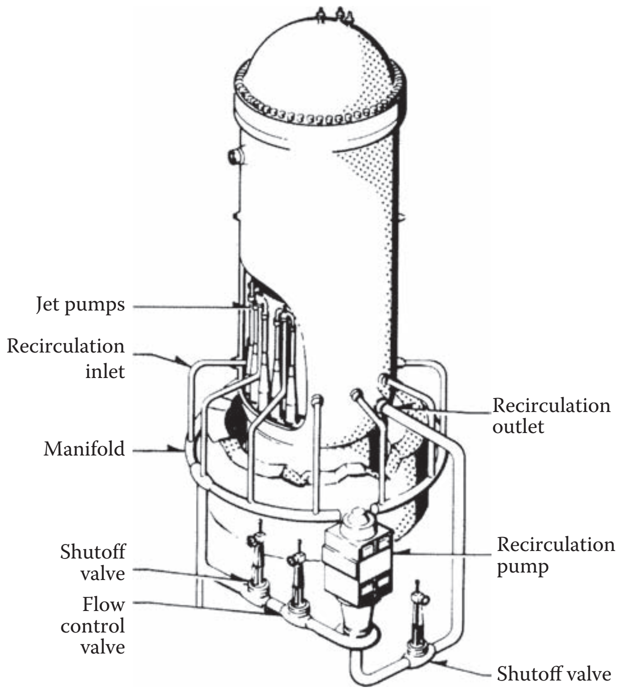


FIGURE 3.8
BWR vessel arrangement for jet pump recirculation system.

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. Feed water 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 flew 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 feet (5.8 m). Instrumentation monitors jet pump flow passages to ascertain their individual and collective flow rates under varying operating conditions.

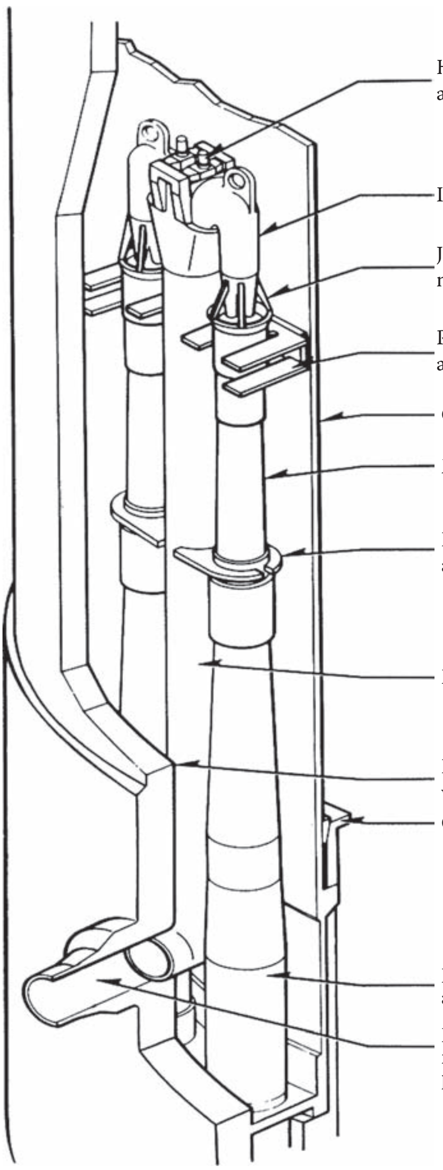


FIGURE 3.9
Jet pump assembly.

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 electro-hydraulic 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 from a forced recirculation mode to a natural circulation mode and eliminates the need for reactor recirculation pumps.

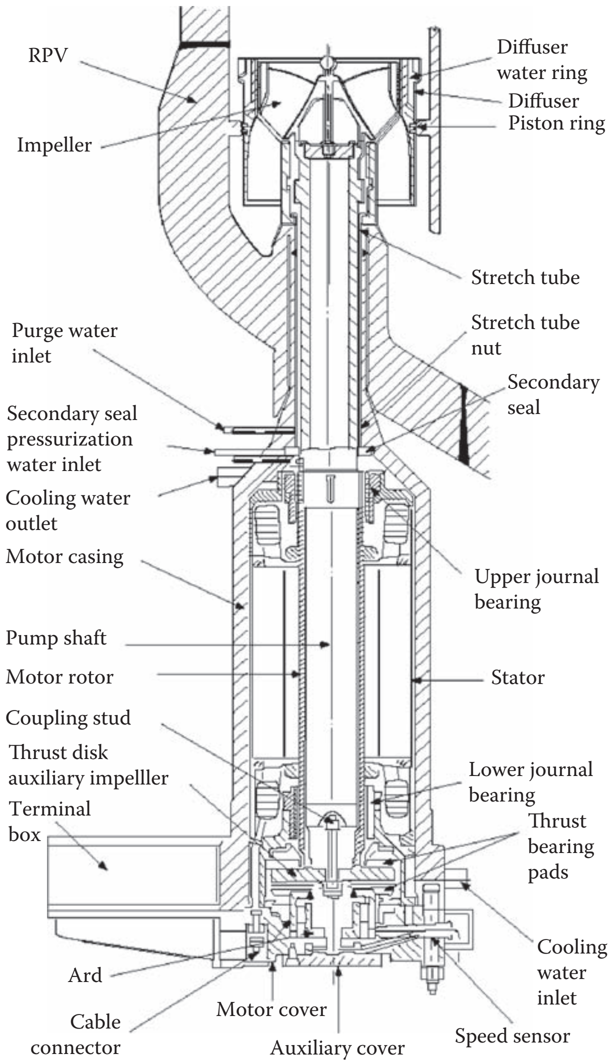


FIGURE 3.10
Cross section of reactor internal pump (RIP).

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 over-pressure 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 low pressure coolant injection 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 seconds to 120 seconds 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 operate each 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 in-flow 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 boiling water reactor.

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).

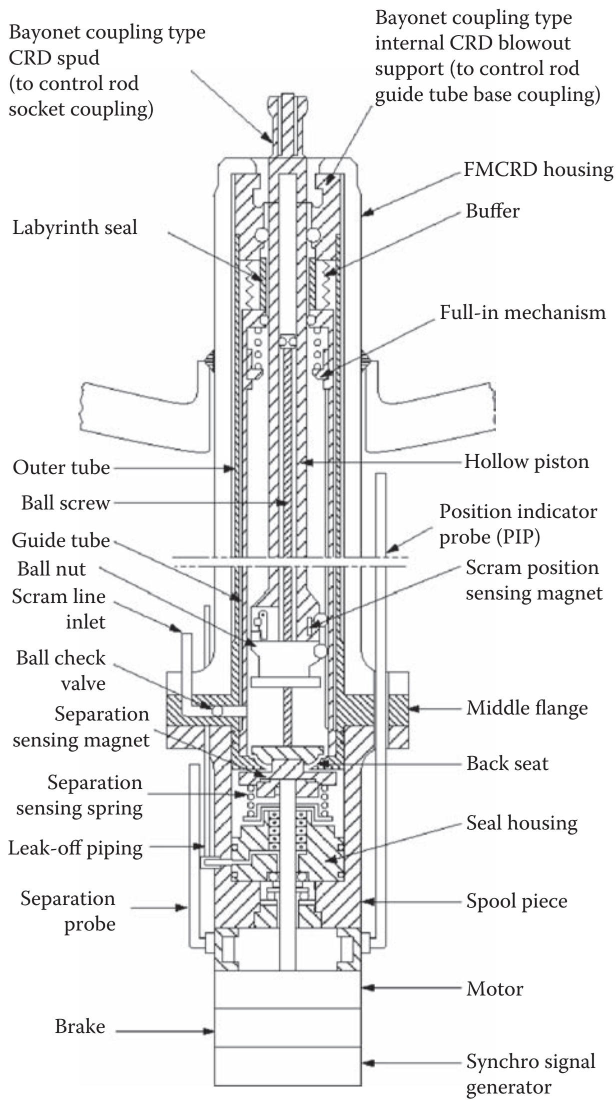


FIGURE 3.11
Fine-motion control rod drive cross-section.

3.3 Reactor Core Design

3.3.1 Introduction

The design of the boiling water reactor 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 levels 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 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.

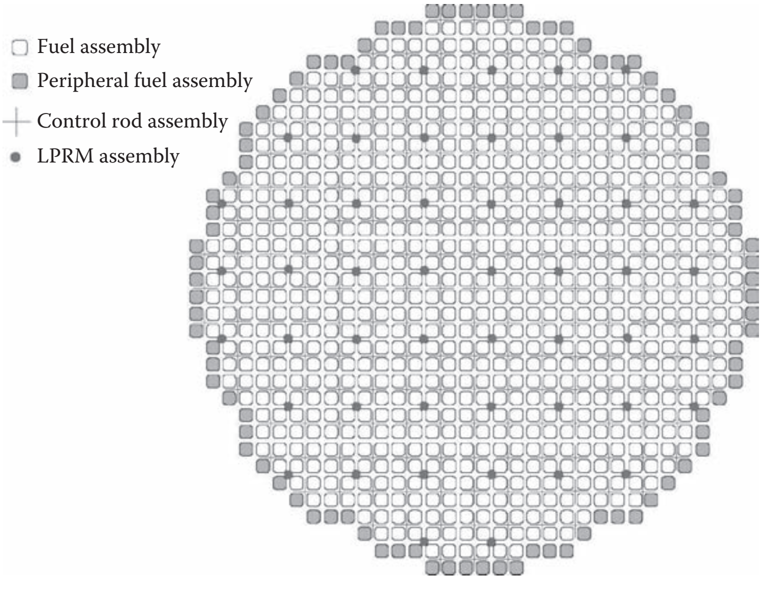


FIGURE 3.12
ABWR core configuration.

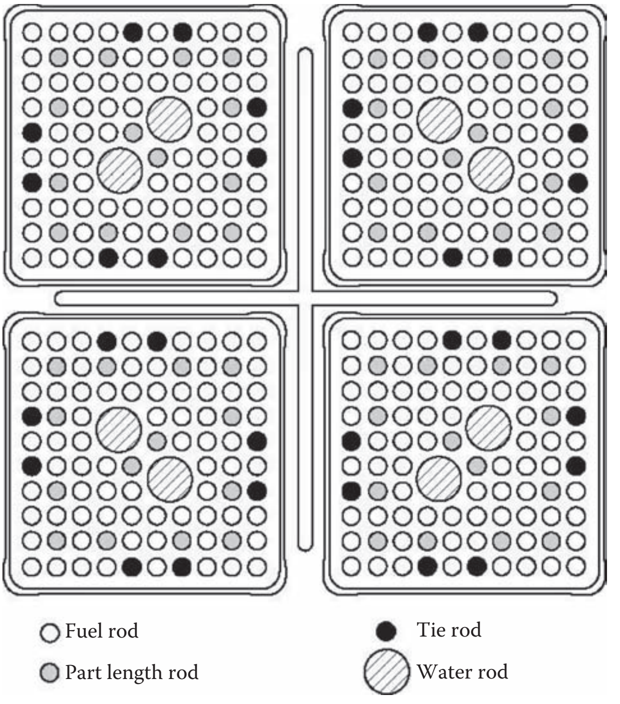


FIGURE 3.13
Fuel module (cell).

- 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 boiling water reactor 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.

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).

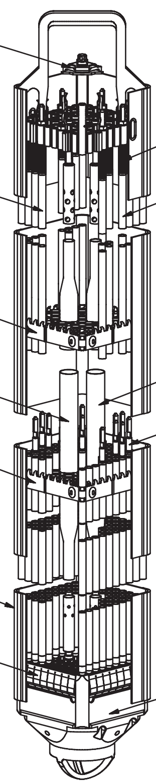


FIGURE 3.14
GE14 fuel assembly.

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 a pressure of 3 atmospheres, and sealed by welding Zircaloy end plugs in each end of the tube. The Zircaloy tube is 0.483 inch (12.3 mm) in diameter, 160-1/4 inches (4.07 m) long, with a 32-mil (0.81 mm) wall thickness. The pellets are stacked to an active height of 150 inches (3.8 m), with the top 9.5 inches (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.

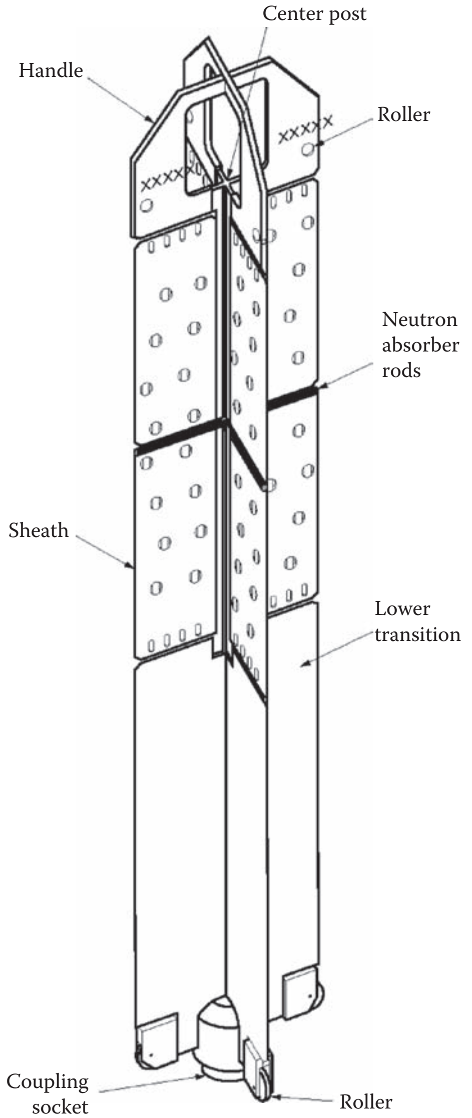


FIGURE 3.15
ABWR control rod.

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, i.e., tubes of Zr-2 cladding without UO_2 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 inches (138.6 mm) × 5.455 inches (138.8 mm) × 166.9 inches (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 boiling water reactor core because of greater steam voids in the center bundles of the care. 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 UO_2 Temperature. The maximum UO_2 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 bailing) 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, UO_2 pellet thermal expansion characteristics, and rate of UO_2 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, UO_2 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 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 sub-critical 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 UO_2 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 UO_2 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 7×7 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 8×8 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 boiling water reactor designs result in power coefficients well beyond the range of instability of xenon. This advantage of the boiling water reactor 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 boron carbide (B_4C) 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, B_4C 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 dk effective. The maximum worth of a rod in a typical power operation pattern will be about 0.01 dk effective. The notch increment dimensions and spacing of the rods are set to limit the reactivity insertion to about 0.0003 dk/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 Gd_2O_3 dispersed in a few selected fuel rods in each fuel assembly provides the desired characteristics. Gd_2O_3 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 RWCS
- 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 reactor core Isolation Cooling System (ISC), 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 Reactor Water Cleanup (RWC) System

The purpose of the reactor water cleanup 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 hours. 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 feed water line. Under normal operation, the water is removed at reactor temperature and pressure and pumped through regenerative and nonregenerative heat exchangers (NRHX) 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
- Standby liquid control 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-demineralizers, 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 (EECS)

The 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 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 cooled by ventilation 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 shut down 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 maintain the nuclear boiler in the standby condition in the event the vessel becomes isolated from the turbine steam condenser and from feed water makeup flow. The system also allows for complete plant shutdown under conditions of loss of the normal feed water 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 seconds 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 feed water 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 feed water 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-drive 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 minutes 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 minutes 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, 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 the 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 to the water in 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 back flow 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 residual heat removal 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-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 back flow 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 minutes 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 a 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 boiling water reactor 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 monitors 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 singe 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 cooldown 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 (SRM)

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 (IRMs).

When the reactor reaches the power range, the counters are moved to a position approximately 2 feet (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 IRM

The intermediate range is from about 10^8 to 1.5×10^{13} nv. In this range, the neutron flux is monitored by a system using a voltage variance method (also known as MSV 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 (LPRM)

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% to 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 burn up 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, 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 (APRM)

Four 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 on-line 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 seconds. 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 emergency core cooling systems.

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 multi-channel, fail-safety. 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 System.

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 scrammed 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:

- 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 arid none can be withdrawn.
- Refuel: This mode is for use during refueling operations. It allows a single control rod to be withdrawn for test purposes.
- Startup and Standby: This mode is for starting up of 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 (RC&I)

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 inches (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.

4

Heavy Water Reactors

Alistair I. Miller
Atomic Energy of Canada Limited

John Luxat
McMaster University, Hamilton

Edward G.Price
Atomic Energy of Canada Limited

Romney B. Duffey
Atomic Energy of Canada Limited

Paul J. Fehrenbach
Atomic Energy of Canada Limited

CONTENTS

4.1 Introduction

In the 1950s and 1960s, heavy water reactor (HWR) technology was explored in most of the countries investigating the application of nuclear fission to energy production. However, it was in Canada that this line of reactors was initially selected as the preferred type, which would become known as CANada Deuterium Uranium (CANDU). The choice was influenced by the early development work in Canada within the Manhattan Project, which took advantage of the superior characteristics of heavy water moderation for the production of plutonium. The attraction for ongoing development was mostly in the comparative simplicity of a system that did not depend on isotopic enrichment of uranium for the fuel. Further simplicity was introduced with the choice of pressure tubes (rather than a pressure vessel) to contain the operating pressure. The use of natural uranium and of pressure tubes make CANDU technology relatively easily accessible. Fuel manufacture has been successfully developed virtually everywhere where CANDUs have been built, and the dependence on very specialized large component fabricators to produce pressure vessels has been avoided.

This approach is possible because of heavy water's excellence as an economical moderator: the normal—mass 1 atom, often called "protium"—hydrogen in light water is more effective in reducing the energy of neutrons than is the heavier deuterium atom—the stable, mass 2 form of hydrogen—but it has a far lower propensity to absorb neutrons than protium. This low-absorption property—often called "good neutron economy"—permits a chain reaction with natural uranium. On account of needing more collisions to achieve moderation, HWR cores are bigger than those for light-water-moderated reactors. This has mixed benefits and disadvantages: it is a source of added costs but a larger source of natural cooling and the comparative difficulty of achieving criticality means that maintaining or achieving criticality in abnormal or unintended circumstances is less likely or impossible.

Light-water-moderated reactors must bear the cost of enriching all of their fuel in ^{235}U throughout their lives. Heavy water moderated reactors avoid this, but must bear the initial cost of producing heavy water. Once produced, however, only minor losses of heavy water occur (typically ≤1%/a as make-up). Uranium enrichment and heavy water production are isotope separations of comparable difficulty. The separation factors exploited in isotope separation are larger for deuterium and protium than for ^{235}U and ^{238}U, but this advantage is balanced by the relatively high natural abundance of ^{235}U (0.71%) compared with deuterium (~0.015%). Overall, the basic costs of light water reactors and HWRs appear to be very comparable.

HWRs are versatile. They can use natural uranium as a fuel, and their good neutron economy gives them superior capabilities to burn almost any fertile or fissile fuel. Some uranium enrichment—though less than for LWRs—may be advantageous. This possibility includes the ability to burn fuel discharged from LWRs, extending its energy output by around 30%. Thorium can be burned in various ways, at or close to sustained breeding. Plutonium and actinides can be consumed. India—which has abundant thorium resources—has pioneered in R&D to use thorium as the main fuel in CANDU-type reactors.

This chapter summarizes the more important aspects of nuclear reactors in which heavy water (D_2O) is used as the moderator. For a more comprehensive description of HWRs and associated references, the reader is referred to IAEA Technical Report Series No. 407, on which this chapter draws extensively.

4.2 Characteristics of HWRs

There are four types of HWRs that have been constructed and operated to produce electricity:

(a) Pressure tube heavy water cooled heavy water moderated reactor
(b) Pressure tube boiling light water, heavy water moderated reactor
(c) Pressure vessel heavy water cooled and heavy water moderated reactor
(d) Gas cooled heavy water moderated reactor

The dominant type of HWR is the heavy water cooled, heavy water moderated reactor as defined by the CANDU and Indian series of reactors. In the following section, the characteristics of the CANDU 6 reactor are used to describe the important features of these reactors.

Key temperature and pressure parameters of Atomic Energy of Canada Limited's (AECL) CANDU 6 and ACR-1000 reactors are summarized in Table 4.1.

TABLE 4.1

Key Temperature and Pressure Parameters of AECL's CANDU 6 and ACR-1000 Reactors

	CANDU-6	ACR-1000
Reactor outlet header pressure (MPa(g))	9.9	11.1
Reactor outlet header temperature (°C)	310	319
Reactor inlet header pressure (MPa(g))	11.2	12.5
Reactor inlet header temperature (°C)	260	275
Single channel flow (kg/s)	28	28
Number of channels	380	520
Steam generator temperature (°C)	260	275.5
Steam generator pressure (MPa(g))	4.6	5.9
Turbine inlet pressure (MPa(g))	4.4	5.7
Turbine inlet temperature (°C)	258	273
Turbine outlet pressure (kPa(a))	4.9	4.9
Turbine outlet temperature (°C)	32.5	32.5
Condenser pressure (kPa(a))	4.9	4.9

4.2.1 Pressure Tube Type HWRs

4.2.1.1 Introduction

The CANDU series of reactors is designed to use natural uranium, but it can also use SEU or a variety of fuels. Typically, the reactor core is contained in a cylindrical austenitic stainless steel tank (calandria) that holds the heavy water moderator at low temperatures (<80°C) and low pressure (~0.1 MPa). The ends of the cylinder are closed with two parallel end shields that are perforated with holes for the fuel channels, the holes being arranged in a square lattice pattern. Thin-walled Zircaloy-2 tubes are fastened to each inner tube sheet and act as stays for the end shields to form a leak tight tank. The holes in each end shield are connected with stainless steel tubes (lattice tubes) (Figure 2.1). Each fuel channel consists of a Zr–2.5%Nb pressure tube joined to martensitic stainless steel end fittings, and occupies the tubular holes or lattice sites formed by each combined lattice tube and calandria tube. The fuel channel end fittings are supported on a pair of sliding bearings at each end, and the pressure tube is supported and separated from the calandria tube by annular spacers Figure 2.1 illustrates the CANDU design.

4.2.1.2 Design and Operating Characteristics

The pressure tube, heavy water cooled, heavy water moderated reactor has certain characteristics which facilitate operation and safety analysis, and which provide fuel options. These are summarized in the following sections.

4.2.1.2.1 Pressure Tubes as the Reactor Pressure Boundary

Pressure tubes are thin-walled components with a simple geometry. This facilitates repetitive manufacture and inspection, both pre-service and in-service. Pressure tubes are replaceable and can be replaced at the end of their life to extend the reactor life, typically for 25–30 years.

As a result of the thin walls, there is no concern as regards overstressing the reactor pressure boundary under a fast cool-down (e.g., steam main break). A growing defect in a pressure tube will, in most cases, leak before the tube breaks.

A leak is detected through ingress of coolant to the annulus gas system, allowing time for a shutdown to replace the tube. Even if a pressure tube should fail, the damage is limited to the channel itself and some surrounding in-core components. Other channels will not fail.

The pressure tube geometry means that fuel element are always within a few centimetres of the moderator, which can act as an emergency heat sink for postulated severe accidents such as a loss of coolant accident (LOCA) combined with loss of emergency core cooling (LOECC). This also provides an inherent limit to metal–water reactions in a severe accident because the fuel bundle is close to the emergency heat sink.

The horizontal channel orientation means that "graceful" sagging occurs in the event of a beyond-design-basis severe core damage accident, i.e., assuming a LOCA with LOECC and loss of moderator cooling, the fuel channels would slump onto the bottom of the calandria, resulting in heat transfer to the water in the shield tank (at which point some melting may occur). Pressure tubes preclude the possibility of a sudden, large, high-pressure melt ejection occurring and eliminate one potential challenge to containment integrity.

There are no large high-pressure pipes directly connected to the reactor structure, so there are no overturning forces placed on the reactor from a large LOCA.

4.2.1.2.2 Fuel

Fuel characteristics are as follows:

- Use of natural uranium fuel allows the storage and handling of new fuel with minimal criticality concerns because the fuel bundles require heavy water to become critical.
- On-power fuelling means that there is very little reactivity hold-up needed in the reactor control system (and no need for boron in the coolant to hold down reactivity, resulting in a simpler design). The control rod reactivity worth can therefore be kept quite small (2 mk per rod or less).
- The high neutron economy, and hence low reactivity hold-up, of HWRs means that the reactor is very unlikely to become critical after any postulated beyond-design-basis severe core damage accident.
- Low remaining fissile content in spent fuel means that there are no criticality concerns in the spent fuel bay.

Fuel design is simple and performs well. Typically, the defect rate in operating CANDUs is <0.1% of all bundles (even smaller, of the order of 0.001%, in terms of fuel elements).

4.2.1.2.3 Fuelling Characteristics

On-power refuelling and a failed fuel detection system allow fuel that becomes defective in operation to be located and removed without shutting down the reactor. This reduces the radiation fields from released fission products, allows access to most of the containment while the reactor is operating, and reduces operator doses.

As a result of on-power fuelling, the core state does not change after about the first year of operation. Thus, the reactivity characteristics remain constant throughout plant life, resulting in simpler operation and analysis. Ability to couple tools to the fuelling machine allows it to be used for some inspections without necessitating removal of the pressure tube and in some instances without de-fuelling the channels.

4.2.1.2.4 Moderator Characteristics

The cool, low-pressure moderator removes 4.5% of the fuel heat during normal operation; about the same as the amount of decay heat removed shortly after shutdown. It can therefore act as a long-term emergency heat sink for a LOCA plus LOECC; the heat transfer is effective enough to prevent melting of the UO_2 fuel and preserve channel integrity.

The HWR has an inherent prompt shutdown mechanism (besides the engineered shutdown systems and the control system) for beyond-design-basis severe core damage accidents. If steam is introduced into the moderator as a result of, for example, multiple channel failures, then the immediate effect of loss of moderation would cause the reactor to shut down.

In the case of a channel failure, the moderator acts as an energy absorbing "cushion," preventing failure of the calandria vessel. Even for beyond-design-basis severe core damage accidents, where a number of channels are postulated to fail, the calandria may leak but would retain its gross structural integrity.

The low-pressure, low-temperature moderator contains the reactivity mechanisms and distributes the chemical trim, boron, for reactivity purposes and gadolinium nitrate for shutdown purposes.

4.2.1.2.5 Heat Transport System (HTS) Characteristics

Given the economic value of heavy water, the designers of HWRs pay great attention to preventing coolant leaks. Leak detection equipment is highly sensitive and leaks from any source can be detected very early. The Heal transport system (HTS) contains minimal chemical additives (only LiOH for pH control and H_2 to produce a reducing chemistry).

4.2.1.2.6 Tank Characteristics

The shield tank contains a large volume of water surrounding the calandria. In the case of beyond-design-basis accidents (BDBA), e.g., severe core damage accidents such as a LOCA plus LOECC plus loss of moderator heat removal plus failure of make-up to the moderator, the shield tank can provide water to the outside of the calandria shell, ensuring that it remains cool and therefore intact, thereby confining the damaged core material within the calandria. Recent HWR designs have added make-up to the shield tank and steam relief to ensure that this remains effective. Heat can be transferred from the debris through the thin-walled calandria shell to the shield tank without the debris melting through. This inherent "core catcher" provides debris retention and cooling functions. Because a severe core damage sequence can be stopped in the calandria, the challenge to containment is much reduced.

4.2.1.2.7 Reactivity Control Characteristics

HWRs using natural uranium have a positive void coefficient, which leads to positive power coefficients. This is accommodated in the design by employing independent fast-acting shutdown systems based on poison injection into the moderator and spring assisted shut-off rods.

The long prompt neutron lifetime (about 1 ms) means that for reactivity transients even above prompt critical, the rate of rise in power is relatively slow. For example, the reactor period for an insertion of 5 mk is about 0.85 s^{-1}, whereas for 7 mk it is about 2.4 s^{-1}. The shutdown systems are, of course, designed to preclude prompt criticality.

Separation of coolant and moderator and the slow time response of moderator temperature eliminates moderator temperature feedback effects on power transients. The only way of diluting moderator poison (if present) is through an in-core break, which is small and hence would have an effect that is slow relative to shutdown system capability.

Reactivity control mechanisms penetrate the low-pressure moderator, not the coolant pressure boundary. They are therefore not subject to pressure-assisted ejection in the event of an accident and can be relied upon to perform their function.

Bulk power and spatial control are fully automated with digital control and computerized monitoring of the plant state, which simplifies the job of the operator and reduces the chances of operator error.

Control is through adjusters and the shut-off rods. These are of simple design with relatively large tolerances (e.g., loose fit in guide tubes). They do not interact with the fuel bundles at all and are not, therefore, subject to jamming in the event of an accident damaging the fuel.

In the case of a severe accident (LOCA plus LOECC), the damaged fuel is confined to the fuel channels, and therefore there is no risk of melting the control rods.

4.2.1.2.8 Shutdown Cooling

HWRs have a shutdown cooling system that can remove decay heat after shutdown from full pressure and temperature conditions. It is not necessary to depressurize the HTS.

4.2.1.2.9 Safety Systems

The safe operation of a reactor necessitates that the fuel be kept adequately cool at all times to prevent loss of fuel cladding integrity and the consequent dispersion of radioactive species into the coolant. The safety systems that prevent or mitigate fuel damage are described below.

4.2.1.2.9.1 Systems that Shut Down the Reactor in the Case of Accidents The emergency core cooling system (ECCS) fulfils this purpose. It is a system that refills the reactor fuel channels with light water to remove residual or decay heat from the fuel. The fuel requires heavy water for the reactor to go critical and the light water of the ECCS suppresses criticality. There is no need to add boron to the ECCS water.

4.2.1.2.9.2 Systems that Prevent Release of Radioactivity into the Environment The major system fulfilling this function is the containment building. Current HWRs have a containment isolation system that has been demonstrated by on-power testing to have a probability of unavailability of <10-3 years/year. The building volumes are relatively large, resulting in low design pressures. Details of the operation of the safety systems are given in Section 4.4.

Most HWRs have two, independent, diverse, reliable, testable, redundant, fail-safe shutdown systems (as well as the control system). The two systems do not share instrumentation, logic actuation devices or in-core components. One system uses rods, the other liquid poison injection. Each of the shutdown systems is effective, by itself, for all design basis accidents. With each one demonstrated by on-power testing to a reliability of 999 times out of 1000 attempts, the risk of a transient or accident occurring without shutdown is negligible.

Each safety shutdown system has the ability during an accident to shut down the reactor from the most reactive state to zero power cold conditions. Moderator poison is only needed in the long-term (hours) to compensate for Xenon decay.

The positive void coefficient, while it must be compensated for in an accident by the shutdown systems, has the advantage of resulting in fast and responsive neutronic trips for a number of accidents. It also ensures an inherent power reduction for rapid cool-down accidents such as steam main failure.

Most HWRs have two sources of emergency electrical power: Group 1 Class III diesels and separate, independent, seismically qualified Group 2 Class III diesels. This greatly reduces the risk of station blackout.

4.2.1.2.10 Licensing

The HWR regulators' licensing philosophy usually places the onus on the proponent to demonstrate that the plant is safe while the regulator audits the result. The regulator does not prescribe the design in detail, thereby avoiding the conflict of interest inherent in reviewing its own design. Besides encouraging innovation, this process places full responsibility for safety on the organization that owns and operates the plant, consistent with IAEA recommendations.

HWR regulations typically specify the classes of accident to be considered in the design. These include not only failures of an operating system (e.g., LOCA), but also such failures combined with a failure of the mitigating system (e.g., LOCA plus LOECC, with credit for only one shutdown system in any accident). The latter are design basis accidents in HWRs and must meet dose limits using deterministic analysis. The requirement to include these "dual" failures means that the least unlikely severe accidents are within the design basis and must not cause severe core damage. This results in a robust design.

Although the list of "design basis" accidents is specified in part in regulations, the proponent is required to demonstrate that the analysis has covered a complete set. This ensures that the scope of analysis is comprehensive.

Regulatory requirements in most HWR jurisdictions imply the use of probabilistic safety assessment (PSA), not just after the design is complete, but very early on in the design phase, when any identified weaknesses can still be rectified relatively inexpensively.

4.2.1.3 Nuclear Steam Supply System (NSSS)

4.2.1.3.1 Introduction

The CANDU 6 is used as the basis for describing the features of the CANDU HWR. All CANDU 6 power plants are fundamentally identical, although there are differences in detail that largely result from different site conditions and from improvements made in the newer designs. The basic design features of the current generation of Indian 220-MWe HWRs and the 500-MWe versions are also generally similar except in some quantitative details.

The heat produced by controlled fission in the fuel is transferred to the pressurized heavy water coolant and circulated through the fuel channels and steam generators in a closed circuit. In the steam generators, the heat is used to produce light water steam. This steam is used to drive the turbine generator to produce electricity. The NSSS is illustrated in Figure 4.1.

4.2.1.3.2 Fuel and Fuel Handling System

The fuel handling system:

- Provides facilities for the storage and handling of new fuel
- Refuels the reactor remotely while it is operating at any level of power
- Transfers the irradiated fuel remotely from the reactor to the storage bay

The fuel-changing operation is based on the combined use of two remotely controlled fuelling machines, one operating at each end of a fuel channel. Either machine can load or receive fuel. New fuel bundles, from one fuelling machine, are inserted into a fuel channel in the same direction as the coolant flow—flow direction alternates between adjacent channels—and the displaced irradiated fuel bundles are received into the second fuelling machine at the other end of the fuel channel.

Typically, either four or eight of the 12 fuel bundles in a fuel channel are exchanged during a refuelling operation. In the case of a CANDU 6 (700 MWe) reactor, a mean of 10 natural uranium fuel channels are refuelled each week.

The fuelling machines receive new fuel while connected to the new fuel port and discharge irradiated fuel while connected to the discharge port. The entire operation is

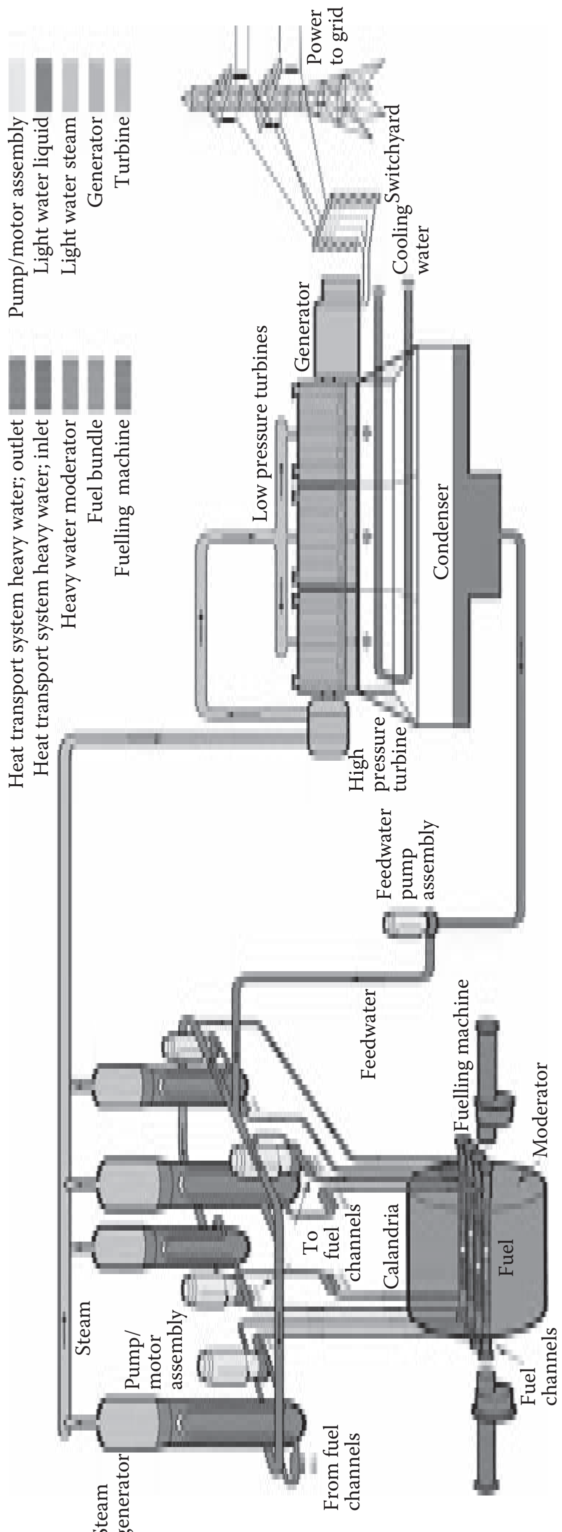


FIGURE 4.1
Nuclear steam supply system.

directed from the control room through a pre-programed computerized system. The control system provides a printed log of all operations and permits manual intervention by the operator. New fuel is received in the new fuel storage room located in the service building. This room accommodates six months' fuel inventory and can store, temporarily, all the fuel required for the initial fuel loading.

When required, fuel bundles are transferred to the new fuel transfer room located in the reactor building. Fuel bundles are identified and loaded manually into the magazines of the two new fuel ports. Transfer of new fuel bundles into the fuelling machines is remotely controlled.

Irradiated fuel received in the discharge port from the fuelling machine is transferred remotely onto an elevator that lowers it into a discharge bay filled with light water. The irradiated fuel is then conveyed, under water, through a transfer canal into a reception bay, where it is loaded onto storage trays or baskets and passed into the storage bay.

The discharge and transfer operations are remotely controlled by station staff. Operations in the storage bays are carried out under water, using special tools aided by cranes and hoists. Defective fuel is inserted into cans under water to limit the spread of contamination before transfer to the fuel bay. The storage capacity of the bays is sufficient to accommodate a minimum of 10 years' accumulation of irradiated fuel. Neither new nor irradiated CANDU fuel can achieve criticality in air or light water, regardless of the storage configuration. Thus, dry storage of fuel is possible after interim storage in the spent fuel bay. Safeguarding of fuel is facilitated by putting an identification number on each bundle. Numbers are recorded at various stages during fuel usage.

4.2.1.3.3 HTS

The primary heat transport system (PHTS) in a CANDU 6 unit consists of two loops arranged in a figure-of-eight configuration with the coolant making two passes in opposite directions through the core during each complete circuit. The two PHTS pumps in each loop operate in series, causing the coolant to transport the fission heat generated in the fuel to the steam generators where it is transferred to light water, producing steam to drive the turbine. Each loop has one inlet and one outlet header at each end of the reactor core. The coolant is fed to each of the fuel channels through individual feeder pipes and returned from each channel through individual feeder pipes to the outlet headers.

Other key features of the circuit are listed below.

- Steam generators consist of an inverted U-tube bundle housed within a cylindrical shell, usually of a lightbulb shape. The steam generators include an integral preheater on the secondary side of the U-tube outlet section, and integral steam separating equipment in the steam drum above the U-tube bundle.
- Heat transport pumps are centrifugal motor-driven pumps, mounted with the shaft vertical and with a single suction and double discharge.
- In the event of electrical power supply interruption, cooling of the reactor fuel is maintained for a short period of time by the rotational momentum of the heat transport pumps during reactor power rundown, and by natural convection after the pumps have stopped.
- Chemistry control is relatively simple because chemicals do not have to be added to the PHTS for reactivity control.
- Carbon steel piping, which is ductile and relatively easy to fabricate and to inspect, is used in the HTS. Low concentrations of chromium are nowadays added to the steel to prevent flow assisted corrosion from outlet water undersaturated in iron.

4.2.1.3.4 Heat Transport Auxiliary Systems

There are four auxiliary systems attached to the HTS, which are required to perform specific functions, as destailed below.

(a) Heat Transport Pressure and Inventory Control System provides:

- Pressure and inventory control for each HTS loop
- Overpressure protection
- A controlled degassing flow

The system consists of a pressurizer, D_2O feed pumps, feed and bleed valves, D_2O storage tank, degasser condenser, liquid relief valves, and safety valves.

The pressure in the PHTS of a CANDU 6 reactor is controlled by a pressurizer connected to the outlet headers at one end of the reactor. Pressure in the pressurizer is controlled by heaters in the pressurizer and by steam bleed. Heavy water in the pressurizer is heated electrically to pressurize the vapor space above the liquid. The volume of the vapor space is designed to cushion pressure transients, without allowing excessively high or low pressures to be generated in the HTS. (Nuclear power plants that do not allow the coolant to boil in the channels, do not use a pressurizer and rely on the feed and bleed system for control.)

The pressurizer also accommodates the change in volume of the reactor coolant occurring in the HTS when the reactor moves from zero power to full power. This permits the reactor power to be increased or decreased rapidly, without imposing a severe demand on the D_2O feed-and-bleed components of the system. The coolant inventory is adjusted by the feed-and-bleed circuit and, with the pressurizer isolated, pressure can also be controlled by this system when the reactor is at low power or when the reactor is shut down. This feed and bleed circuit is designed to accommodate the changes in coolant volume that take place during heat-up and cool-down.

(b) D_2O Collection System whose main purpose is to:

- Collect leakage from mechanical components
- Receive D_2O sampling flow
- Receive D_2O drained from equipment prior to maintenance

The collected D_2O is pumped from the collection tank to the storage tanks of the pressure and inventory control system for reuse in the HTS. However, if the isotopic purity of the collection tank contents is low, the D_2O can be pumped into drums for upgrading.

(c) The Shutdown Cooling System, that is capable of:

- Cooling the HTS from 177°C down to 54°C, and holding the system at that temperature indefinitely
- Providing core cooling during maintenance work on the steam generators and heat transport pumps when the HTS is drained down to the level of the headers
- Being put into operation with the HTS at full temperature and pressure

The shutdown cooling system consists of two independent circuits, one located at each end of the reactor. Each circuit consists of a pump and a heat exchanger, connected between the inlet and outlet headers of both HTS loops. The system is normally full of D_2O and isolated from the HTS by power-operated valves.

The shutdown cooling pumps are sized to ensure that boiling does not occur in any of the fuel channels at initial startup. During normal cool-down, steam from the steam generators bypasses the turbine and flows into the turbine condenser, thereby reducing the HTS temperature to 177°C in approximately 30 minutes.

In order to achieve cool-down from 177°C to 77°C, the isolating valves at the reactor headers are opened and all heat transport pumps are kept running. The heat transport pumps force a portion of the total core flow through the shutdown cooling heat exchangers where it is cooled by recirculated cooling water flowing around the heat exchanger coils.

After cooling to below 100°C, the heat transport pumps are shut down and the shutdown cooling system pumps started. The system is then cooled to 54°C in this mode, enabling D_2O to be drained down to the level just above the reactor headers, if required for maintenance of the steam generators or pumps.

(d) The Heat Transport Purification System:

- Limits the accumulation of corrosion products in the coolant by removing soluble and insoluble impurities
- Removes accumulations of fine solids following their sudden release due to chemical, hydraulic, or temperature transients
- Maintains the pD (pH of D_2O) within the required range

Flow is taken from one reactor inlet header of each heat transport loop, passed through an interchanger, cooler, filter, and ion-exchange column before being returned through the interchanger to a pump inlet in each circuit. The pressure generated by the heat transport pump produces the flow through the purification system. The interchanger–cooler combination minimizes the heat loss in the D_2O purification cycle.

4.2.1.3.5 Moderator and Auxiliary Systems

The moderator absorbs 4.5% of reactor thermal power. The largest portion of this heat is from gamma radiation. Additional heat is generated by moderation (slowing down) of the fast neutrons produced by fission in the fuel and a small amount of heat is transferred to the moderator from the hot pressure tubes. For reactivity control, gadolinium, and occasionally boron, can be added or removed from the moderator fluid.

The moderator system includes two 100% capacity pumps, two 50% flow capacity heat exchangers cooled by recirculated light water, and a number of control and check valves. Connections are provided for the purification, liquid poison addition, D_2O collection, supply and sampling systems. The series/parallel arrangement of the moderator system lines and valves permits the output from either pump to be cooled by both of the heat exchangers and ensures an acceptable level of moderator cooling when either of the two pumps is isolated for maintenance. Reactor power must be reduced to about 60% if one moderator heat exchanger is isolated. The primary functions of the system are to:

- Provide moderator cooling
- Control the level of heavy water in the calandria
- Maintain the calandria outlet temperature at approximately 70°C

The normal electric power supplied to the moderator system is backed up with an emergency power supply.

The heavy water in the calandria functions as a heat sink in the unlikely event of a LOCA in the HTS coinciding with a failure of emergency core cooling.

Helium is used as a cover gas for the moderator system because it is chemically inert and is not activated by neutron irradiation. Radiolysis of the heavy water moderator in the calandria results in the production of deuterium and oxygen gases. Circulation of the cover gas to catalytic recombiners reforms heavy water and prevents accumulation of these gases. The deuterium and oxygen concentrations are maintained well below levels at which an explosion hazard would exist.

The cover gas system includes two compressors and two recombination units that form a circuit for the circulation of cover gas through the calandria relief ducts. Normally, one compressor and both recombination units are operated, with the other compressor held on stand-by.

The moderator purification system:

- Maintains the purity of D_2O, thereby minimizing radiolysis which can cause excessive production of deuterium in the cover gas
- Minimizes corrosion of components by removing impurities present in the D_2O and by controlling the pD
- Reduces, under operator command, the concentration of the soluble poisons, boron, and gadolinium, in response to reactivity demands
- Removes the soluble poison gadolinium after shutdown system 2 (SDS2) has operated

Isolation valves in the purification system inlet and outlet lines are provided for maintenance purposes. The valves also allow drainage of the HTS coolant to just above the elevation of the headers without the need to drain the purification system. These valves close automatically in the event of LOCA.

The D_2O sampling system allows samples to be taken from the:

- Main moderator system
- Moderator D_2O collection system
- Moderator purification system
- D_2O cleanup system

Analyses may be performed on the samples to establish whether the chemistry of the heavy water falls within the specified range of chemistry parameters. These parameters include pD, conductivity, chloride concentration, isotopic purity, boron and gadolinium concentrations, tritium concentration, fluoride concentration, and organic content.

4.2.1.3.6 Reactor Regulating System

The fundamental design requirement of the reactor regulating system is to control the reactor power at a specified level and, when required, to maneuver the reactor power level between set limits at specified rates. The reactor regulating system combines the reactor's neutron flux and thermal power measurements by means of reactivity control devices and a set of computer programs to perform three main functions:

- Monitor and control total reactor power in order to satisfy station load demands
- Monitor and control reactor flux shape
- Monitor important plant parameters and reduce reactor power at an appropriate rate if any parameter is outside specified limits

Reactor regulating system action is controlled by digital computer programs that process the inputs from various sensing devices and activate the appropriate reactivity control devices.

All neutron flux measurement and control devices, both vertical and horizontal, are located in the low-pressure calandria, perpendicular to and between the horizontal fuel channels.

Computer programs provide the following:

- Reactor power measurement and calibration
- Demand power routine
- Reactivity control and flux shaping
- Set-back routine
- Step-back routine
- Flux mapping routine

The principal instrumentation utilized for reactor regulation includes:

- Ion chamber system
- Self-powered, in-core flux detectors
- Thermal power instrumentation

The nuclear instrumentation systems are designed to measure reactor neutron flux over the full operating range of the reactor. These measurements are required as inputs to the reactor regulating system and safety systems. The instrumentation for the safety systems is independent of that used by the reactor regulating system.

The reactivity control devices provide short-term global and spatial reactivity control. The devices are of two major types: mechanical and liquid.

The mechanical devices are the mechanical control absorbers and adjuster assemblies. The mechanical control absorbers comprise tubes containing cadmium (neutron absorber) that can be inserted to reduce power quickly. The adjuster assemblies comprise stainless steel tubes that are used to produce axial flattening of the fuel bundle powers as necessary. They can be removed from the core to add reactivity.

The liquid reactivity devices consist of the light water zone control units and the liquid poison addition system.

The function of the zone control system is to maintain a specified amount of reactivity in the reactor, this amount being determined by the deviation from the specified reactor power set point. If the zone control system is unable to provide the necessary correction, the program in the reactor regulating system draws on other reactivity control devices. Positive reactivity can be added by withdrawal of absorbers. Negative reactivity can be induced by insertion of mechanical control absorbers or by automatic addition of poison to the moderator.

The reliability of the reactor regulating system is of paramount importance and is achieved through having:

- Direct digital control from dual redundant control computers
- Self-checking and automatic transfer to the stand-by computer on fault detection
- Control programs that are independent of each other
- Duplicated control programs
- Duplicated and triplicated inputs
- Hardware interlocks that limit the amount and rate of change of positive reactivity devices

4.2.1.3.7 Balance of Plant

The balance of plant comprises the steam lines from the steam generators, the steam turbines and the alternating electrical generator, the condenser, various moisture separators and equipment to achieve de-aeration, demineralization, oxygen scavenging, reheating, and pH control of the feedwater returned to the steam generator.

The turbine generator system comprises steam turbines directly coupled to an alternating current (AC) electrical generator operating at synchronous speed.

The steam turbine is a tandem compound unit, generally consisting of a double-flow, high-pressure turbine and three double-flow, low-pressure turbines, which exhaust to a high vacuum condenser for maximum thermal efficiency. The condenser may be cooled by sea, lake or river water, or by use of atmospheric cooling towers.

The generator is a high-efficiency, hydrogen-cooled machine arranged to supply AC at medium voltage to the electric power system.

4.2.1.3.7.1 Feedwater and Main Steam System Feedwater flows from the condenser via the regenerative feedwater heating system and is supplied separately to each steam generator. The feedwater is pumped into the steam generators by feedwater pumps with the flow rate to each steam generator regulated by feedwater control valves. A check valve in the feedwater line of each steam generator is provided to prevent backflow in the unlikely event of feedwater pipe failure. An auxiliary feedwater pump is provided to satisfy low-power feedwater requirements during shutdown conditions, or in the event that the main feedwater pumps become unavailable.

The chemistry of the feedwater to the steam generators is precisely controlled by demineralization, de-aeration, oxygen scavenging, and pH control. A blowdown system is provided for each steam generator that allows impurities collected in the steam generators to be removed to prevent their accumulation and possible long-term corrosive effects. In some reactors, the blowdown is collected and recirculated.

The heat supplied to the steam generators produces steam from the water that flows over the outside of the tubes. Moisture is removed from the steam by the steam separating equipment located in the drum (upper section) of the steam generator. The steam then flows via four separate steam mains, through the wall of the reactor building, to the turbine where they connect to the turbine steam chest via a main steam line isolation valve.

Steam pressure is normally controlled by the turbine governor valves, which admit steam to the high-pressure stage of the turbine. If the turbine is unavailable, up to 70% of full power steam flow can bypass the turbine and go directly to the condenser. Turbine bypass valves control pressure during this operation. Auxiliary bypass valves are also provided to permit up to 10% of full-power steam flow to discharge to the condenser during low-power operation. Steam pressure can be controlled by discharging steam directly into the atmosphere via four atmospheric steam discharge valves that have a combined capacity of 10% of full power steam flow. These valves are used primarily for control during warm-up or cool-down of the HTS.

Four safety relief valves connected to each steam main provide overpressure protection of the steam system.

4.2.1.3.7.2 Turbine Generator System The steam produced in the steam generators enters a single high-pressure turbine and its water content increases as it expands through this high-pressure stage. On leaving this stage, the steam passes through separators where the moisture is removed. It then passes through reheaters where it is heated by live steam taken directly from the main steam lines. The reheated steam then passes through the low-pressure turbines and into the condenser where it condenses to water that is then returned to the steam generators via the feedwater heating system.

The steam turbine is a tandem compound unit, directly coupled to an electrical generator by a single shaft. It comprises one double-flow, high-pressure cylinder followed by external moisture separators, live steam reheaters and three double-flow, low-pressure cylinders (recent and future plants have two low-pressure cylinders). The turbine is designed to operate with saturated inlet steam. The turbine system has main steam stop valves, governor valves, and reheat intercept and emergency stop valves, depending on the arrangement preferred by the architect/engineer. All of these valves close automatically in the event of a turbine protection system trip.

The generator is a three-phase, four-pole machine that typically operates at 1800 rpm to serve 60-Hz electrical systems, and at 1500 rpm to serve 50-Hz systems.

The associated equipment consists of a solid-state automatic voltage regulator that controls a thyristor converter which in turn supplies the generator field via a field circuit breaker, generator slip rings and brush gear. The main power output from the generator to the step-up transformer is by means of a forced air-cooled, isolated phase bus duct, with tap offs to the unit service transformer, excitation transformer and potential transformer cubicle.

The turbine condenser consists of three separate shells, each shell being connected to one of the three low-pressure turbine exhausts. Steam from the turbine flows into the shell where it flows over a tube bundle assembly through which cooling water is pumped and is condensed. The condenser cooling water system typically consists of a once-through circuit, using sea, lake or river water. The condensed steam collects in a tank at the bottom of the condenser (termed the "hot well"). A vacuum system is provided to remove air and other non-condensable gases from the condenser shells. The condenser is designed to accept turbine bypass steam, thereby permitting the reactor power to be reduced from 100 to 70% if the turbine is unavailable. The bypass can accept 100% steam flow for a few minutes, and 70% of full-power steam flow continuously. On its return to the steam generators, condensate from the turbine condenser is pumped through the feedwater heating system. First, it passes through three low-pressure feedwater heater units, each of which contains two heaters fed by independent regenerative lines. This permits maintenance work to be carried out on the heaters with only a small effect on the turbine generator output. Two of the heater units incorporate drain cooling sections and the third a separate drain cooling stage.

Next, the feedwater enters a de-aerator where dissolved oxygen is removed. From the de-aerator, the feedwater is pumped to the steam generators through two high-pressure feedwater heaters, each incorporating drain cooling sections.

4.2.1.3.7.3 Power System Station Services The other major system of a nuclear plant is the electric power system. The normal electric power system comprises a main power output transformer, unit and service transformers, and a switchyard. This system steps up (increases) the generator output voltage to match the electric utility's grid requirements for transmission to the load centers and also supplies the power needed to operate all of the station services. The main switchyard portion of the electric power system permits switching outputs between transmission lines and comprises automatic switching mechanisms and lightning and earthing protection to shield the equipment against electrical surges and faults.

The station services power supplies are classified according to their required levels of reliability. The reliability requirement of these power supplies is divided into four classes that range from uninterruptible power to power that can be interrupted with limited and acceptable consequences. The electric power system station services comprise the supply systems described below.

(1) **Class IV Power Supply:** Power to auxiliaries and equipment that can tolerate long duration interruptions without endangering personnel or station equipment is obtained from a Class IV power supply. This class of power supply comprises:
   - Two primary medium-voltage buses, each connected to the secondary windings of the system service and unit service transformers in such a way that only one bus is supplied from each transformer.
   - Two medium-voltage buses supplied from the secondary windings of two transformers on the primary medium voltage buses. These buses supply the main heat transport pumps, feed pumps, water circulation pumps, extractor pumps, and chillers.

A complete loss of Class IV power will initiate a reactor shutdown.

(2) **Class III Power Supply:** AC supplies to auxiliaries that are necessary for the safe shutdown of the reactor and turbine are obtained from the Class III power supply with a stand-by diesel generator backup. These auxiliaries can tolerate short interruptions in their power supplies. This class of power supply comprises:
   - Two medium-voltage buses supplied from the secondary windings of the two transformers on the Class IV primary medium voltage buses, which supply power to the pumps in the service water system, ECCS, moderator circulation system, shutdown cooling system, HTS feed lines, steam generator auxiliary feed line, and the air compressors and chillers.
   - Several low-voltage buses.

(3) **Class II Power Supply:** Uninterruptible AC supplies for essential auxiliaries are obtained from the Class II power supply, which comprises:
   - Two low-voltage AC three phase buses that supply critical motor loads and emergency lighting. These buses are each supplied through an inverter from a Class III bus via a rectifier in parallel with a battery.
   - Three low-voltage AC single-phase buses that supply AC instrument loads and the station computers. These buses are fed through an inverter from Class I buses, which are fed from Class III buses via rectifiers in parallel with batteries. In the event of inverter failure, power is supplied directly to the applicable low-voltage bus and through a voltage regulator to the applicable instrument

bus. If a disruption or loss of Class III power occurs, the battery in the applicable circuit will provide the necessary power without interruption.

(4) **Class I Power Supply:** Uninterruptible direct current (DC) supplies for essential auxiliaries are obtained from the Class I power supply, which comprises:
   - Three independent DC instrument buses, each supplying power to the control logic circuits and to one channel of the triplicated reactor safety circuits. These buses are each supplied from a Class III bus via a rectifier in parallel with a battery.
   - Three DC power buses that provide power for DC motors, switchgear operation and for the Class II AC buses via inverters. These DC buses are supplied from Class III buses via a rectifier in parallel with batteries.

(5) **Automatic Transfer System:** In order to ensure continuity of supply in the event of a failure of either the unit or system power, an automatic transfer system is incorporated on the station service buses. Transfer of load from one service transformer to the other is accomplished by:
   - A manually initiated transfer of power under normal operating conditions, or an automatically initiated transfer for mechanical trips on the turbine.
   - A fast, open transfer of power, supplied automatically to both load groups of the Class IV power supply system, when power from one transformer is interrupted. This fast transfer ensures that the voltage and the phase differences between the incoming supply and the residual on the motors have no time to increase to a level that would cause excessive inrush currents.
   - A residual voltage transfer, comprising automatic closure of the alternate breaker after the residual voltage has decayed by approximately 70%. This scheme is time-delayed, and may require load shedding and could result in reactor power cut-back. It is provided as a backup to the above transfers.

(6) **Station Battery Banks:** The station battery banks are all on continuous charge from the Class III power supply and in the event of a Class III power disruption will provide power to their connected buses.

(7) **Stand-by Generators:** Stand-by power for the Class III loads is supplied by diesel generator sets, housed in separate rooms with fire-resistant walls. Redundant diesel generators are available, capable of supplying the total safe shutdown load of the unit. The Class III shutdown loads are duplicated, one complete system being fed from each diesel generator. In the event of a failure of Class IV power, diesel generators will start automatically. The generators can be up to speed and ready to accept load in less than two minutes. The total interruption time is limited to three minutes. Each generator automatically energizes half of the shutdown load through a load-sequencing scheme. There is no automatic electrical tie between the two generators, nor is there a requirement for them to be synchronized. In the event of one generator failing to start, the total load will be supplied from the other generator.

(8) **Emergency Power Supply System:** The emergency power supply system can provide all shutdown electrical loads that are essential for safety. This system and its buildings are seismically qualified to be operational after an earthquake. The system provides a backup for one group of safety systems (SDS2, emergency water supply (EWS), and secondary control area) if normal electric supplies become

unavailable or if the main control room becomes uninhabitable. The system comprises two diesel generating sets, housed in separate fire-resistant rooms, which are self-contained and completely independent of the station's normal services. There is adequate redundancy provided in both the generating distribution equipment and the loads.

4.2.1.3.7.4 Station Instrumentation and Control Digital computers are used for station control, alarm annunciation, graphic data display and data logging. The system consists of two independent digital computers (DCCX and DCCY), each capable of station control. Both computers run continuously, with programs in both machines switched on, but only the controlling computer's outputs are connected to the station equipment. In the event that the controlling or directing computer fails, control of the station is automatically transferred to the "hot" stand-by computer. In the event of a dual computer failure, the station will automatically shut down.

Individual control programs use multiple inputs to ensure that erroneous inputs do not produce incorrect output signals. This is achieved by rejecting:

- Analog input values that are outside the expected signal range
- Individual readings that differ significantly from their median, average or other reference

A spare computer is provided as a source of spare parts for the station computers. It is also used for:

- Program assembly and checkout
- Operator and maintainer training
- Fault diagnosis in equipment removed from the station computers

Computerized operator communication stations replace much of the conventional panel instrumentation in the control room. A number of human–machine communication stations, each essentially comprising a keyboard and colour cathode ray tube monitor, are located on the main control room panels. The displays provided on the monitors include:

- Graphic trends
- Bar charts
- Status displays
- Pictorial displays
- Historical trends

Printed copies of the displays on any display monitor the operator wishes to record can be obtained from the line printers. The digital computers are also used to perform the control and monitoring functions of the station and are designed to be:

- Capable of handling normal and abnormal situations
- Capable of automatically controlling the unit at startup and at any pre-selected power level within the normal loading range

- Capable of automatically shutting down the unit if unsafe conditions arise
- Tolerant of instrumentation failures

The functions of the overall station control system are performed by control programs loaded into each of the two unit computers. The major control function programs are

- The reactor regulation program, which adjusts the reactivity control devices to maintain reactor power equal to its desired set point
- The steam generator pressure program, which controls steam generator pressure to a constant set point by changing the reactor power set point (normal mode), or by adjusting the station loads (alternate mode)
- The steam generator level control program, which controls the feedwater valves in order to maintain the water level in the steam generators at a reactor power dependent level set point
- The HTS pressure program, which controls the pressurizer steam-bleed valves and heaters in order to maintain HTS pressure at a fixed set point

There are also programs for:

- HTS control
- Moderator temperature control
- Turbine run-up and monitoring
- Fuel-handling system control

There are two modes of operation of the reactor: the "reactor-following-turbine" mode and the "turbine-following-reactor" mode.

In the reactor-following-turbine mode of operation, the turbine generator load is set by the operator: the steam generator pressure control program "requests" variations be made to reactor power in order to maintain a constant steam generator pressure. This control mode is termed reactor-follows-turbine or "reactor-follows-station loads."

In the turbine-following-reactor control mode (i.e., turbine-follows-reactor), station loads are made to follow the reactor output. This is achieved by the steam generator pressure-control program, which adjusts the plant loads in order to maintain a constant steam generator pressure. This mode is used at low reactor power levels, during startup or shutdown, when the steam generator pressure is insensitive to reactor power. It is also used in some upset conditions when it may not be desirable to manoeuver reactor power.

4.2.1.4 Features of Other HWRs

4.2.1.4.1 Integrated 4-unit CANDU HWRs

Ontario Power Generation and Bruce Power utilities operate the majority of operating CANDU plants as 4-unit stations either as 525-MW or 540-MW units (Pickering A and B) or 825-MW units (Bruce A and Bruce B) or 935-MW units (Darlington). These integrated 4-unit stations, although nominally similar (featuring common control room area, emergency coolant injection, and electrical and service water systems) differ in the number of channels, number of fuel bundles in the channels, outlet temperatures and support components inside and outside the channels. There are also differences in the shutdown mechanisms as well as the design of shield tank and shielding material.

The HTS differs also by having preheaters separate from the steam generators and in the number of steam generators and in Bruce A having the steam generators attached to a common steam drum. Also the containment of each unit is connected to a large vacuum building by shafts and sealed with valves that can be opened after a severe system accident to draw radioactivity into the vacuum building.

4.2.1.4.2 Carolinas–Virginia Tube Reactor (CVTR)

The CVTR heavy water cooled and moderated pressure tube reactor was built as a power demonstration reactor at Parr, South Carolina, U.S. Construction started in 1960 and the reactor was completed and connected to the grid by the end of 1963. The CVTR generated 19 MWe and, after about four years of operation, a planned experimental program having been completed, it was shut down and eventually decommissioned. The reactor circuit contained many of the features of later pressurized heavy water cooled and moderated reactors, including a pressurizer and, notably, an oil-fired superheater to upgrade the quality of steam being fed to the turbine.

4.2.1.4.3 Pressure Tube Boiling Light Water Coolant, Heavy Water Moderated Reactors

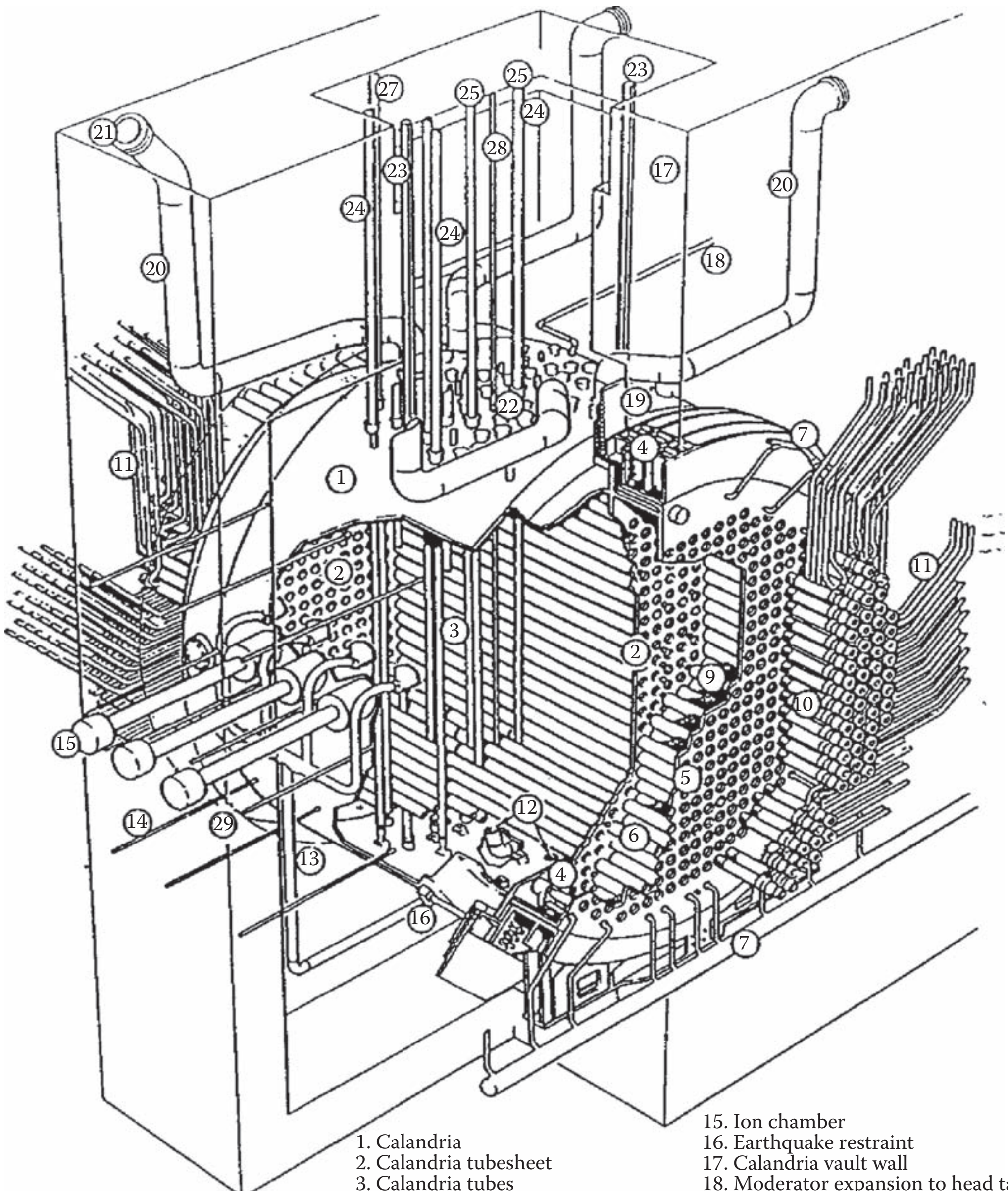
Four countries have evaluated the reactor system in which light water is brought to boiling in vertically oriented pressure tubes, the steam–water mixture being sent to steam drum separators and the steam used directly to drive a turbine. The arrangement simulates the conventional recirculation boiler. The Russian RBMK is a similar type of reactor, except that graphite is used as the moderator.

Each country had different reasons for initiating studies of this type of reactor. In the UK, there was a search for a more economic thermal reactor for electricity production than the Magnox or the Advanced Gas-cooled Reactors, and one that would avoid the use of graphite as a moderator as well as the use of a large-pressure vessel. Pressure vessel property changes during life and potential problems with resealing the vessel after refuelling were then current concerns. In Canada in the early 1960s, there was concern that the heavy water coolant system in the PHWRs would not be sufficiently leak-tight to produce acceptable heavy water losses, and there was a desire to develop a less capital-intensive reactor by using light water coolant.

In Italy, the intention was to develop a reactor that was independent of enriched fuel, while in Japan the HWR was seen as part of a future fuel recycling strategy where spent fuel from PWRs would be recycled through HWRs to make use of the fissile material remaining in the fuel.

Thus, a prototype reactor was built in each country using the experience gained, which is described in subsequent sections. It should be noted that an Advanced CANDU Reactor (ACR) design is being developed by AECL. It incorporates a light water coolant, heavy water moderator and enriched fuel, with horizontal channels.

The general characteristics of a pressure tube boiling light water heavy water moderated (BLW-HWM) system are illustrated in Figure 4.2. The pressure tubes (vertically oriented) contain the fuel and the light water entering at the bottom of the fuel channel is brought to boiling, about 10 wt% of water being converted to steam. The steam–water mixture passes to the steam drums and virtually dry, saturated steam is supplied directly to the turbine. The exhaust steam is condensed and returned to the water space in the steam drums via a feed heating train.



1. Calandria
2. Calandria tubesheet
3. Calandria tubes
4. Embeoment ring
5. Fuelling machine tubesheet
6. End shield lattice tubes
7. End shield cooling pipes
8. Inlet–outlet strainer
9. Steel ball shieloing
10. End fittings
11. Feeder pipes
12. Moderator outlet
13. Moderator inlet
14. Horizontal flux detector unit
15. Ion chamber
16. Earthquake restraint
17. Calandria vault wall
18. Moderator expansion to head tank
19. Curtain shieloing slabs
20. Pressure relief pipes
21. Rupture disc
22. Reactivity control unit nozzles
23. Viewing port
24. Shutoff unit
25. Adjuster unit
26. Control absorber unit
27. Zone control unit
28. Vertical flux detector unit
29. Liquid injection shutdown nozzle

FIGURE 4.2
CANDU PHWR (schematic).

4.2.1.4.4 Steam Generating Heavy Water Reactor (SGHWR)

This reactor started operation in 1968 with a designed output of 100 MWe. The use of light water coolant and heavy water moderator means that with the choice of the appropriate fuel to moderator ratio the void coefficient could be made to approach zero and even to be slightly negative. The reactor operated for 23 years at average capacity factor of 60%. However, a commercial design was not considered economic and the program was cancelled in 1977.

4.2.1.4.5 Gentilly-1

The Gentilly-1 pressure tube reactor was a 250-MWe heavy water moderated and boiling light water cooled design fuelled with natural uranium dioxide. The reactor concept had been developed in the early 1960s and in 1966 the reactor was committed for construction. First power was produced in 1971 and full power attained in May 1972. It was shut down in April 1979, and by 1984 had been decommissioned.

4.2.1.4.6 Fugen

The 165-MWe Fugen reactor was the prototype of what was to be a line of 600-MWe reactors that would form, in conjunction with PWRs, the Japanese fuel recycling strategy. A 600-MWe design was found to be too costly and the development of MOX fuels led to the demise of the project.

4.2.1.4.7 Cirene

The Cirene reactor was a 40-MWe prototype power plant constructed at Latina, 80 km south of Rome. Construction started in 1976 and completion was scheduled for 1984. Commissioning stopped in 1988, before work to reduce the positive void reactivity coefficient was complete, by the general moratorium on nuclear reactor operation imposed by the Italian Government following the Chernobyl accident.

4.2.1.5 Summary

The comments given in the earlier section can be applied generally to this line of BLW-HWM reactors. The use of light water coolant, dispensing with steam generators, is economically attractive. These advantages have been offset by necessary design modifications in the fuel and other features, and by a necessary increase in the number of channels needed to achieve the same output as pressurized PHWR versions. However, provided these problems are addressed in revised designs, this line of reactors should be cheaper to build and to operate than the heavy water cooled versions. A conceptual design for a reactor which uses plutonium and thorium fuel is being pursued by India as part of its overall plan for nuclear-based electricity generation.

4.2.2 Characteristics of Pressure Vessel HWRs

HWRs of the pressure vessel type have been designed and constructed in Sweden, Germany, and Argentina. The main references of this line are: the Ågesta reactor in Sweden (shutdown), the MZFR reactor in Germany, and the Atucha 1 and Atucha 2 (the latter under construction) reactors in Argentina.

4.2.2.1 Ågesta

In Sweden, the first pressure vessel pressurized HWR was constructed at Ågesta. This was a project that combined the objectives of two separate concepts: one for a district heating reactor and the other for a heat and power reactor. The pressure vessel reactor was conceived as a 65 MWth prototype plant that was to supply district heating and electricity (10 MWe). The reactor was located in an underground chamber excavated in solid rock and serviced a suburb of Stockholm. The reactor operated with a good degree of reliability. Operation was interrupted over the summer months when district heating was not required. The reactor was shut down in 1975 and decommissioned because it had ceased to be an economical source of power.

4.2.2.2 MZFR

The 57-MWe MZFR reactor was built by Siemens-KWU at the Karlsruhe Research Center for limited electricity supply and district heating. It was the prototype for the Atucha-1 and Atucha-2 reactors built in Argentina. The principal features are similar to those incorporated in Atucha-1.

4.2.2.3 Atucha-1

The reactor core is approximately cylindrical and consists of vertical fuel assemblies located in the same number of fuel channels. The coolant channels are arranged on a triangular lattice pitch and penetrate the top and bottom plenums located inside a cylindrical pressure vessel containing the moderator heavy water at a similar pressure to the HTS.

As reactor heavy water coolant and the moderator heavy water are kept at nearly the same pressure, thin-walled tubes were sufficient to separate the fluids. The fuel channel tubes can thus be categorized as reactor internals. Also, the two systems use the same auxiliary systems to maintain water quality.

The moderator water at a temperature of 210°C is used to preheat the feedwater, producing a net efficiency of operation of approximately 29% for Atucha 1 and 32% for Atucha 2.

Reactivity can be controlled by "black" and "grey" absorbers arranged in groups or banks of three azimuthally symmetric absorber rods. These penetrate the vertical matrix of fuel channel tubes at an angle to the vertical. The reactivity can also be controlled by boron additions and by varying moderator temperature.

The fuel is a long string with 37 elements with extensions to allow the fuelling machine to extract the fuel. It can be refuelled on-power with a single fuelling machine operating above the reactor vessel cover head.

The containment is a spherical stainless steel housing which is protected against external impacts by the surrounding reinforced concrete reactor building.

The HTS consists of the reactor vessel, two steam generators, two primary pumps and the pressurizer that keeps pressure at approximately 11.65 MPa. The system has two loops, and for Atucha 1 the exit temperature from the pressure vessel is ~300°C and the inlet temperature of the return coolant into the pressure vessel is 265°C. Atucha-2, which has yet to be completed, is a larger version of Atucha-1 with more channels.

4.2.2.4 Characteristics of Heavy Water Moderated, Gas-Cooled Reactors

4.2.2.4.1 Introduction

Four gas cooled pressure tube reactors of relatively small size were built in the 1960s with the object of exploring the use of CO_2 as a heat transport fluid in combination with heavy water moderation instead of graphite. The reactors had innovative fuel designs and most had the pressure tubes vertically oriented although the most successful unit, the EL4 plant in France, had the pressure tubes horizontal.

The potential advantages were low-neutron absorption by the coolant and high outlet coolant temperatures available at moderate pressures. The disadvantages lay in the relatively poor heat transfer and heat transport properties of CO_2.

The advantage of using CO_2 is that the heat transport gas can be heated to much higher temperatures than is possible with water and achieve higher thermal efficiencies at the turbine. Typically, the temperature reached by the CO_2 is about 500°C. The heat is exchanged in steam generators to produce the steam to drive turbines.

4.2.2.4.2 The EL4 Reactor

The EL4 reactor (70 MWe) was constructed at the Mont d'Aree site near Brennilis, France. The heavy water moderator is contained in a horizontal cylinder 4.6-m long and 4.8 m in diameter. The 216 fuel channels, arranged on a square pitch of 234 mm, are contained in Zircaloy tubes. The Zircaloy pressure tubes (107-mm inside diameter and 3.2-mm wall thickness) can operate at a low-temperature by virtue of their being thermally isolated from the hot CO_2 gas by a stainless steel guide tube and by thermal insulation between the guide tube and the pressure tube.

The EL4 reactor started up in l965. It had initial problems with steam generators, which were overcome in the first two years of operation, and it was not able to use beryllium alloy fuel cladding as intended. However, it operated successfully until 1985 when it was shut down, together with some other gas cooled reactors, because Electricité de France (EDF) decided to concentrate on PWRs.

The advantages of this reactor were the relatively low-cost per unit of electricity and the low-fields occurring in the reactor vault. As a result of the absence of activity transport, the reactor face and vault were accessible when the reactor was on-power.

4.2.2.4.3 The Niederaichbach Reactor

The 100-MWe Niederaichbach reactor was designed by Siemens in the early 1960s and constructed between 1965 and 1970 in the Isar valley, about 70 km northwest of Munich. The reactor contained 351 vertical channels on a square pitch of 24.5 cm. The channels penetrated a tank or calandria containing heavy water. Basic control was achieved by adding a burnable poison, $CdSO_4$, to the moderator. The moderator level could also be adjusted and the moderator dumped to shut down the reactor.

The Niederaichbach reactor reached full power in 1970 and was connected to the grid in 1973. It was shut down in 1974 when it was deemed to have become uneconomic compared with other water cooled reactors, and the subsequent decommissioning activity had the objective of demonstrating the ability to return a reactor site to a greenfield condition.

4.2.2.4.4 Lucens Reactor

The Lucens reactor was constructed in the period 1962–1968 in underground caverns at a site between Lausanne and Berne. It was a 30 MWth/8.3 MWe pressure tube reactor with CO_2 cooling and was designed to combine features of the French reactors and the British Magnox units with heavy water moderation. The reactor only operated for a few months before a three-month shutdown was required for maintenance. During the shutdown, a blockage was caused by the accumulation of corrosion products in some channels resulting from the effects of water condensation on the magnesium alloy fuel cladding. At startup, the flow blockage remained undetected during the subsequent rise to power owing to flow bypass of the blocked sub-channels. The cladding melted and further obstructed the flow, leading to a uranium fire, graphite column contact with the pressure tube as a result of bowing, and pressure tube failure by overheating and subsequent rupture. The calandria tube was also ruptured. As a result, the reactor was shut down and eventually decommissioned. Before commissioning, it was recognized that the design was not supported by the Swiss electrical utilities and its operation was intended for experimental purposes.

4.2.3 Unique Features of HWR Technology Fuel Channel Technology

Fuel channels are a common feature of HWRs of all types. Components of fuel channels can be grouped into three main elements:

- Pressure-retaining components, including the out-of-core channel extensions and the mechanical closures accessed by fuelling machines in re-fuelling the channels
- Channel support components, which are more obvious as the end bearings and spacer/calandria tube components in CANDU 6 horizontal channels
- Channel internals, which may include radiation-shielding plugs, thermal shielding plugs, flow straighteners/modifiers, fuel supports, and the fuel

Because many of the fuel channel designs were "one-offs," there was little development of most concepts. In the case of the CANDU channel, development has been toward larger diameters and longer channels as the means of achieving higher power outputs at higher temperatures. This part of the development has now reached a limit as regards pressurized water conditions and development activities are now being directed toward achieving a longer channel life with limited modifications being made to the basic design.

In the previous sections, several reactor designs using heavy water moderation are briefly described. Based on the pressure tube boundary conditions, the fuel channels designs can be divided into three types:

- Channels with a high-temperature, high-pressure boundary
- Channels with a high-temperature, low-pressure boundary
- Channels with a low-temperature, moderate-pressure boundary

The aspects of fuel technology to be described in the various reactor designs will thus be addressed on the basis of the above divisions.

4.2.3.1 Channels with a High-Temperature, High-Pressure Boundary

The CANDU fuel channels are of this type, and typified by the CANDU 6 channel illustrated in Figure 4.3. Pressure-retaining components are the pressure tubes, end fittings and closure seals. The Zr–2.5%Nb pressure tube (104-mm inside diameter, with a 4-mm wall thickness and 6.1 m length) is produced by extrusion, cold working and stress relieving. The tube is roll-expanded into AISI type 403 stainless steel end fittings by a procedure that leaves low-tensile residual stresses at the end of the rolled zone. The total length of the fuel channel, including the end fittings, is 10.1 m. The channel is accessed at each end for fuel removal and replacement. New fuel is inserted at the inlet end and used fuel removed at the outlet end and there are 12 bundles in each channel. (In the initial build of Bruce

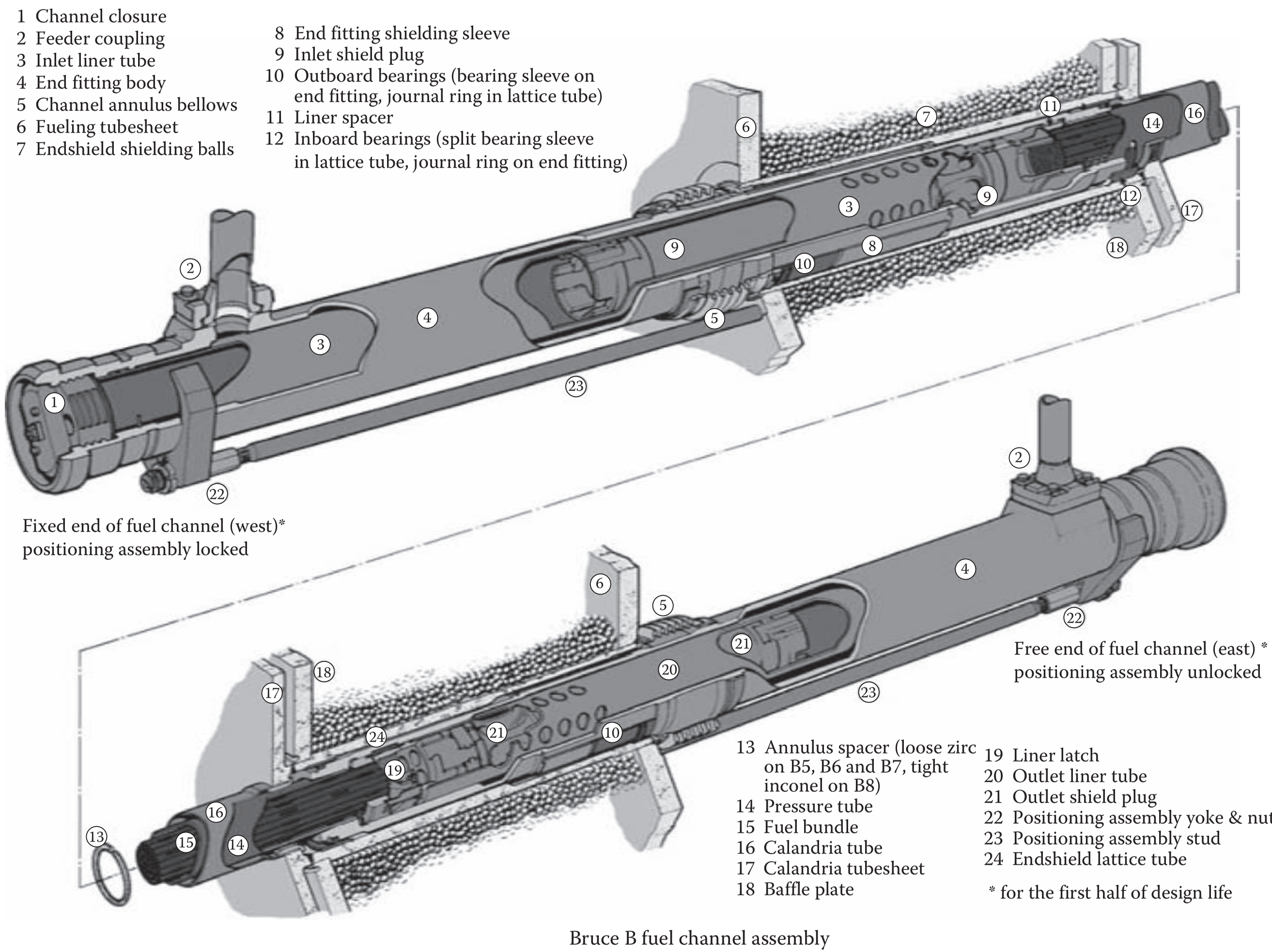


FIGURE 4.3
Bruce B fuel channel assembly.

reactors there are 13 bundles in each channel and fuel was inserted at the outlet end.) The fuel, in the form of 37-element bundles, can be stored in the rotating magazine of the fuelling machine before or after removal.

The pressure tube and contents are supported by linear sliding bearings at each end of the reactor. The journal bearings are formed by ring bearings on the end fittings mating with sleeve bearings in the lattice tube. The calandria tube supports the in-core section of the pressure tube through toroidally coiled spacers that accommodate relative axial and diametral movement between the pressure tube and the calandria tube.

Positioning assemblies at each end of the channel locate the channel in the reactor. Typically, the channel is positioned to allow elongation (caused by neutron irradiation) to take place on the full length of the bearings at one end by locking the end fitting at the other end to the positioning assembly. At half-life, the channels are relocated by releasing the channel and pushing it to the inboard extremity of the unused bearing length and locking it to the other positioning assembly. Each end fitting contains: (i) a liner tube to prevent the fuel bundles experiencing cross-flow on entering or leaving the fuel channel; (ii) a shield plug which supports the fuel at the outlet end and whereby flow is directed into the annulus between the liner tube and the end fitting body (and out through the side port) or from the liner annulus, through the shield plug and into the fuel without causing instability in the fuel; and (iii) a closure plug which can be opened by the fuelling machine. In the CANDU 6 channel, the seal forms part of a flexible dome that is pressed against a step in the end fitting in order to achieve a pressure face seal.

In response to the neutron flux, high temperatures, water environment and wear, the channels (mostly the pressure tube) change as follows:

- Dimensions change: the pressure tubes sag, expand and elongate. Typically, a CANDU 6 Zr–2.5%Nb pressure tube will expand >4%, elongate by 180 mm and sag up to 76 mm in 30 years. The calandria tubes sag (and support the pressure tubes) and the pressure tube will sag between spacers but is designed not to make contact with the calandria tubes.
- Pressure tubes pick up hydrogen (as deuterium) from corrosion and crevice reactions. The concentration of hydrogen after 30 years is predicted to be just above the terminal solid solubility at operating temperatures and the presence of hydrides during operation is not considered to influence behavior. The surface oxide resulting from corrosion has no structural effect.
- Mechanical properties of the in-core components change as a result of the fast neutron flux damage. The strength increases and ductility and fracture toughness decrease to shelf levels that are acceptable for service. Recent developments in pressure tube technology have made the pressure tubes more resistant to decreases in fracture toughness caused by irradiation.
- Pressure tubes wear. Light scratching by fuel bundle movement can occur. Debris, which can enter the channels from maintenance activities, can become trapped in the fuel and wear the pressure tube through vibration in the flowing water.

Each of these types of change must be monitored by inspection of periodically removed pressure tubes. Debris fretting must be prevented by operating with a "clean" HTS.

The fuel channels of the Pickering, Bruce and Darlington reactors and the Indian series of reactors are nominally similar, but differ in specific features.

4.2.3.1.1 The SGHWR Fuel Channel

The pressure tube was of Zircaloy 2. The pressure tube was reduced in diameter at the lower rolled joint where it was rolled into a hub that was, in turn, welded to the stainless inlet piping.

In the upper part of the channel, the pressure tube was rolled into the hub of the upper standpipe which had a side port connected for the coolant outlet, the emergency cooling inlet and, at the top, the closure seal for refuelling.

4.2.3.1.2 The Gentilly-1 Fuel Channel

The pressure tube in this reactor is heat treated Zr-2.5Nb and, at 2.41-mm wall thickness, was significantly thinner than the pressure tubes in PHWRs.

An insert was required to attach the thin-walled pressure tube to the end fittings. The calandria tube was separated from the pressure tube by spacers supported on interlocking support rings. The fuel was attached to a central structural tube, which was supported at the bottom and at the top by lower and upper shield plugs, respectively.

4.2.3.1.3 The Fugen Fuel Channel

As with the Gentilly 1 pressure tube, the Fugen channel was also made of heat-treated Zr-2.5%Nb, and the wall was only 2.2-mm thick, which also required the use of inserts in forming the rolled joint.

The lower rolled joint has an internal insert to “sandwich” the pressure tube between the insert and the end fitting. However, the upper end fitting sandwiches the pressure tube between an external insert and the end fitting. An upper extension tube connects the channel to the external piping via a reducer. The connection to the inlet feeder is made via a side port and the closure plug at the bottom makes a bore seal with the end fitting extension using a flexed dome component. The Fugen channel has functioned without problems.

4.2.3.1.4 Cirene Fuel Channel

The Cirene fuel channel is of similar design to the Gentilly 1 and Fugen channels. The pressure tube is made of Zircaloy 2 (106.1-mm inside diameter, 3.15-mm wall thickness).

4.2.3.2 Channels with a High-Temperature Low-Pressure Boundary

4.2.3.2.1 Atucha 1 Fuel Channel

In the original design of the Atucha 1 channel, the main shroud tube enclosing the fuel assembly was made of Zircaloy 4 and comprised a seam-welded tube (108.2-mm inside diameter and 1.6-mm or 1.72-mm in wall thickness. A thin (0.1-mm wall thickness) Zircaloy tube, dimpled to maintain separation, surrounded the shroud tube between it and the seam welded Zircaloy 4 insulation tube (0.4-mm wall thickness). These were attached to austenitic stainless steel end fittings. The bottom end fitting sat in the bottom plenum lattice port allowing a small gap for the circulation of heavy water into the moderator space. The upper end was fastened to the upper plenum.

In the replacement channels of Atucha 1 and proposed for Atucha 2, the Zircaloy 4 isolation tube has been eliminated in favor of a shroud tube and a surrounding insulation tube.

4.2.3.3 Channels with a Low-Temperature, Moderate-Pressure Boundary

4.2.3.3.1 The EL4 Fuel Channel

The pressure boundary tube of the EL4 channel contains a Zircaloy 2 tube rolled into the end shields of the moderator tank. Inside the Zircaloy tube is a stainless steel guide tube, insulated from the Zircaloy 2. The guide tube thus sees the 233–475°C temperatures of the CO_2 and carries the fuel assemblies. The seal plugs at the channel ends incorporate a ball valve for fuelling machine access.

4.2.3.3.2 Niederraichbach Fuel Channel

The Zircaloy 2 pressure tube operated at <100°C and was isolated from the hot CO_2 by a thin foil tube and a stainless steel insulating tube.

4.2.3.3.3 Lucens Fuel Channel

In the Lucens fuel channels, the Zircaloy 2 pressure tube was kept to the temperature of the inlet CO_2 gas (225°C) by passing the inlet gas between the carbon matrix fuel and the pressure tube. Low-temperature CO_2 also flowed in the annulus between the pressure tube and the calandria tube.

4.2.3.3.4 CVTR Fuel Channel

The fuel channels of the CVTR were made to a U-tube design, each leg containing one fuel assembly. The pressure tubes were made of Zircaloy 2. The fuel contained in the pressure tube was isolated from the wall of the pressure tube by inner and outer circular thermal baffle tubes, 0.7 mm and 0.3 mm in wall thickness, respectively. In addition, a hexagonal flow baffle tube, positioned inside the thermal baffles, concentrated the flow through the fuel.

The pressure tube was in contact with the moderator water and heat shielded from the fuel, and thus operated in a cold pressurized condition. The pressure tube was rolled into the U-fittings at the bottom of the reactor and into end fittings at the top of the reactor.

The channel tubes were made of aluminum alloy and arranged on a square lattice pitch. The channel tubes were isolated from the fuel assembly by a protective internal magnesium alloy tube which surrounded 150–200 small diameter (4 mm) fuel rods of natural uranium arranged in seven concentric rings around the center rod.

4.3 Heavy Water

4.3.1 Purpose of Heavy Water

For thermal—as opposed to fast—reactors, neutrons must be slowed down from the high speeds at which they are emitted by fissions, a process known as "moderation." Normal hydrogen (protium—symbol H as an isotope), because its atomic mass (1) is almost identical to that of a neutron, is the most effective moderator. To achieve high density and because oxygen atoms are almost transparent to neutrons, hydrogen is normally deployed in the form of water. Despite its unequalled performance in slowing neutrons, protium suffers from a serious disadvantage of capturing so many neutrons that reactors using light-water moderation require significant enrichment in ^{235}U of their uranium fuel. The rare heavier isotope of hydrogen deuterium (atomic mass = 2, symbol D) is a poorer match to the mass of the neutron and so requires more collisions (and a larger moderator volume) to effect moderation but is almost immune to capturing neutrons. Consequently, a reactor moderated with deuterium in the form of heavy water (D_2O) can use uranium with natural concentrations of ^{235}U.

There is, therefore, a choice between using light water and burning fuel that must always be enriched in fissile material or doing a one-time enrichment of heavy water and natural uranium as fuel. Because separating the isotopes of hydrogen is comparatively easy, taking the heavy-water route would be the natural choice *except* that deuterium is rare, occurring only in abundances of around one part in 7000 terrestrially. This has the effect of having to process very large volumes of feed material to produce heavy water and is the main reason behind its relatively high cost, typically $300/kg.

4.3.2 Heavy Water Production

The standard textbook on isotope separation remains Benedict, Pigford, and Levi's 1981 "Nuclear chemical engineering." It provides much more detail than is possible here. Because of the large (2:1) mass ratio of deuterium to protium, the affinities for the two isotopes are quite different in many chemical species. Over the practical operating temperature ranges for the different processes, with water and hydrogen, the equilibrium ratio varies from 3.8 excess in water at 25°C to 2.0 at 200°C. With ammonia and hydrogen, the ratio varies from 6.0 at −40°C to 3.5 at +30°C. With water and hydrogen sulfide, the ratio is around 2 and relatively weakly affected by temperature, ranging from 2.3 at 32°C to 1.8 at 130°C. Figure 4.4 shows the effect of temperature on the separation factors.

However, for a process, one needs a significant difference in equilibrium and adequate rates of reaction. Of the three systems mentioned above, only water and hydrogen sulphide come rapidly to equilibrium. The other systems require catalysts to reach workable rates of exchange. It was in this context that a process based on water and hydrogen sulfide became the near-standard technology for heavy water production even though hydrogen sulfide is a dangerous and highly corrosive substance and the variation of the separation factor is quite weak (2.3–1.8). The workable range of temperatures is limited (to between just under 30°C by formation of a solid gas hydrate and to around 130°C by the vapor pressure of

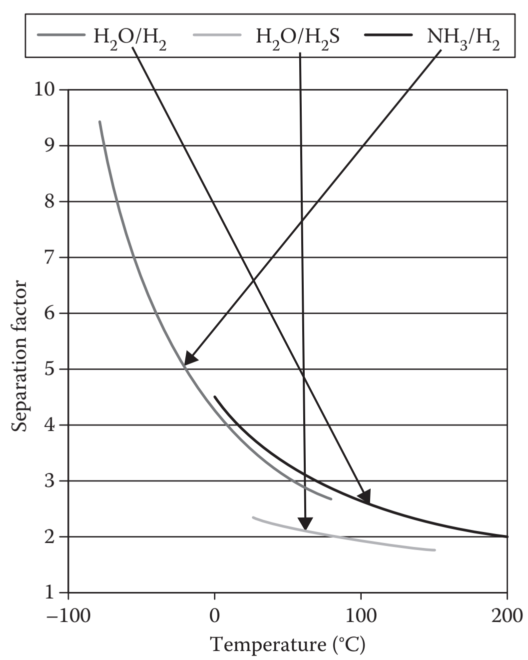


FIGURE 4.4
Equilibrium constants.

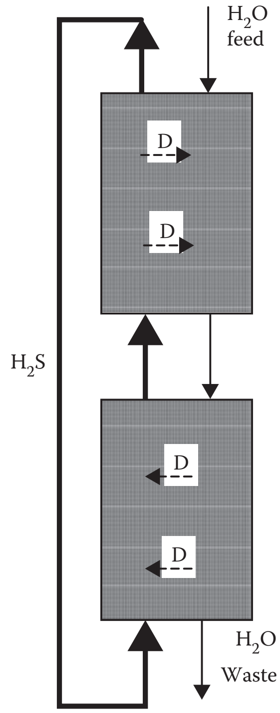


FIGURE 4.5
Schematic of a bithermal water–hydrogen sulfide process.

water) where the operating pressure is bounded by gas liquefaction at pressures above 2.2 MPa. The two advantages of this system are that one species is a gas and the other a liquid and that the deuterium can enter the process as water, with its limitless abundance.

In a conceptual, ideal process, water would be continuously converted into hydrogen sulphide and back into water in the arrangement illustrated in Figure 4.5. This is known as a "monothermal" process because deuterium exchange occurs at only one temperature. Between the two conversions, the two species are repeatedly contacted. With a equilibrium ratio around 2.3 at 30°C, the descending water grows steadily richer in deuterium and the hydrogen sulfide is steadily stripped of its deuterium content. At the bottom, water rich in deuterium is converted in hydrogen sulfide at the same concentration, providing an excellent driving force behind the water's enrichment. A large driving force leads to relatively few contact steps in the exchange column. At the top, water at less than half the natural deuterium abundance is produced and the difference from natural gives the extractive capacity of the process. Unfortunately, there is no practical way of implementing this conceptual process since the conversions of water to hydrogen sulfide and the reverse are too difficult to be economic.

So instead of this monothermal process, a bithermal process (usually known as Girdler-Sulfide or G-S) was developed. Instead of converting water into hydrogen sulfide, a second "hot" contact tower operating at a higher temperature is included (Figure 4.5). At 130°C, the equilibrium ratio for deuterium is 1.8. By having twice the molar flow of hydrogen sulfide to that of water, deuterium can be made to move back from the water into the hydrogen sulfide stream in the hot tower. Deuterium still moves from hydrogen sulfide to water in the cold tower but with lower driving forces because the water must absorb twice as much deuterium as would occur with the monothermal process. The deuterium content of the hydrogen sulfide

FIGURE 4.6
G-S plant.

leaving the top of the "cold" tower is low, enough to allow it to be recycled to the bottom of the hot tower and to initiate extraction of deuterium from the down-flowing water stream.

Because the variation in equilibrium ratio is rather small, there is only a small difference in deuterium concentrations between the feedwater entering the top of the cold tower and the water leaving the hot tower. In practice, about 17% of natural water's deuterium content can be extracted, compared with the 50% that would be possible if a monothermal process were practicable. This further raises the already large mass of water that a G-S plant must process to about 37,000-times the product, including allowance for heavy water being 10% heavier than normal water. In consequence, G-S plants are very large (Figure 4.6).

Despite its intrinsic limitations and difficulties, the G-S process was commercialized in the United States and subsequently on a larger scale in Canada and elsewhere. It had provided the preponderance of the 20,000 tonnes of heavy water produced worldwide to 2007. Production in Canada peaked around 1980 at around 1200 tonne/a from four plants. Since then, G-S production has been phased-out in Canada because ample stockpiles exist and more ACR designs require less heavy water. G-S production does continue in Romania and India.

Because of the G-S process's limitations, alternatives have been developed. The alternative that has been deployed in India and Argentina relies on ammonia–hydrogen exchange. This needs a catalyst and the potassium salt of ammonia, KNH_2, dissolved in the ammonia is used. Because of the high vapor pressure of ammonia, the exchange process is carried out at around −30°C. At that temperature, the catalyst has limited activity and its performance must be enhanced by devices that provide large surface area or intense agitation. The process has been configured as a monothermal and a bithermal process, with a hot tower temperature around 40°C, limited by the linked considerations of operating pressure and ammonia vapor pressure. A monothermal process requires cracking of ammonia to hydrogen and nitrogen below the exchange tower and their re-synthesis above the exchange tower. This is economically practical but means that the hydrogen flow is diluted, 25% of the gas being nitrogen. The potassium salt must also be stripped of deuterium and transferred to the re-synthesized ammonia. Configured monothermally or bithermally, plants must depend on processing a large stream of hydrogen or on an exchange step between ammonia and water. In the first instance, even the largest hydrogen production plants are only big enough to produce about 70 tonne/a of heavy water. (The typical scale of G-S plants has been 200–400 tonne/a.) Using ammonia–water exchange avoids this constraint but the ammonia must be very carefully dried to avoid reaction between it and the potassium salt. Because the available amount of hydrogen is constrained or a fairly demanding exchange step is substituted for hydrogen feed, the process is usually configured to give at least 80% extraction of the available deuterium.

AECL did extensive development of a variant of the ammonia–hydrogen process based on aminomethane (CH_3NH_2) rather than ammonia. This has better kinetics and a wider envelope of operating temperatures, but can only be configured bithermally. This process was superseded by development of processes based on water–hydrogen exchange.

Processes based on hydrogen–water exchange are attractive because their operating temperature range (between 25°C and 180°C) is easy to accommodate, neither substance is toxic, and the equilibrium constant is comparable in size and variation with temperature with that for ammonia–hydrogen. The lack of a suitable catalyst was the only obstacle. Platinum was known to catalyse the exchange in the vapor phase, but the low-solubility of hydrogen in liquid water produced almost zero catalytic activity when water was present. This impasse was resolved by invention of special wetproofed platinum catalysts, devised and extensively developed by AECL. A catalyst of this type is illustrated in Figure 4.7. It is composed of catalyst-coated hydrophobic plates where water vapor and hydrogen are able to exchange deuterium and hydrophilic plates where the water vapor can come to equilibrium in deuterium content with liquid water.

FIGURE 4.7
Structured wetproofed catalyst for water–hydrogen exchange.

FIGURE 4.8
Prototype CIRCE plant at Hamilton, Ontario, Canada.

Based on this development, AECL built a three-stage prototype plant at a small hydrogen plant owned by Air Liquide in Hamilton, ON (Figure 4.8). The prototype characterized and proved (1) a monothermal first stage in which the steam-methane reformer did the conversion of deuterium-enriched hydrogen; (2) a second stage of enrichment using a bithermal hydrogen–water process; and (3) a final stage of enrichment using a further monothermal process with water electrolysis converting water into hydrogen. Electrolysis is an energy-intensive process but can easily be made leak-tight and so is particularly suited to handling the high-concentration heavy water product. Because the flows in this final stage are small, electrical energy consumption by electrolysis is not a large consideration. This combination of hydrogen–water processes is now AECL's preferred technology for heavy water production. It is called Combined Industrial Reforming and Catalytic Exchange (CIRCE).

A similar application of the monothermal stage with water electrolysis is now the reference design for "upgrading" in-service or recovered heavy water that has been contaminated with ingress of ordinary water. This process is called Combined Electrolysis and Catalytic Exchange (CECE). Should large-scale production of hydrogen displace large-scale hydrogen production by steam-methane reforming—quite likely with the rising price of natural gas used in steam-methane reforming and the costs that will be associated with sequestration and storage of carbon dioxide—production of heavy water entirely by CECE would become economic and would be simpler than CIRCE.

4.3.3 Tritium

Although deuterium's superiority as a moderator arises from its resistance to absorbing neutrons, neutron absorption does very occasionally occur. This produces the third, atomic mass 3, isotope of hydrogen. This is known as tritium and designated T—only the isotopes of hydrogen have their own accepted chemical symbols. Tritium has a half-life of 12.3 years and decays with a very weak beta emission. In a CANDU moderator, the concentration of tritium rises over many years to equilibrium at around 25 ppm. The moderator is a low-pressure system at about 75°C and can be made very leak-tight. Where heavy water is also used as the reactor coolant (which has been the norm but will not apply to the ACR), providing near-absolute leak-tightness is not possible. So a system of high-performance dryers is installed in zones where leakage is likely to occur. This water is usually mixed with some inleakage of normal water and so has to be upgraded.

4.3.4 Upgrading to Remove Light Water

Upgrading of this recovered water has always been done by water distillation at sub-atmospheric pressure. This is a very simple process with no moving parts: water is boiled below an exchange column and condensed above the column. The equilibrium constant between liquid and water vapor is, however, very small (ranging from 1.055 at 14 kPa (abs) to 1.035 at 50 kPa (abs) – the practical range used in water distillation. As a consequence, hundreds of contact steps are needed to re-enrich the heavy water and strip deuterium from the vapor stream so that it can be discarded. However, even though the internal flows within the exchange column must be more than orders of magnitude larger than the feed flow, they remain small in absolute terms and are economically manageable. The columns are rather large, typically around 50-m long and 0.8 m in diameter.

As mentioned previously, water distillation is now considered superseded by the CECE process, which is far more compact and cheaper.

4.3.5 Tritium Extraction

Most operators of CANDU reactors have chosen to apply tritium extraction (also known as "detritiation") after their reactors have operated for some years, and tritium levels have risen some way toward equilibrium. This not required for considerations of environmental release but lower levels can simplify reactor maintenance.

For detritiation, variations of water–hydrogen exchange have been used for primary tritium extraction. The CECE process can be used to effect some further tritium enrichment, but above around 300 ppm water becomes sufficiently tritiated as to constitute a radiological hazard if direct skin contact occurs. In the elementally form, tritium is many orders of magnitude less hazardous because the human body neither absorbs not significantly retains it. Consequently, cryogenic distillation of hydrogen isotopes is employed to produce further enrichment, which can extend to virtually pure tritium. This low-temperature distillation process operates at 22–24 K, but has a good equilibrium ratio (about 1.4).

4.4 HWR Safety

4.4.1 Background

The nuclear safety philosophy and the regulatory licensing processes for heavy water pressure tube reactors developed relatively independently of other jurisdictions and, in part, were driven by the unique aspects of HWR design. Early operational experience with research reactors led to the requirements for functional and physical separation of special safety systems from process systems, the requirement for fast-acting shutdown systems, the requirements for demonstrating high availability and testing of passive safety systems, and the incorporation of an elementary risk-based licensing framework. All of these requirements were encapsulated at a high level in a licensing guidance developed in Canada and referred to as the "Siting Guide" (Hurst and Boyd 1972) which provided a simple and relatively effective licensing framework. Central to the Siting Guide framework were three concepts that in retrospect align well with the concept of the five levels of defence-in-depth scheme articulated by INSAG (IAEA 1996).

First, the process systems should be of high quality to limit the frequency of failures that could lead to accidents and the special safety systems should be highly reliable such that their unavailability is a low-probability condition. This corresponds to level one in the five-level defence-in-depth scheme.

Second, the "*safety systems shall be physically and functionally separate from the process systems and from each other*." This ensured clear separation between level two and three defence-in-depth provisions. The process systems, providing level two defence-in-depth, include all systems necessary for control of the plant during normal power operation and shutdown conditions and for equipment and component protection, such as the reactor regulating system, boiler pressure and feedwater control, shutdown cooling, moderator and end-shield cooling, service water, electrical power and instrument air. The safety systems, providing level three defence-in-depth, include the reactor shutdown system (SDS), the ECCS and containment. Additionally, a requirement was imposed to demonstrate high system reliability through on-line testing aimed at mitigating against lack of operating experience with any new design.

Third, the concept of risk was introduced in an elementary manner by requiring lower dose consequence limits for more probable failures (single failure of a process system) and applying higher dose consequence limits for less probable events (dual failures involving failure of a process system and coincident unavailability of a special safety system).

4.4.2 Basic Nuclear Safety Functions

Similar to other thermal reactor designs, there are three basic functions that are necessary to mitigate the consequences of fission product releases during a postulated accident. The functions, referred to as the "**3 Cs**," are Control, Cool and Contain. "Control" refers to safe reactor shutdown. "Cool" involves the removal of heat—from the fuel produced by the fission process (at power) or by the decay heat after reactor shutdown—and rejection of the heat to a heat sink. "Contain" is simply the physical means to prevent the release of radioactive material to the atmosphere by provision of containment systems.

4.4.3 Reactor Shutdown

The majority of pressure-tube HWRs have protective functions in the reactor regulating system—a normally operating process system—that reduce reactor power when required to maintain process conditions in a safe operating range and provide protection of components and equipment. The power reduction can be gradual (power setback) or rapid due to absorber rod drop into the core (stepback).

In addition to the reactor regulating system provisions, current HWR designs have two diverse, independent, fast-acting, equally effective and fail-safe safety shutdown systems, referred to as Shutdown System 1 (SDS1) and SDS2 (Figure 4.9).

SDS1 utilizes spring-loaded mechanical shutoff (absorber) rod mechanisms. Upon receipt of a reactor trip signal, an electromagnetic clutch in each mechanism is de-energized, releasing a stainless steel-clad cadmium absorber element that drops into the moderator under gravity, with initial rod acceleration provided by spring thrust. SDS1 is the primary method of quickly shutting down the reactor in an accident.

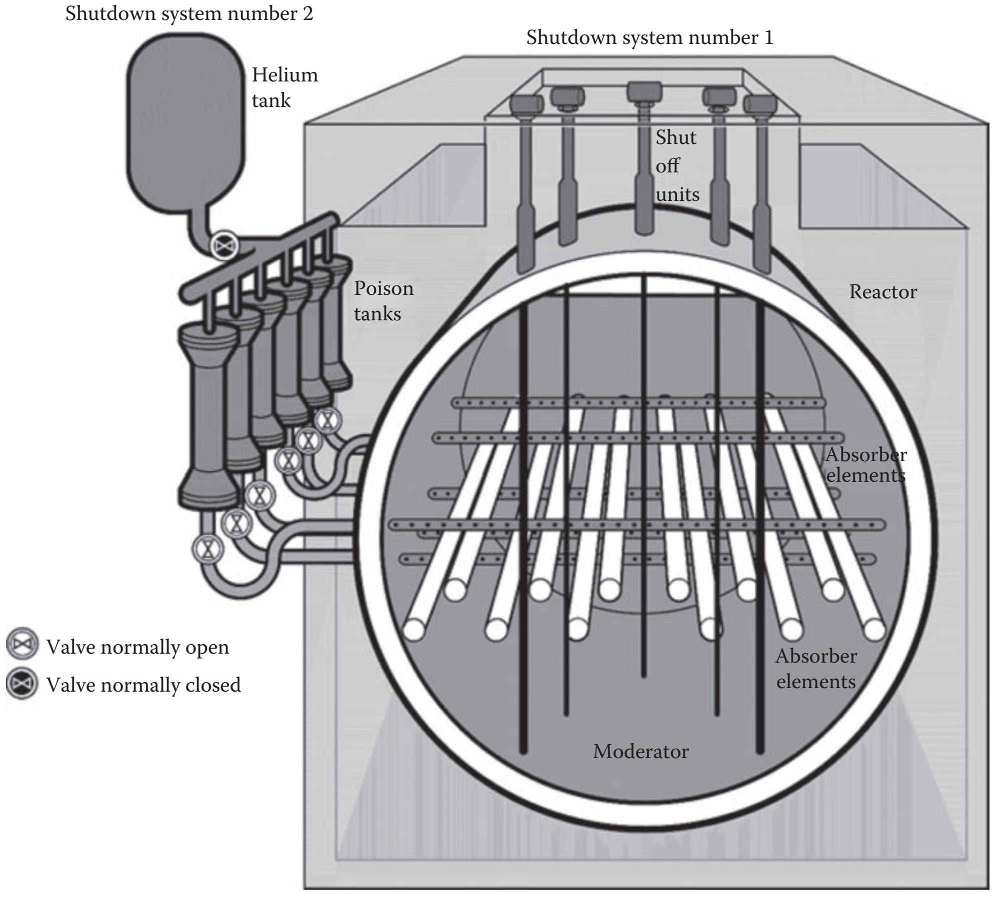


FIGURE 4.9
Shutdown systems.

SDS2 utilizes liquid poison (absorber) injection into the moderator. Upon receipt of a reactor trip signal, fast-acting valves between a high-pressure helium tank and the poison tanks open to pressurize and inject the liquid poison into the moderator. Injection occurs from several perforated horizontal tubes in the calandria, through which gadolinium nitrate solution jets into the moderator. An earlier design, the 220-MWe Indian HWRs, employs a set of vertical empty tubes in the reactor core that can be filled with a liquid poison (lithium pentaborate solution).

Reactor shutdown is initiated by several trip parameters selected to ensure that there are at least two parameters on each shutdown system to detect any serious malfunction requiring a reactor shutdown. Triplicated sensors and instrumentation channels, independent of those used in regulation, are provided for each of the trip parameters. The sensors and instrumentation channels of the two shutdown systems are separate and, to the extent practicable, employ diverse components and designs. Typical trip parameters are: high reactor power, HTS high and low pressure, HTS high-temperature, low-flow in the HTS, low-pressure differential across the reactor core, high containment pressure, normal electric supply failure, and low-level in the steam generators.

Independent triplicated logic is employed in each system, with two out of three coincidence logic for generating the trip signal. Each trip channel is testable on power. In SDS1 and SDS2, loss of electrical power to the shutdown systems will result in the reactor shutdown mechanisms being deployed to ensure fail-safe operation, either by disengaging the electromagnetic shutoff rod clutches or opening the fast-acting valves to pressurize the liquid poison tanks.

4.4.4 Heat Sinks

The normal heat sink for the fuel in the core is provided by forced circulation of HTS coolant, which transfers the core heat to the secondary coolant in the steam generators, with feedwater supplied to the steam generators by steam generator feedwater pumps. Following shutdown, the core decay heat can normally be removed through two alternate independent and diverse paths: through steam generators, with heat rejected by boiling off feedwater, or through the shutdown cooling system, with heat rejected to process/service water, which ultimately rejects heat to atmosphere (through cooling towers) or to cooling water from the ultimate heat sink (river, lake, ocean).

Under reactor shutdown conditions, cooling of the core to remove decay heat can be performed by natural circulation of the HTS coolant through the steam generator tubes (i.e., forced circulation is no longer necessary). On the secondary side, feedwater flow to the steam generators, at a substantially reduced rate (about 4% of normal feed flow), is provided by auxiliary steam generator feed pumps. The power source for these pumps depends on the station design: some are electric, using Class III (diesel generator) power; some stations use steam-driven pumps; and some use direct diesel drive. In the event of failures in the secondary side (either in the feedwater or main steam supply systems) additional safety-related systems, such as the steam generator/boiler emergency coolant system or the emergency water system are available to provide separate water supply to the steam generators to maintain the heat sink. The Emergency Water System is usually a seismically qualified system.

For certain accident conditions, other cooling paths are also provided. During a LOCA, the ECCS is used to refill the core and remove decay heat from the reactor. High-pressure injection is supplied by a system of accumulators containing water and pressurized by nitrogen gas tanks; or by high-pressure pumps in some plants. Intermediate pressure injection and long-term recovery and recirculation is provided by pumps (powered by Class III electric supply) with water drawn initially from a tank, and subsequently from a sump (which collects spilled water from the break) via the ECCS heat exchanger. The heat picked up by ECCS water is rejected to process water in ECCS heat exchangers. In current HWRs, the entire sequence is automated, whereas in some older HWRs operator action is required to switch from intermediate pressure injection to recovery mode.

ECCS is accompanied by "crash cooldown" of the steam generators, involving blowing off steam to atmosphere through the Main Steam Safety Valve. This ensures that the HTS pressure stays below ECCS injection pressure, especially for small LOCA, and also for large LOCA in the long-term.

The minimum design objective for the ECCS is to limit the release of fission products from the fuel. While specific acceptance criteria may differ from country to country, typical requirements in this regard are listed below.

- For LOCAs with break size smaller than, and up to, the largest feeder break, there shall be adequate cooling of the core to prevent gross fuel sheath failures. (However, in single channel events, failure of fuel in the affected channel may not be prevented.)
- For LOCAs larger than feeder pipe breaks, fuel failures shall be limited such that the radiological consequences to the public are within limits acceptable to the regulator for this class of event.

- For all LOCAs, the integrity of fuel channels shall be maintained; and fuel geometry shall allow continued coolability of the core by ECCS.
- Adequate long-term cooling capability of the fuel following the LOCA shall be ensured.

A unique feature of pressure tube HWRs is large volumes of heavy water or light water surrounding the fuel channels and the calandria vessel, respectively. These water volumes provide an inherent means for removal of decay heat from the core during BDBA that progress to severe core damage. The two water sources are the heavy water moderator surrounding the fuel channels and the light water shielding surrounding the calandria vessel. The moderator provides an effective heat sink to prevent the development of severe core damage during LOCA events with failure of ECCS. The shielding water surrounding the calandria vessel provides an inherent heat sink in the event that moderator system cooling is lost, causing moderator boil-off and subsequent disassembly of fuel channels inside the calandria vessel. Both of these inherent heat sinks are normally in place and require no operator actions under severe accident management guidelines (SAMG) to activate them. They provide important passive means that can stabilize core damage during severe accidents or significant time delay during severe accident progression to allow alternative SAMG candidate actions to be undertaken.

4.4.5 Containment

The chief function of the containment system is to limit the accidental release of radioactivity to the environment to within acceptable limits. The containment system consists of a leak-tight envelope around the reactor and associated nuclear systems, and includes a containment isolation system (for fast closure of valves/dampers in lines penetrating the containment), containment atmosphere energy removal (cooldown) systems, and clean-up systems. Hydrogen control is provided in the newer, larger HWRs to cater for thermochemical hydrogen generation from the zircaloy–steam oxidation reaction due to overheated fuel and from long-term hydrogen build-up due to radiolysis after a LOCA.

Current pressure-tube HWR designs employ the following major types of containment:

(a) Single unit containment, with a dousing system for pressure suppression, as used in CANDU 6
(b) Multiple unit containment with a common vacuum building, as used in the Pickering, Darlington, and Bruce stations
(c) Double containment system used in Indian HWRs, incorporating a double envelope, and pressure suppression

These concepts are described briefly in the following sections. Note that the HWR design does not determine the type of containment; that decision is driven by other factors such as single vs. multiple unit philosophy, national regulatory requirements, allowable leak rates, and construction cost.

4.4.6 Single-Unit Containment System

The CANDU 6 single-unit containment consists of a cylindrical pre-stressed, post-tensioned concrete building with a concrete dome (Figure 4.10). The concrete provides strength and

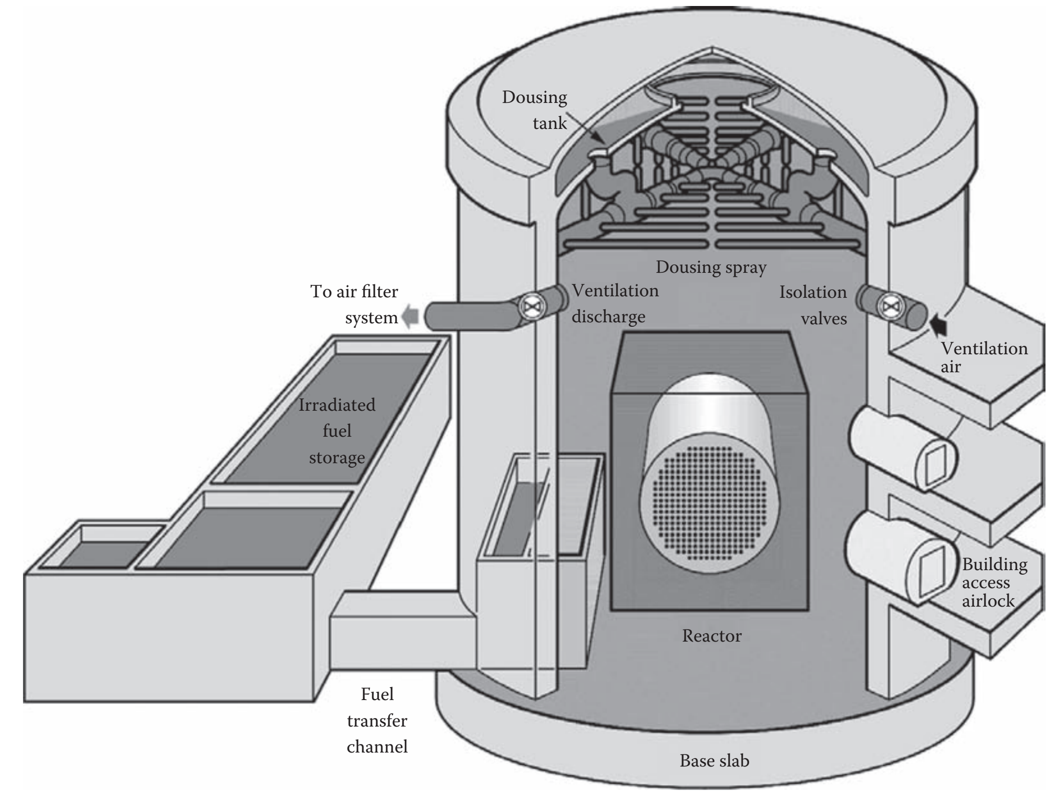


FIGURE 4.10
Single unit containment.

shielding; the building is lined with an epoxy coating to improve leak-tightness. Beneath the outer dome there is an inner dome having an opening in the crown. The double dome together with the perimeter wall forms a container, providing storage at an elevated level for water for dousing and emergency core cooling.

On a rise in pressure or a release of radioactivity to the containment, the containment isolation system would close all penetrations open to the outside atmosphere, mainly the containment ventilation system. This is a subsystem of the containment safety system.

A sufficiently large pressure rise (e.g., from a LOCA or steam line break) would trigger the dousing spray system through valves in the dousing spray headers. The purpose of the dousing spray is to suppress the short-term pressure rise caused by the accident, thus the flowrate is very high. Dousing turns on when the overpressure rises to >14 kPa, and turns off when it falls to <7 kPa, resulting in a cyclic operation for small LOCA. Operation of redundant dousing valves maintains the pressure following a LOCA below the containment design pressure.

4.4.7 Multi-Unit Containment System

In the multi-unit vacuum system, employed in the CANDU stations in Ontario, 4–8 reactors, each with its own local containment, are connected by large ducting to a separate, common vacuum building kept, as its name implies, at near zero absolute pressure. Should steam be released from a pipe break in the reactor building, the pressure causes banks of self-actuating valves connecting the vacuum building to the ducting to open. Steam and radioactivity are then sucked along the duct; the steam is condensed by dousing in the vacuum building and soluble fission products such as iodine are washed out. The dousing is passively actuated by the difference in pressure between the main body of the vacuum building and the vacuum chamber; it does not require electrical power or compressed air supplies to operate. This concept, which was developed because of the economics of multi-unit sites, has several unique safety characteristics:

- After an accident, the entire containment system is sub-atmospheric for several days; thus leakage is inward, rather than outward. This is true (for less time, of course) even with an impairment in the containment envelope.
- The overpressure period in the reactor building is very short, of the order of a couple of minutes, so the design pressure is reduced and the design leak-rate can be increased relative to single unit containment.
- Even if the vacuum building is not available, the large interconnected volume of the four or eight reactor buildings provides an effective containment.
- Several days after an accident, when the vacuum is gradually depleted, and the containment pressure rises toward atmospheric, an Emergency Filtered Air Discharge System (EFADS) is used to control the pressure and ensure the leakage is filtered.

4.4.8 Containment System of Indian HWRs

Current Indian HWRs use a double containment design. The annular space between the primary and secondary containment envelopes is provided with a purging arrangement to maintain a negative pressure in the space. This arrangement will significantly reduce the ground level releases to the environment during accidents where there is a radioactivity release into the primary containment.

4.4.9 Safety Analysis

The basic purposes of safety analysis are to assist in the design of safety-related systems, and then to confirm that the radiation dose limits are met. As such, safety analysis requires predictions of the consequences of hypothetical accidents.

In most cases, the approach to HWR licensing has been performance-based rather than prescriptive. That is, the regulator sets overall requirements on the classes of accidents to be considered, and on the public dose limits as a function of accident class, but leaves it up to the licensee to a large extent to determine how best to meet the requirements and limits. In particular most HWR regulators do not specify the design requirements in great detail, nor do they specify prescriptive assumptions on accident analysis methods.

Safety analysis covers a wide range of initiating event failures and combinations of coincident or subsequent failures of process and safety systems. In addition, mitigating process system actions are normally not credited in demonstrating the effectiveness of the safety systems.

A unique aspect resulting from the application of the Siting Guide is that the design basis analysis has included a class of events called dual failures. A dual failure is defined as the simultaneous failure of a process system and the unavailability of a safety system or subsystem. Safety analysis is therefore performed for the failure of each process system in the plant; then for each such failure combined with the unavailability or impairment of each relevant safety system or subsystem in turn. Examples of major impairments are

- Unavailability of one of the two shutdown systems (always assumed)
- No emergency coolant injection
- No containment isolation
- Failure of dousing
- Deflated airlock door seals
- Failure of vault coolers
- Partial or total loss of vacuum (vacuum containments)

Dual failures in HWR are in many respects equivalent to severe accidents considered for other reactor types.

4.4.10 Safety Analysis Scope

The general scope of safety analysis for HWRs, covering the accident categories considered, safety barriers challenged and the technical disciplines involved are summarized in Table 4.2. Failures in safety support systems (such as instrument air) are addressed in the PSA.

HWR accident analysis practice has been to use physically realistic models of the system behavior, with conservatisms incorporated in assumptions on input parameters, and

TABLE 4.2

Scope of HWR Safety Analysis

Accident Category	Barriers Challenged	Technical Discipline
Loss of regulation	Fuel sheath HT system boundary	Reactor physics System thermalhydraulics
Loss of reactivity control	Fuel sheath HT system boundary	Reactor physics System thermalhydraulics
Loss of HT flow	Fuel sheath HT system boundary	System thermalhydraulics Reactor physics
Loss of HT coolant (Small LOCA AND single channel events)	Fuel sheath containment	Reactor physics System thermalhydraulics Containment thermalhydraulics Fuel & fuel channel thermal-mechanical behavior Fission product release & transport
Loss of HT coolant (large LOCA)	Fuel pellet Fuel sheath containment	Reactor physics System thermalhydraulics Moderator thermalhydraulics Containment Thermalhydraulics Fuel & fuel channel thermal-mechanical behavior Fission product release & transport
Feedwater system failure	Fuel sheath HT system boundary	System thermalhydraulics Containment thermalhydraulics
Steam supply system failure	Fuel sheath HT system boundary	System thermalhydraulics Containment thermalhydraulics Fission product release & transport

operator response. This requires relatively detailed models in the technical disciplines identified above.

4.4.11 LOCA

The LOCA imposes the most severe challenge to all three safety systems (shutdown, emergency core cooling, and containment) and sets many of their design requirements (Luxat 2003). The LOCA are categorized according to the magnitude of the pipe rupture and the resultant process systems response.

A very small LOCA (or leak) is defined as having a break discharge flow rate that can be handled by the heavy water makeup system without the need for any safety system intervention. A small break LOCA is defined as a pipe break that cannot be compensated by the heavy water makeup system and extends multiple feeder pipe ruptures such that the reactor regulating system, without credit for stepback action, is capable of limiting any power excursion. A large break LOCA is defined as a pipe break beyond the range of breaks in multiple feeder pipes which give rise to uncompensated coolant void reactivity and a resultant power excursion.

For large and small LOCAs, one of the shutdown systems, the emergency core cooling and containment are all required and are initiated automatically or, at the lower end of the range, by the operator. For very small LOCAs, shutdown may be manual or automatic; emergency core cooling and containment are not required, and might not be initiated automatically.

4.4.11.1 Small Break LOCA

Because of the total length of feeders and pressure tubes, a small break is about 100-times more probable than a large break. Clearly this range is analyzed for both economic and safety considerations.

The requirement for ECCS for small breaks is to limit or prevent fuel damage, mainly for economic reasons. For breaks up to an equivalent area of the severance of several feeder pipes, the pumps are much more influential in determining channel flow than the break. Thus the flow is always forward or recirculating, although as steam quality builds up with time, the pump head decreases and the resistance of the circuit increases; so the magnitude of the flow falls with time.

The first requirement is to shut down the reactor before fuel sheaths experience prolonged dryout at high power. Prevention of dryout is sufficient but not necessary to prevent fuel sheath failure. Shutdown occurs on process parameters signals such as low-pressurizer level, low-storage tank level, low-flow, low-pressure, low-core pressure drop or high-pressure within containment. For a feeder size break, shutdown is initiated within the first three or four minutes. As the circuit continues to empty after shutdown, the flow in the headers or channels eventually falls low enough that the coolant phases separate. This could result in steam cooling of some of the upper fuel elements in a channel or of some of the channels connected to the mid-plane of the header. Prolonged stratification can lead to sheath damage in the order of a few minutes, so this defines the time at which ECC must become effective. Once refill has occurred, the pumps maintain recirculating flow, and this pattern continues into the long-term until the pumps are tripped. Pump trip has been automated on certain HWRs in the longer-term, after ECC refill, to avoid pump cavitation once the circuit has refilled with cold water.

The break is a major heat sink for decay heat. Breaks greater than the cross sectional area of a feeder pipe can remove all the decay heat from one HTS loop. The steam generators, however, if not cooled, can hold-up HTS pressure; thus the steam generators must be cooled down fast enough to ensure that ECC injection can proceed. Crash cooldown takes the steam generators from normal operating pressure to close to atmospheric in about 15 minutes and this allows continued ECC makeup flow.

Special cases of a small break LOCA are those involving failures in single fuel channels. Such events include, pressure tube ruptures, feeder stagnation break, channel flow blockage, in-core LOCA involving failure of pressure tube and calandria tube, and failures of fuel channel end-fittings, The in-core LOCA introduces additional phenomena such as interaction of the broken channel with neighboring channels and the reactivity mechanism guide tubes. All of these failures can damage fuel: the first by mechanical damage following the pressure tube rupture; the second and third by overheating due to reduced flow, the fourth by mechanical damage following the channel rupture and ejection of the fuel into the calandria vessel; and the last by ejection of the fuel into the calandria vault, followed by mechanical damage and oxidation in the vault atmosphere.

A severe flow blockage >90% of the channel flow area is required to cause pressure tube failure due to overheating. A single channel event leading to channel failure also requires analysis of the pressure transient within the calandria, to show that the calandria vessel itself remains intact, that the shutdown system devices within it can still perform their function, and the break does not propagate by causing failures of other reactor channels.

For a steam generator tube failure (categorized as a leak) it is first necessary to identify the failure by detecting the leakage using a D_2O-in-light water detection system. Because a single steam generator tube failure is within the capability of the D_2O makeup system, fuel cooling is not at risk. The operator will shut down the reactor and depressurize the HTS, thereby stopping the leak to the secondary side. The HTS can then be drained to below the level of access to the lower steam generator head so that the tube can be isolated (plugged). Cooldown and fuel cooling is achieved by using the shutdown cooling system. The shutdown cooling system can remove decay heat at full system pressure, so the operator is not dependent on the secondary side for depressurization/cooldown.

4.4.11.2 Large Break LOCA

The range of HTS behavior is encompassed by large pipe ruptures at three locations: at the core inlet (inlet header), at the core outlet (outlet header), and upstream of a main heat transport pump (pump suction pipe). Breaks at these locations affect the two core passes of a loop in different ways. The core pass upstream of the break (upstream and downstream directions are defined relative to the normal flow direction) always has flow that is accelerated toward the break. The fuel cooling tends to be increased and emergency core coolant refill is rapid in the flow direction toward the break. The core pass downstream of the break has its flow reduced by the flow that is diverted out of the break and is therefore more likely to experience degraded fuel cooling.

A doubled-ended guillotine rupture of an inlet header reverses the flow in the downstream core pass. On the other hand, a small break at the inlet header will maintain the flow in the normal flow direction. Therefore, it is possible to select a break size that leads to a period of sustained very low-flow in the downstream core pass. This low-flow arises from a balance between the break flow and the flow delivered by the upstream pump. Such breaks, referred to as “critical breaks,” tend to more limiting with respect to cooling of the fuel and fuel channels than other large breaks and are therefore analyzed in detail. After a short period of about ≤30 seconds of very low-channel flows in the affected pass, voiding at the pump suction degrades the pump head causing channel flows in the downstream pass to reverse toward the break.

Flows in the long-term are determined by the balance between the break and the pumps (which may be tripped at some point in the accident). At the lower end of the large break spectrum ECCS refill flows and long-term flows will be in the forward direction. If the break is larger, ECCS refill will be in the reverse direction and this flow direction will persist into the long-term. Flow for intermediate breaks may reverse when the pumps are tripped.

Breaks at the outlet end of the core (outlet header break) will cause increased flow in the upstream pass and reduced flow in the downstream pass. Large outlet header breaks may be able to reverse the flow in the downstream core pass. For the largest outlet header break the voiding of the downstream core pass is slower than for inlet breaks because the path from the break to the core is longer and the resistance is higher. Thus, when sustained low-flow does occur there is less stored heat in the fuel. Fuel temperature increases during sustained low-flow are lower than for the inlet header case; however ROH breaks are limiting for sheath strain failures because the sheath temperature is high when the coolant pressure is low. Smaller outlet header breaks allow continued forward flow: ECCS refill is in the forward direction and a long-term recirculating flow pattern will occur. At first the flow goes backwards through the downstream pump but as the circuit depressurizes the pump acts more like a check valve. Injection water into the inlet header of the downstream core pass is therefore prevented from going through the pump to the break and instead is forced in the forward direction through the core pass. Refill of both core passes is in the forward direction. The long-term flow pattern for the largest reactor outlet header break is recirculating until the pumps are tripped; then flows are directed in each pass toward the break. Pump suction breaks are hydraulically similar to reactor outlet header breaks.

4.4.11.3 Analysis Methods

As shown in Table 4.2, large break LOCA events involve the most physical phenomena and, therefore, require the most extensive analysis methods and tools. Typically, 3D reactor space–time kinetics physics calculation of the power transient is coupled with a system thermalhydraulics code, to predict the response of the heat transport circuit, individual channel thermal-hydraulic behavior and the transient power distribution in the fuel. Detailed analysis of fuel channel behavior is required to characterize fuel heatup, thermochemical heat generation and hydrogen production, and possible pressure tube deformation by thermal creep strain mechanisms. Pressure tubes can deform into contact with the calandria tubes, in which case the heat transfer from the outside of the calandria tube is of interest. This analysis requires a calculation of moderator circulation and local temperatures, which are obtained from computational fluid dynamics (CFD) codes. A further level of analysis detail provides estimates of fuel sheath temperatures, fuel failures and fission product releases. These are inputs to containment, thermal-hydraulic and related fission product transport calculations to determine how much activity leaks outside containment. Finally, the dispersion and dilution of this material before it reaches the public is evaluated by an atmospheric dispersion/public dose calculation. The public dose is the end point of the calculation.

Traditionally, a "conservative" approach to safety analysis has been employed. In this approach, pessimistic assumptions, bounding input data and even conservative physical models are used to obtain a pessimistic bounding analysis. This approach has been required to a greater or lesser extent by most regulators and for most reactor types. HWRs also followed this approach for licensing analysis, although the physical models used were realistic.

The advantage of the approach was that the answer was known to be pessimistic; and in some cases the safety analysis could be simplified by using bounding rather than realistic assumptions (as long as the results were still acceptable). There are a number of disadvantages to the conservative analysis, namely:

- The margin between the expected behavior and the conservative predicted behavior is unknown (how "conservative" the answer is).
- There is a tendency over time for more and more conservatisms to be added in, in an unsystematic fashion.
- As the conservative prediction gets close to the regulatory acceptance limit, regulators become uncomfortable at the apparent lack of margin, despite the conservatisms.
- The predictions of the computer codes can yield physical conditions in areas where validation is impossible (e.g., very high fuel temperatures).

Recently, however, limited use of "best estimate plus uncertainty analysis" methods has been undertaken. This is consistent with the international trend toward use of such methods. In this approach, more physically realistic models, assumptions, and plant data are used to yield analysis predictions that are more representative of expected behavior. This requires a corresponding detailed analysis of the uncertainties in the analysis and their effect on the calculated consequences. Typically, the probability of meeting a specific numerical safety criterion, such as a fuel centerline temperature limit, is evaluated together with the confidence limit that results from the uncertainty distributions associated with governing analysis parameters. The "Best Estimate Plus Uncertainties" approach addresses many of the problematic issues associated with conservative bounding analysis by:

- Quantifying the margin to acceptance criteria
- Allowing rational combination of uncertainties
- Evaluating "cliff-edge" effects
- Highlighting parameters that are important to safety
- Focusing safety Research and Development on areas of true importance
- Providing the basis for more realistic compliance monitoring

4.4.12 Severe Accidents

Severe accidents may be defined simply as accidents for which heat removal from the reactor fuel is insufficient to remove the heat generated for a sufficiently prolonged period of time such that damage occurs to fuel or structures within the reactor core. Despite the existence of engineered safety systems the possibility exists (albeit at a very low level of likelihood) that an accident can progress beyond the acceptable design basis envelope and develop into a severe accident. The severity of such accidents can be characterized by the nature and extent of core damage that occurs during the progression of an accident.

The progression of events to an accident with severe fuel or core damage in an HWR involves several broad stages in which thermal-hydraulic behavior of the reactor fuel, fuel channels, HTS and a number of key process systems govern the rate at which severely degraded cooling conditions develop and the extent of resultant damage to the reactor core (Blahnik, Luxat, and Nijhawan 1993; Luxat, J.C. 2007; Rogers et al. 1995). As indicated previously, the moderator can provide an inherent passive heat sink in the event of a LOCA with failure of the ECCS. In this case, if moderator cooling can be assured then the reactor damage state is limited to severe fuel damage with fuel channels remaining intact. Should moderator system cooling be lost and moderator boiloff occur, then the fuel channels will disassemble in the calandria vessel forming a debris bed, including the formation of a molten corium pool within a solid debris crust. If shield cooling water can be maintained on the outside of the calandria vessel such that adequate heat transfer from the vessel wall can be assured, then the core debris can be retained in the vessel (in-vessel retention) and further progression of the accident is terminated (Luxat, J.C. 2007; Muzumdar et al. 1998). However, if in-vessel retention cannot be assured (Luxat, D.L. et al. 2007), then ex-vessel debris coolability issues and the adequacy of containment heat sinks have to be addressed to demonstrate that containment integrity is not impaired.

The unique inherent and passive heat sink design features of HWRs result in severe accidents that are expected to progress with ample opportunity for operator actions to stabilize the plant and mitigate the consequences.

4.5 Beyond the Next Generation CANDU: CANDU X Concepts

4.5.1 Introduction

In AECL's view, the next generation of pressure-tube HWRs encompasses an evolutionary set of technologies that lead ultimately to the "CANDU X." The CANDU X is not a specific reactor design, but embodies concepts extrapolated from the current knowledge base that we believe can be achieved in our development programs over the next 25 years. Therefore, CANDU X is a changing set of technologies and targets, with the overall goal of a further 50% cost reduction beyond the best current technology at any point in time (Spinks, Pontikakis, and Duffey 2002).

One element of AECL's development program is to continue to improve plant economics by increasing thermal efficiencies. This will require development of materials and systems that can withstand the higher temperatures and pressures that this improvement entails. The first step in this long-term evolution is a reactor utilizing supercritical water (SCW) as the HTS coolant. This will require HTS components that operate at about 430°C and 25 MPa and will boost the thermal efficiency to about 40%. The ultimate goal is to improve the efficiency to about ≥50% by reheating the coolant to 625°C at 5–7 MPa and operating the HTS under mixed supercritical and subcritical channel conditions. Such a system could make use of existing small direct cycle turbines with single reheat, currently used in thermal power plants, located inside the containment building to generate electric power, and a steam generator with an external turbine/generator outside containment to produce additional power (Figure 4.11).

Pressure-tube reactors like CANDU are very amenable to using SCW. SCW coolant cycles result in substantial coolant density variations (particularly if the water temperature crosses the critical temperature in the core). There can be a density change (by as much as a factor of seven) through the core, complicating flux gradients and flux shaping requirements. These complications are less important in pressure-tube type reactors for two reasons. First, because the moderator is located in the calandria vessel and is separated from the coolant,

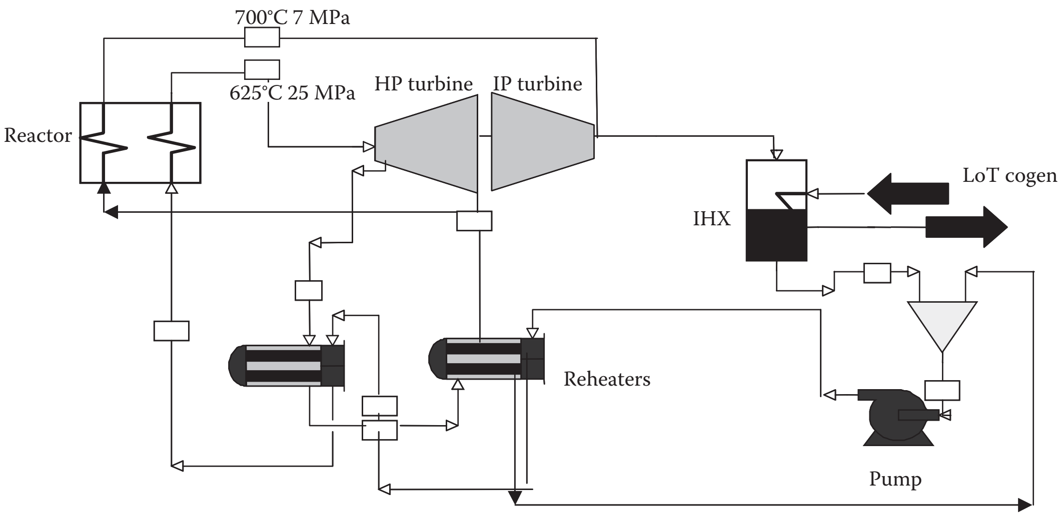


FIGURE 4.11
CANDU CWR (schematic).

the coolant has relatively less effect on the neutronics. Second, depending on orientation, the channel flows can be bi-directionally interlaced (opposite flow direction in adjacent channels) or can use re-entrant flow paths, so the density gradients are balanced and a more axially uniform flux profile is achievable.

Another major reason why pressure-tube reactors are suitable for SCW coolant is their ability to adapt the pressure boundary to accommodate much higher pressures. At the 25-MPa pressures required for SCW coolant, there would be challenging requirements for developing a large-pressure vessel. For pressure-tube reactors, it will be far easier to meet the requirements by evolving the design of the fuel channel. A new, insulated fuel channel concept is being developed in which the high-pressure boundary is kept at a relatively low temperature where the material strength is higher. This fuel channel concept will experience lower creep strains during aging (even with SCW coolant) than the current fuel channel design. Pressure-retaining components can be tested directly at full-scale, which greatly facilitates development of the SCW coolant technology for the CANDU system.

Heat transfer under supercritical conditions has already been widely investigated (Pioro, Khartabil, and Duffey, 2003) provides a literature survey. The IAEA is co-ordinating an international cooperative research program on heat transfer behavior and thermo-hydraulic code testing for SCWRs, which is intended to lead to issue of a TECDOC report.

Figure 4.12 shows the CANDU X concept co-producing hydrogen, process heat and distilled water alongside electricity. Hydrogen can be produced by conventional low-temperature electrolysis, but thermo-chemical and high-temperature electrolytic processes are being developed. If these high temperature approaches prove to be economically superior, heat could be provided directly from the SCWR or, if higher temperatures are needed, by the use of re-entrant superheater tubes located in the reactor's periphery. Distilled water—either using heat to enhance reverse osmosis or for a distillation-based approach—requires very modest temperatures. This application is expected to become more widespread with rising pressures on water supply in many parts of the world and can easily applied with any reactor type. However, HWRs offer the novel possibility of using the moderator—in which about 5% of reactor heat is deposited—as the heat source.

Generation of electricity with supercritical steam is existing technology from fossil-fired plants. The distinctive challenges in developing the CANDU SCWR concern the materials

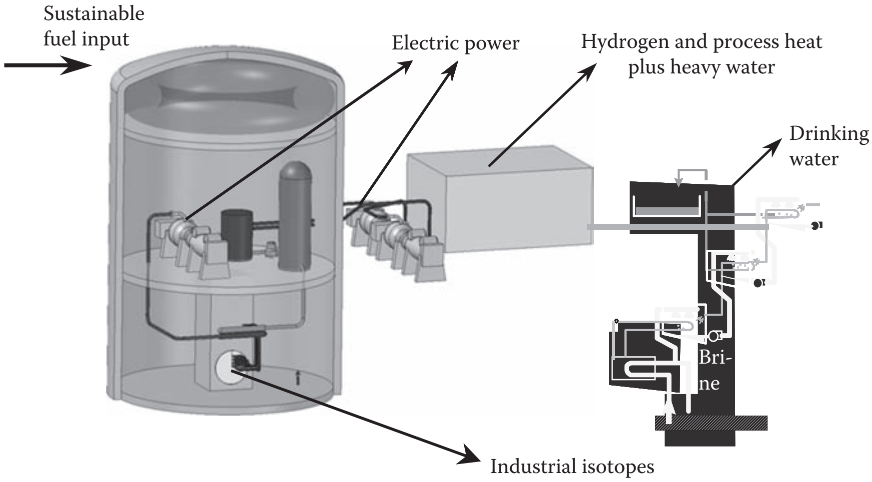


FIGURE 4.12
CANDU X concept co-producing hydrogen, process heat and distilled water alongside electricity.

that will be exposed to the high-temperatures of the supercritical steam and the intense neutron flux in the reactor's core. With the pressure tubes shielded by the insulator from high temperature, materials robust to the effects of very high-temperature and neutron flux are only required for the fuel cladding and the insulator.

Several variants of SCWR concepts have been proposed and are reviewed in (Duffey and Pioro 2006), and the major pressure tube variants are listed in the Table 4.3 from Canada and Russia.

4.5.2 Passive Safety: Eliminating Core Melt

A key element of future reactor designs is improved safety. The unique channel layout for pressure-tube designs uses the moderator as a backup heat sink for emergency heat removal and provides a "walk away" safety argument that requires no active sytems to activate or be operated (Vasic and Khartabil 2005). By testing passive moderator heat rejection systems, the most promising design is a passive, flashing-driven moderator cooling system. One of the main objectives of CANDU SCWR is to optimize the advanced fuel channel design to ensure that the passive moderator-cooling loop can remove reactor decay heat in the unlikely event of a LOCA combined with LOECC, i.e., normal cooling is lost and emergency heat removal systems do not activate.

Using the results of code simulations, it is possible to optimize the performance of the insulated fuel channel under decay heat generation conditions and variable thermal-physical characteristics of the insulating layer. This study shows that the advanced channel design is promising and can prevent overheating of the fuel even during very severe accident scenarios. The results show that advanced fuel channel design, combined with the passive moderator heat rejection and selection of the fuel-clad design have the potential to avoid core melting and to reduce significantly or even eliminate the possibility of core damage.

4.5.2.1 Beyond Uranium

Development of reactor technology to allow burning of a range of fissile fuels is another important avenue of HWR development.

TABLE 4.3

Typical Pressure Tube SCWR Parameters

Parameters	Unit	SCWR CANDU	ChUWR	ChUWFR	KP-SKD
Country	–	Canada	Russia	Russia	Russia
Organization	–	AECL		RDIPE (НИКИЭТ)	
Reactor type	–	PT	PT	PT	PT
Spectrum	–	Thermal	Thermal	Fast	Thermal
Power thermal	MW	2540	2730	2800	1960
electrical	MW	1140	1200	1200	850
linear max/ave	kW/m		38/27		69/34.5
Thermal eff.	%	45	44	43 (48)	42
Pressure	MPa	25	24.5	25	25
T_{in} coolant	°C	350	270	400	270
T_{out} coolant	°C	625	545	550	545
Flow rate	kg/s	1320	1020		922
Core height	m		6	3.5	5
Diameter	m	~4	11.8	11.4	6.45
Fuel	–	UO_2/Th	UC	MOX	UO_2
Enrichment	% wt.	4	4.4		6
Cladding material	–	Ni alloy	Stainless steel	Stainless steel	Stainless steel
# of fuel assemblies		300	1693	1585	653
# of rods/ FA		43	10	18	18
D_{rod}/δ_w	mm/mm	11.5 and 13.5	12/1	12.8	10/1
Pitch	mm				
T_{max} cladding	°C	<850	630	650	700
Moderator	–	D_2O	Graphite		D_2O

Source: Adapted from Duffey, R., I. Pioro and H. Khartabil, Supercritical Water-Cooled Pressure Channel Nuclear Reactors: Review and Status. Proceedings of GLOBAL 2005, Tsukuba, Japan, October 9–13, 2005, Paper No. 020.

A huge, long-term economic opportunity is opening up in the nuclear energy field with growing appreciation that nuclear fission is likely essential to any global program to stabilize atmospheric levels of greenhouse gases (GHGs). Alternative technologies—wind, solar, tidal, geothermal, hydrocarbon-combustion with CO_2 sequestration—all produce electricity more expensively than nuclear. Some are intrinsically intermittent; some are still in early development. In this situation and because the need to curtail GHG emissions is urgent, extensive nuclear deployment seems essential. Objective assessments suggest that nuclear deployment of 5000 to 10,000 new reactors—10–20-times the present number—will be needed to stabilize GHG concentrations by ~2050. This is in addition to the envisaged massive investments in renewable technologies such as wind and solar and to extensive conservation and efficiency measures (Miller, Suppiah and Duffey, 2006). Deployment on this scale will be difficult to sustain if it is based on today's technology, which is entirely dependent on uranium and predominantly uses once-through fuel cycles, cycles that typically extract only 0.6–0.7% of the energy available in the uranium resource. Higher reactor temperatures can help, but the issue is eliminated only if the industry develops fuel recycle and/or utilizes thorium.

Worldwide, there is renewed interest in nuclear power as a result of concerns not only about climate change, but also air pollution, energy security, and the cost and availability of fossil fuels. Nuclear power program decisions will be increasingly based on political, strategic and economic considerations involving the complete nuclear fuel cycle, including resource utilization, radioactive waste disposal, proliferation resistance, and supply assurances. The global nuclear industry needs to address each of these issues. The industry's long-term growth and its capacity for substantial GHG abatement will depend on following a path that addresses these issues by developing advanced fuel cycles and reactor designs optimized for such fuel cycles. The overall direction of the global power reactor development program should be refocused to provide a greater emphasis on integrated and complementary reactor and fuel cycle development, including the development and deployment of fuel enrichment and reprocessing technology and services.

Technical and policy developments in the field of energy that are linked and overlapping are detailed below.

- In the global energy context, the inexorably growing demand, increasing costs of alternative energy sources, and concerns about security of energy supply and environmental emissions of carbon dioxide and other GHGs, are all driving the need for more extensive deployment of nuclear energy worldwide. In 2007 October, the World Nuclear Association reported that 439 reactors with a total capacity of 372 GW were supplying 16% of the world's electricity; 33 (27 GW) were under construction; 94 (102 GW) were planned; and a further 222 (193 GW) were proposed. These large increases include several countries announcing plans to build and deploy nuclear reactors for the first time (e.g., Turkey, Egypt, Chile). Deployments are occurring for several reasons, including GHG-abatement and enhanced security of energy supply.
- In the business context, the international trends to more effective uranium utilization, closed fuel cycles with reprocessing and recycle of spent fuel, and more effective and efficient management of spent fuel and reduction of eventual wastes, are becoming obvious. These trends require major exporters of nuclear reactors and uranium fuel with international commitments, to develop an effective international presence and new technical processes to keep technology relevant and competitive (e.g., as evidenced by the Global Nuclear Energy Partnership efforts of the United States).
- In the strategic economic context, it is necessary to develop a longer-term view that will synergistically benefit the economy, the global environment, and the furtherance of interests at large. Such a view will provide a sound technological and scientific framework for the future, and must also address the collateral issues such as nuclear non-proliferation more effectively and realistically (e.g., the Iran, Iraq, Israel, Pakistan, India, and North Korea positions).
- Technically, this will open-up an important opportunity to transform "waste" disposal into "fuel" management. By recycling fissile and fertile material and burning actinides, the true waste that remains—fission fragments—represents a few percent or less of the fuel discharged from current reactor types. These fission fragments will decay almost entirely to stable isotopes within a few hundreds of years and their secure disposal should be a far less emotive source of objection than has been the case with unprocessed spent fuel, which retains significant levels of activity for as much as 250,000 years, albeit at levels comparable with uranium ore.

4.5.3 Two Views of Fuel Cycles: DFC or AFC

Based on the natural evolution of using enriched fuel from weapon-based ^{235}U enrichment technology, present thermal reactors use uranium as the main fuel supply with some recycling of Pu mixed-oxide fuel (MOX). The cycle is essentially a once-through system, with fuel irradiated to about 40,000 MWd/t, and then stored until cooled and ready for Pu separation, or kept in interim storage buildings (e.g., Zwileg Facility in Switzerland) until ready for sending to the underground repositories planned in many countries. As an order of magnitude, an operating 1-GW(e) LWR today requires mining of 180 t/a of uranium (House of Representatives Standing Committee 2006). (To produce uranium enriched to slightly over 3% ^{235}U, five-times as much depleted uranium at 0.24% ^{235}U is produced as depleted uranium.)

So with >372 GW in operation today (predominantly supplied from conventional uranium mines) present world demand is ~70,000 t/a. We can provide an upper bound estimate of demand for 5000 GW of new reactors needing ~ one million t/a by 2050. Today's estimates of proven uranium reserves at a cost of <$130/kg is about 6 million tonnes (IAEA and OECD-NEA 2005). Even allowing that exploration will likely lead to a doubling or tripling of the resource estimate to, say, 20 MtU, just 2000 reactors operating for 60 years would use all the world's cheapest uranium with present fuel cycles technology.

Just the present and planned 650 reactors could be kept going for another 150 years, but that falls far short of the scope for reactor deployment. This is not a cause for alarm, there is plenty of uranium, and more uranium reserves will be found but at higher prices. Moreover, aggressively adopting recycling and increased fuel utilization with existing reactor types might allow up to 1500 reactors.

So there emerges at least two views of fuel cycles, which we may summarize as shown below.

(a) *Traditional Demonstrated Fuel Cycle (DFC) View*

For those already with access to or reserves of uranium, such as the United States, France, and Canada, the uranium fuel cycle is an already demonstrated fuel cycle (DFC), and is fine while uranium is cheap and assuredly available.

There is always more uranium to find, even though the cycle is known to be unsustainable (as per the above calculations) because most current reactors (LWRs and HWRs) are very inefficient fuel users.

In DFC's unhurried view, *when* uranium becomes too scarce and/or expensive, one can switch to technologies able to breed more fissile material than they consume. This could employ fast reactors, accelerator-driven breeders, and/or Pu recycle, even if it is more expensive and requires a different reactor technology. Given its greater complexity and higher costs, transition to one or more of these advanced technologies is still decades away. In the meantime, existing thermal reactors will continue to discharge spent fuel, which can be stored retrievably and considered more as a future resource for recycling than waste.

(b) *New Alternate Fuel Cycle (AFC) View*

Without introduction of radically different reactor types, countries without access to large uranium reserves and needing energy supply surety, can initiate a turn much sooner to a new alternate fuel cycle (AFC) that will ensure sustainable supply and smaller waste streams. Preferably, this should be a more intrinsically non-proliferation cycle, with no significant Pu generation, thus not requiring all of today's policing and international stress.

As well as not requiring the introducing a new reactor technology, this perspective should acknowledge the constraints on U-enrichment ownership and deployment as a proliferation concern while still allowing vastly expanded reactor builds.

Burning thorium rather than uranium is the most obvious AFC. Thorium is about three times more plentiful globally than uranium and with careful fuel design and recycling, an HWR gives a near breeding cycle. So it is more sustainable with much lower waste amounts and storage needs (as little as 10%). Switching to thorium would enable more reactor deployment using today's reactor technologies and help stabilize fuel cost and supply. This approach avoids having to introduce many fast reactors. In the long-term, full benefit from thorium requires extracting and recycling the ^{233}U. LWRs cannot achieve a near-self-sufficient thorium cycle.

This AFC opportunity is real and potentially could totally alter the global fuel cycle and the reactor deployment opportunities and India (Kakodkar and Simha 2006) has already chosen to develop it as a national priority. Such AC concepts are in fact not new; what is new is the concept that an alternate *sustainable and closable* fuel cycle may enable greater benefits from nuclear energy deployment worldwide.

4.5.3.1 Link Between Fuel Natural Resources and Reactor Technology

There is a link between the choice of fuel cycle and the optimal reactor, qualified by noting that we would not be pursuing a "perfect" cycle, but a practical and at the same time economic one. The conventional answer has always been that we need to move to a "breeding" fast reactor, with a higher energy neutron spectrum that produces more plutonium, from neutron capture in ^{238}U, than it burns to produce power, and hence can be used to generate its own fuel. The fast-spectrum reactor is quite a complicated technology, and more expensive generally, and is one commonly adopted optimization solution. But there are others.

We must *optimize nuclear technology*: reduce the volume of waste drastically by re-use/burn of discharged fuel, and ensure sustainability and maximization of resource utilization. This means that the core design, fuel design, and safety must be harmonized with the waste streams and fuel supply.

In addition, we must *agree on definitions of sustainable and closable global fuel cycle*, recognizing the inexorable international pressure to restrict the number of countries who would be "allowed" to enrich and reprocess fuel.

This means we must examine fuel cycles that use different processing, different separation systems for isotopes, and also place different and smaller demands on enrichment facilities.

The response to these challenges involves some new thinking, new R&D, and a new fuel development strategy, using alternate cycles that include recycling *and* thorium fuels.

4.5.3.2 Global Realities and Directions: Emphasis on Nuclear Energy and Fuel Cycles

An increasing emphasis on global energy supplies is inevitable. World nuclear use will grow as energy demand, economic needs, environment issues and supply security concerns grow. As part of the effort to address concerns over potential climate change, massive switching to non-carbon sources is needed and should be anticipated.

LWRs will provide much of the anticipated expansion, but the co-existence of HWRs adds flexibility and the capacity to extend the fuel resource. In the short-term, although expectations of nuclear expansion have already led to a near ten-fold price increase for uranium, this is stimulating a wave of exploration that is already defining large additions to the known reserve. However, exploration will not address the supply issue indefinitely. Deeper into the expanded deployment of nuclear power, extracting more of the available energy in uranium and adding thorium will become important and the flexibility and superior efficiency inherent in HWR will become increasingly attractive, ultimately providing the best gateway into utilizing the world's large thorium resource.

Glossary

AFC	– Alternate fuel Cycle
BLW	– Boiling Light Water
CANDU	– CANada Deuterium Uranium
CANFLEX	– CANDU FLEXible fueling – an advanced fuel bundle design
CANLUB	– Graphite-lubricated CANDU fuel
CNSC	– Canadian Nuclear Safety Commission
DFC	– Demonstrated fuel Cycle
DUPIC	– Direct Use of Spent PWR Fuel in CANDU
ECCS	– Emergency Core Cooling System
HTS	– Heat Transport System
HWR	– Heavy Water Reactor
HWM	– Heavy Water Moderated
IAEA	– International Atomic Energy Agency
LOCA	– Loss of Coolant
LOECC	– Loss of Emergency Core Cooling
LVRF	– Low Void-Reactivity Fuel
MOX	– Mixed (uranium and plutonium) Oxide fuel
MWe	– Megawatts electrical power
MWth	– Megawatts thermal power
NPD	– Nuclear Power Demonstration reactor
NRU	– Nuclear Reactor Universal (an AECL test reactor located at Chalk River, ON)
OREOX	– Oxidation and REduction of Oxide fuels (reprocess technology for spent reactor fuel)
OTT	– Once-Through Thorium fueling cycle
pD	– hydrogen-ion concentration in heavy water, analogous to pH in ordinary water
PHWR	– Pressurized Heavy Water Reactor
PIE	– Post-Irradiation Examination
RU	– Recycled Uranium fuel
SEU	– Slightly Enriched Uranium (enriched in ^{235}U)
SGMB	– Sol-Gel Microsphere Pelletisation
SSET	– Self-Sufficient Equilibrium Thorium Cycle

References

Benedict, M., T.H. Pigford, and H.W. Levi. 1981. Nuclear Chemical Engineering, 2nd Ed., McGraw-Hill, Inc., New York.

Blahnik, C., J.C. Luxat, and S. Nijhawan. 1993. CANDU response to loss of all heat sinks. Proceedings of Ninth Nuclear Thermal Hydraulics Topical Meeting. ANS, November, San Francisco, California, USA.

Duffey, R.B. and I. Pioro. 2006. Advanced high temperature concepts for pressure-tube reactors, including co-generation and sustainability. Proceedings of the HTR2006 Conference, Johannesburg, South Africa.

House of Representatives Standing Committee on Industry and Resources. 2006. November, Canberra, Australia.

Hurst, D.G. and F.C. Boyd. 1972. Reactor licensing and safety requirements, Paper No. 72-CNA-102, 12th Annual CNA Conference of the Canadian Nuclear Association, June 11–14, in Ottawa, Canada.

IAEA Australia Uranium Government Report, Citing in Table 3.1 on p.150 and OECD/NEA, Uranium. 2005. Resources, production and demand, pp. 15–16, 102–103.

International Atomic Energy Agency (IAEA). 1996. Defence in depth in nuclear safety, Report INSAG-10.

International Atomic Energy Agency (IAEA). 2002. Heavy water reactors: Status and projected development, Technical Report Series. No. 407, Vienna.

Kukodkar, A. and R.K. Simha. 2006. The twin challenges of abundant nuclear energy supply and proliferation risk reduction: A view. 50th IAEA general conference, 18–22 September, Vienna, Austria.

Luxat, D.L. and J.C. Luxat. 2007. Evaluation of failure modes of CANDU calandria vessels under severe accident conditions. Proceedings of the 19th SMiRT Conference, August, Toronto, Ontario, Canada.

Luxat, J.C. 2003. Historical perspective of the design basis in Canada. Proceedings of the OECD/CSNI/CNRA workshop on redefining large break LOCA, June 2, Zurich, Switzerland.

Luxat, J.C. 2007. Thermalhydraulic aspects of progression to severe accidents in CANDU reactors. Keynote Lecture, Proceedings of the 12th International Topical Meeting on Nuclear Reactor Thermalhydraulics (NURETH-12), September 30–October 4, Sheraton Station Square, Pittsburgh, Pennsylvania, USA.

Miller, A.I., S. Suppiah., and R.B. Duffey. 2006. Climate change gains more from nuclear substitution than from conservation. *Nuclear Engineering and Design* 236: 1657–1667.

Muzumdar, A.P., J.T. Rogers, P.M. Mathew, and M.L. Lamari. 1998. Analysis of core melt retention capability of CANDU reactors. Proceedings of the 6th International Conference on Nuclear Engineering - ICONE-6, May 10–15. San Diego, CA.

Pioro, I.L., H.F. Khartabil, and R.B. Duffey. 2002. Literature survey devoted to the heat transfer and hydraulic resistance of fluids at supercritical and near-critical pressures, AECL-12137, FFC-FCT-409, February, Ontario, Canada.

Rogers, J.T., D.A. Meneley, C. Blahnik, V.G. Snell, and S. Nijhawan. 1995. Coolability of severely degraded CANDU cores. International seminar on heat and mass transfer in severe reactor accidents, CESME, May, Turkey.

Spinks, N.J., N. Pontikakis, and R.B. Duffey. 2002. Thermo-economic assessment of advanced, high-temperature CANDU reactors. Proceedings of the 10th International Conference on Nuclear Engineering (ICONE 10), ICONE10-22433, April, Arlington, Virginia, USA.

Vasic, A. and H.F. Khartabil. 2005. Passive cooling of the CANDU SCWR fuel at LOCA/LOECC conditions. Proceedings of GLOBAL 2005, Paper No. 184, October 9–13, Tsukuba, Japan.

5

High-Temperature Gas Cooled Reactors

Arkal Shenoy and Chris Ellis
Modular Helium Reactor Division

CONTENTS

5.1 History of High-Temperature Gas Cooled Reactors (HTGR)

The HTGR concept evolved from early air-cooled and CO_2-cooled reactors. The use of helium in lieu of air or CO_2 as the coolant, in combination with a graphite moderator, offered enhanced neutronic and thermal efficiencies. The combination helium cooling and graphite moderator makes possible production of high temperature nuclear heat, and hence the name HTGR.

To date, seven HTGR plants have been built and operated (Table 5.1). The first was the 20-MW(t) Dragon test reactor in the UK. Dragon was followed by construction of two relatively low power plants, the 115-MW(t) Peach Bottom I (PB-1) in the United States and the 49-MW(t) AVR in Germany. PB-1 and AVR demonstrated electricity generation from HTGR nuclear heat using the Rankine (steam) cycle. These two plants were followed by the construction of two mid-size steam cycle plants, the 842-MW(t) Fort St. Vrain (FSV) plant in the United States and the 750-MW(t) THTR plant in Germany. In addition to demonstrating the use of helium coolant (with outlet temperatures as high as 950°C) and graphite moderator, these early plants also demonstrated coated particle fuel, a fuel form that employs ceramic coatings for containment of fission products at high temperature, which is a key feature of HTGRs. Figure 5.1 displays pictures of most of the HTGR plants and which elements of the HTGR technology program influenced modern plants.

5.1.1 Description of PB-1

The PB-1 active core is a cylinder, 2.8-m high, containing 804 fuel elements, 36 control rods, and 19 shutdown rods (Figure 5.2) [Melese 1984]. The fuel elements, 89-mm in diameter, are vertically oriented in a closely packed triangular array with helium flowing up between the elements. The bottom and top graphite reflector sections are an integral part of the fuel element, which has a total height of 3.66 m including the fuel element end fittings. The side reflector, ~60-cm thick, consists of an inner ring of hexagonal graphite elements surrounded by a segmented graphite ring, with a 4-m outer diameter. Helium coolant at 345°C enters the reactor vessel from the outer annuli in the concentric ducts in each of the two loops. It cools the vessel walls and the reflector before flowing up through the core and leaving through the inner concentric ducts at 725°C. The steel reactor pressure vessel, 4.2 m in diameter and 11-m high, is designed for 385°C and 3.1 MPa (31 atm), the actual helium pressure being 2.4 MPa.

The fuel elements are solid and semi-homogeneous with graphite serving as the moderator, cladding, fuel matrix and structure. They consist of an upper reflector section, a fuel-bearing section, an internal fission product trap, and a bottom reflector. A low permeability graphite sleeve, ~ 3-m long, is joined to the upper reflector at one end and to a

TABLE 5.1

HTGR Plants Constructed and Operated

Feature	Dragon	Peach Bottom	AVR	Fort St. Vrain	THTR	HTTR	HTR-10
Location	UK	USA	Germany	USA	Germany	Japan	China
Power (MW(t)/Mwe)	20/ -	115/40	46/15	842/330	750/300	30/-	10/-
Fuel elements	Cylindrical	Cylindrical	Spherical	Hexagonal	Spherical	Hexagonal	Spherical
He temp (In/Out°C)	350/750	377/750	270/950	400/775	270/750	395/950	300/900
He press (Bar)	20	22.5	11	48	40	40	20
Pwr density (MW/m^3)	14	8.3	2.3	6.3	6	2.5	2
Fuel coating	TRISO[a]	BISO[b]	BISO[b]	TRISO[a]	BISO[b]	TRISO[a]	TRISO[a]
Fuel kernel	Carbide	Carbide	Oxide	Carbide	Oxide	Oxide	Oxide
Fuel enrichment	LEU[c]/ HEU[d]	HEU[d]	HEU[d]	HEU[d]	HEU[d]	LEU[c]	LEU[c]
Reactor vessel	Steel	Steel	Steel	PCRV[e]	PCRV[e]	Steel	Steel
Operation years	1965–1975	1967–1974	1968–1988	1979–1989	1985–1989	1998–	1998–

[a] TRISO refers to a fuel coating system that uses three types of coatings, low density pyrolytic carbon, high density pyrolytic carbon and silicon carbide.

[b] BISO refers to a fuel coating system that uses two types of coatings, low density pyrolytic carbon and high density pyrolytic carbon.

[c] LEU means low enriched uranium (<20% U^{235}).

[d] HEU means high enriched uranium (>20% U^{235}).

[e] PCRV means Prestressed Concrete Reactor Vessel.

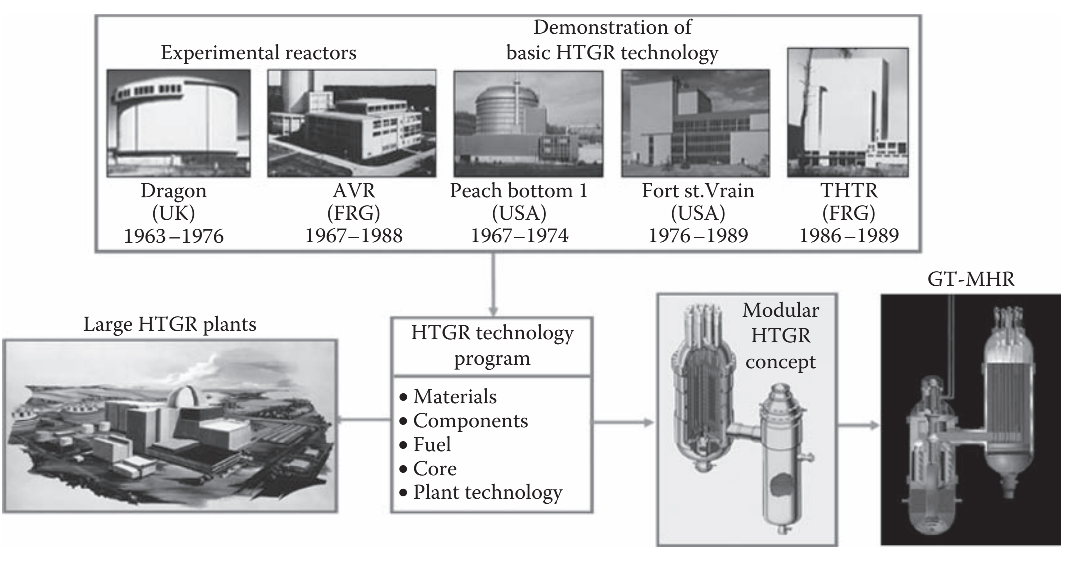


FIGURE 5.1
Broad global foundation of helium reactor technology.

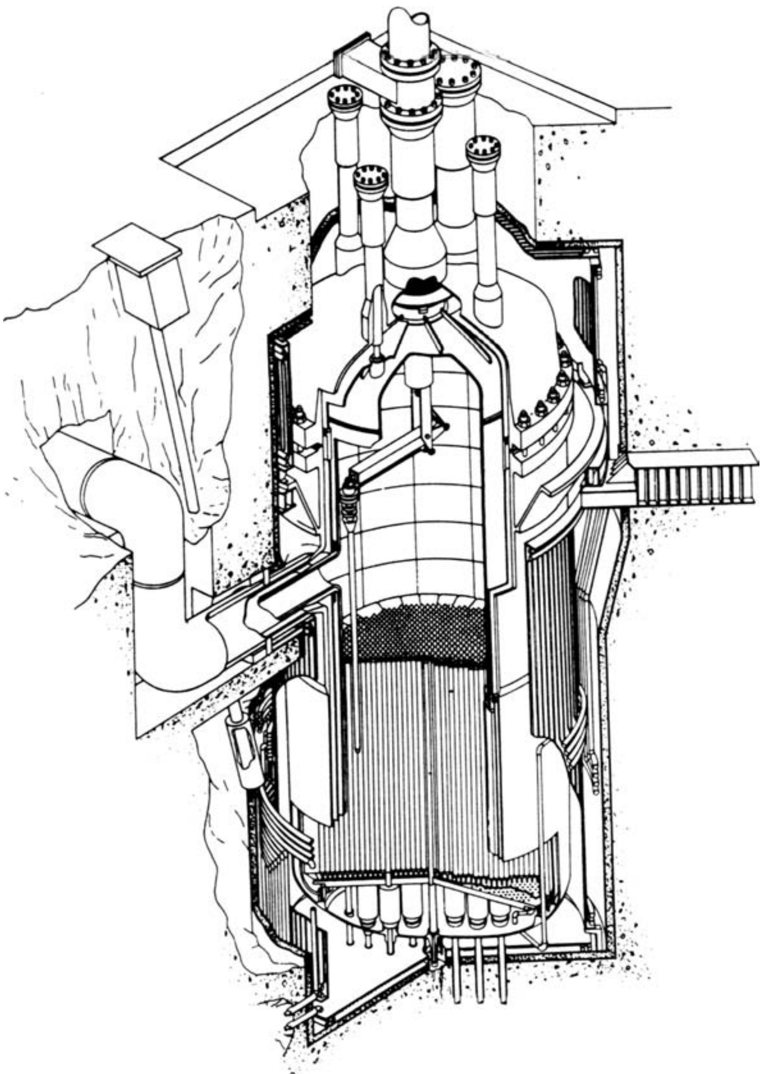


FIGURE 5.2
Isometric view of the peach bottom reactor.

bottom connector fitting at the other end. It contains the fuel bodies consisting of annular fuel compacts, 44 × 69 mm and 76-mm long, stacked on a graphite spine. Uranium and thorium carbide particles, coated with PyC, are uniformly dispersed in the fuel compacts. The total fuel loading is 236 kg of 93.5% enriched uranium and 1450 kg of thorium. A small fraction of the helium coolant enters the fuel element through a porous plug in the upper reflector piece and flows downward between the fuel compacts and the graphite sleeve. After sweeping fission product gases from the active core zone, the purge gas flows through the internal traps and then through a purge line to external traps. The fuel elements are designed to stay in the reactor for three years and are batch-loaded with the reactor shut down, consistent with U.S. light water reactor conditions. Helium leaving the reactor flows through the two steam generators (one per loop) before being returned to the reactor by horizontal single-stage 3200-rpm electrically driven centrifugal compressors (1.85-MW each). Vertical shell-and-tube forced recirculation steam generators are used, each section of which is constructed of a bank of U-tubes. Each steam generator shell, ~2.4 m in diameter by 9-m high, is cooled by cold helium leaving the economizer. The secondary reactor containment is a vertical, cylindrical-shaped steel shell, 30.5 m in diameter and 49.5-m high, designed for an internal over-pressure of 0.055 MPa (0.55 atm) at 65°C.

5.1.2 Fort St. Vrain Description

The Fort St. Vrain reactor was designed to produce 842 MW(t) and 330 MW(e), and had many design features similar to the Gas-Turbine Modular Helium Reactor (GT-MHR) discussed later in this chapter, e.g., graphite moderation, helium coolant, and very similar designs for fuel particles, fuel elements, and control rods [Baxter 1994]. The fuel compacts, which were inserted into machined blind holes in the fuel element, were composed of TRISO-coated fuel particles in a carbonaceous matrix. The TRISO coatings on the fuel particles had been shown to be a highly impervious barrier to radionuclide release in irradiation tests [IAEA 1997]. Coolant holes, slightly larger in diameter than the fuel holes, were drilled in parallel through the block to allow the helium to be circulated through the fuel element coolant holes and remove the heat generated in the fuel. The Fort St. Vrain reactor core is composed of 247 columns of fuel elements, with six fuel elements stacked in each column. Axial reflector blocks are also located above and below the core. The core columns were grouped into 37 refueling regions with the flow in each region controlled by an adjustable inlet flow control valve at the top of the core to maintain a fixed core outlet temperature as power changed due to fuel burnup. The 37 refueling regions contain five or seven columns. About one-sixth of the 37 regions were refueled each reactor year. The elements in the central column of each of the 37 refueling regions contained two holes for insertion of control rod pairs, and one hole for insertion of reserve shutdown pellets. The control rods consisted of pairs of metal-clad boronated graphite control rods and were operated by electric drives and cable drums. The reserve shutdown pellets were boronated graphite cylinders with spherical ends which could be dropped into the core to provide an independent and diverse reactor shutdown system.

Figure 5.3 is a cut-away view of the FSV reactor core in a prestressed concrete reactor vessel (PCRV), with control rods inserted into the top of the core. The PCRV acted as a pressure vessel, containment, and biological shield. The bottom head had 12 penetrations for the steam generator modules, four penetrations for the helium circulators, and a large central opening for access. A 3/4 inch-thick carbon steel liner anchored to the concrete provided a helium-tight membrane. Two independent systems of water-cooled tubes welded to the concrete side of the liner and kaowool fibrous insulation of the reactor side of the liner limited the temperatures in both the liner and the PCRV.

The primary coolant circuit was wholly contained within the PCRV with the core and reflectors located in the upper part of the cavity, and the steam generators and circulators located in the lower part. The helium coolant flowed downward through the reactor core and was then directed into the reheater, superheater, evaporator, and economizer sections of the 12 steam generators. From the steam generators, the helium entered the four circulators and was pumped up, around the outside of the core support floor and the core barrel before entering the plenum above the core. The superheated and reheated steam was converted to electricity in a conventional steam cycle power conversion turbine-generator system.

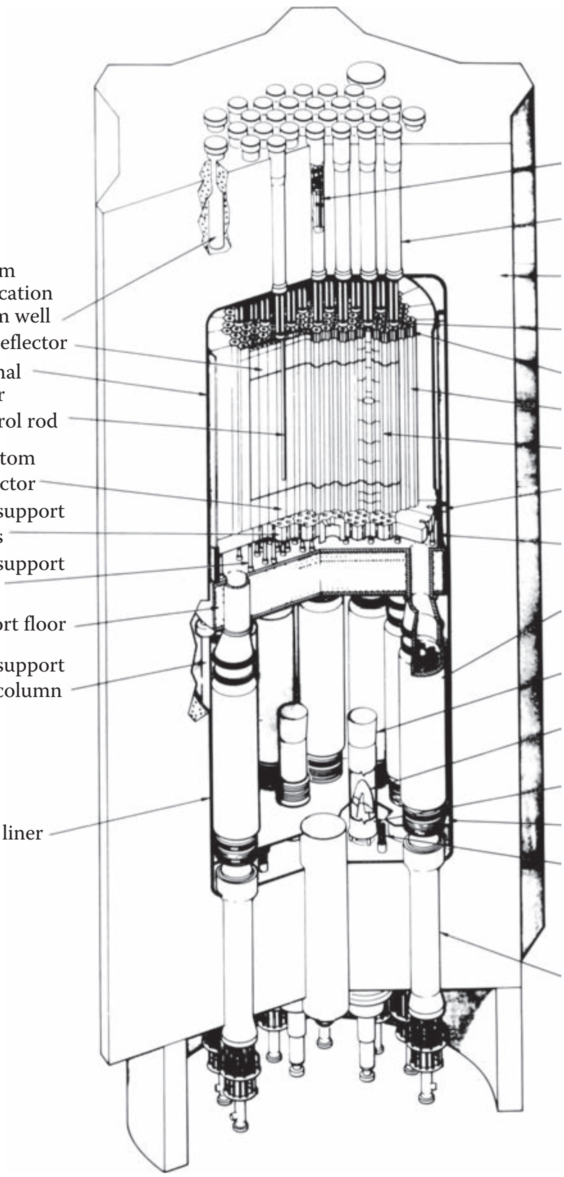


FIGURE 5.3
Isometric view of FSV PCRV, core and primary system.

5.1.3 Steam Cycle/Variable Cogeneration (SC/C) HTGR Plant Description (Conceptual Design, Circa 1985)

Based on experience obtained from Peach Bottom, Fort St. Vrain, and international programs, the effort in the United States in 1983 was concentrated on the design of a 2240-MW(t) four-loop HTGR SC/C system [Melese 1984]. It would have a maximum electrical output of 820 MW, in the all-electric mode, or a minimum of 231 MW(e) while providing 631 kg/s of high-quality steam (5.9 MPa/538°C). These conditions, obtained with a maximum helium temperature of 690°C, would lead to 38% net efficiency in the all-electric version. Such use of a topping steam cycle for electricity production and of reduced pressure steam for process heat applications has the dual advantage of resource conservation and of savings in electricity and/or steam costs. There is a large market potential for steam in the United States; the main problems appear to be institutional. Several modifications have been included in the design of the 2240-MW(t) HTGR compared with the Fort St. Vrain system: a multi-cavity PCRV rather than a single cavity; a non-reheat steam cycle instead of nuclear reheat; electric motor-driven circulators compared to steam drives; a core auxiliary heat removal system; a reactor secondary containment building; a reduced outlet helium temperature (690°C versus 775°C); and a flexible fuel cycle, i.e., 20% to 93% uranium enrichment. Those changes were expected to improve plant performance and reliability, to simplify plant operation and maintenance, and to satisfy projected licensing and regulatory requirements. Participants in the U.S. Department of Energy-funded program included utilities (Gas-Cooled Reactor Associates), a national laboratory (Oak Ridge), and industry (GA Technologies and General Electric) with, as main subcontractors, Bechtel Power, Combustion Engineering, and United Engineers and Constructors. Research and development was also performed on advanced HTGR systems, such as process heat or steam reforming applications, or direct-cycle design for cogeneration applications.

At the time, renewed interest arose in the United States as well as in other countries, in small or modular HTGR systems. To obtain acceptable economics compared to fossil-fired plants, the goals of small HTGR power plant designs were: simplification of the overall system: reduction of construction time; standardized design: and maximum inherent safety with passive systems. Modular systems could be built on a phased basis, thus relieving the initial investment risk. Improved reliability with multiple units should facilitate applications to process heat. These modular, and more modern, prismatic HTGR designs shall be discussed in Section 5.3.

5.2 HTGR Type Comparison and Contrast

5.2.1 HTGR Type Similarities

All HTGR designs utilize a refractory coated particle fuel. These particles are identified as BISO- or TRISO-coated particle fuel, which consists of a spherical kernel of fissile and/or fertile fuel material (as appropriate for the application), encapsulated in multiple layers of refractory coatings. The multiple coating layers form a miniature, highly corrosion-resistant pressure vessel and an essentially impermeable barrier to the release of gaseous and metallic fission products. The BISO type is no longer in use within currently operating HTGRs, as seen in Table 5.1, mainly due to its inferior performance and fission product retention compared with the TRISO type [Hanson 2004]. The BISO is therefore just a historical note, and for the remainder of this chapter all fuel particles shall refer to the TRISO type, as shown in Figure 5.4. The overall diameter of standard TRISO-coated particles can vary between 650 microns to 850 microns, depending upon the burnup goal and type of fuel utilized (fissile versus fertile).

The fuel kernel may be of oxide, carbide, or oxycarbide in composition. For high burnup applications, an oxycarbide kernel is preferred to enhance performance [Hanson 2004]. The carbide component of the kernel undergoes oxidation to getter excess oxygen released during fission. If the carbide component were not present, excess oxygen would react with carbon in the buffer to form carbon monoxide. High levels of carbon monoxide can lead to failure of the coating system by overpressurization and kernel migration.

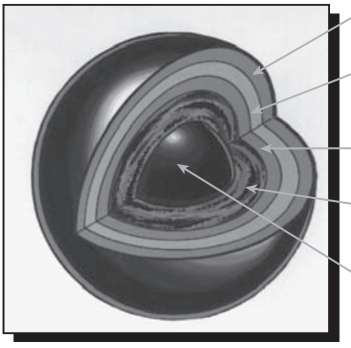


FIGURE 5.4
TRISO coated fuel particle.

The buffer layer is deposited over the kernel and consists of low-density, porous pyrocarbon.

The buffer attenuates fission fragments that recoil from the kernel and provides sufficient void space to accommodate gases, including gaseous fission products and carbon monoxide. The buffer also acts as a sacrificial layer to accommodate potential kernel migration and swelling and isolates the kernel from load-bearing layers of the coating system.

The high-density inner pyrolytic carbon (IPyC) layer protects the kernel and buffer from chemical attack by chlorine compounds, which are generated as byproducts during deposition of the silicon carbide (SiC) layer. The IPyC layer also provides a surface for deposition of the SiC layer and delays transport of radionuclides to the SiC layer. The IPyC layer shrinks with the accumulation of fast neutron fluence, which helps to maintain the SiC layer in compression, provided the bond between the IPyC and SiC layers remains strong and continuous during irradiation.

The SiC layer is deposited under conditions that produce a high-density, high-strength coating with a fine-grain microstructure. This layer provides the primary structural support to accommodate stresses generated by internal gas pressure and irradiation-induced dimensional changes of the pyrocarbon layers. The SiC layer provides an impermeable barrier to gaseous, volatile, and most metallic fission products during normal operation and hypothetical accidents. Dimensional changes of the SiC are very small during irradiation, and it is considered to be dimensionally stable.

The high-density outer pyrolytic carbon (OPyC) layer protects the SiC layer from mechanical damage that may occur during fabrication of fuel compacts and fuel elements, and provides a bonding surface for the compact matrix. The OPyC layer also shrinks during irradiation, which helps to maintain the SiC layer in compression. The OPyC layer prevents the release of gaseous fission products if both the IPyC and SiC layers are defective or fail in service.

The TRISO coatings provide a high-temperature, high-integrity structure for retention of fission products to very high burnups. The coatings do not start to thermally degrade until temperatures approaching 2000°C are reached (Figure 5.5). For example, typically for a reactor outlet coolant temperature of 850°C, normal operating fuel kernel temperatures do not exceed about 1250°C and worst-case accident temperatures are maintained below 1600°C. Extensive tests in the United States, Europe, and Japan have demonstrated the performance potential of this fuel, but tests still need to be done to demonstrate it satisfies Generation IV performance requirements or normal operating and accident conditions [Hanson 2004].

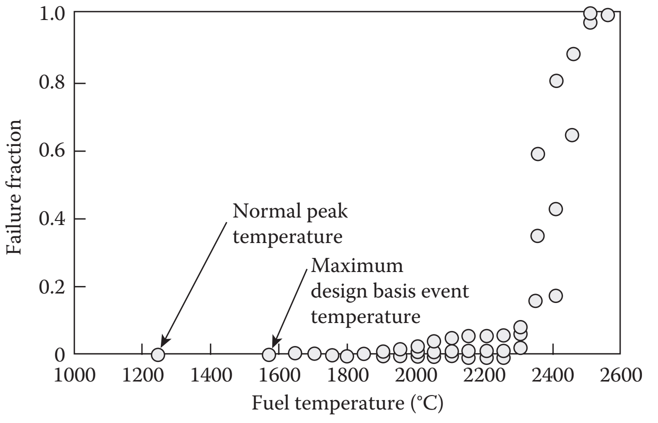


FIGURE 5.5
TRISO coated particle fuel temperature capability.

5.2.2 HTGR Type Differences

The most important HTGR design distinction is spherical fuel elements versus hexagonal/cylindrical fuel elements. As shown in Table 5.1, the AVR and THTR HTGRs in Germany utilize spherical fuel elements, known as a "pebble bed reactor." The remaining HTGRs in Table 5.1 utilize hexagonal/cylindrical fuel elements, known as a prismatic block reactor. As previously mentioned, both of these HTGR design concepts use TRISO-coated fuel particles, but the fuel particles are contained in fuel elements having quite different configurations, as described below.

In a prismatic block reactor, the TRISO-coated fuel particles are mixed with a carbonaceous matrix and bonded into cylindrical fuel compacts normally 12.5 mm outer diameter × 50-mm long and loaded into fuel holes in hexagonal-shaped graphite fuel blocks that are about 80-cm in height and 36-cm across flats (Figure 5.6). The fuel is cooled by helium that flows downward through vertical coolant channels in the graphite blocks. Spent fuel blocks are removed and replaced with fresh fuel blocks during periodic refueling outages.

In a pebble bed reactor, the fuel particles are contained in billiard-ball sized spherical fuel elements (i.e., pebbles), as shown in Figure 5.7 [PBMR 2005]. The fuel is cooled by helium flowing downward through a close-packed bed of the spherical fuel elements. These pebbles are removed continuously from the core during reactor operation, measured for radionuclide content, and returned to the core or replaced with a fresh fuel element depending on the amount of fuel depletion. With this continuous on-line refueling approach, there is no need for refueling outages.

5.3 HTGR Design Evolution

Past HTGR designs were first challenged in 1984, when the U.S. Congress asked the HTGR industry to investigate the potential for using their technology to develop a "simpler, safer" nuclear power plant design. This goal of developing a passively safe HTGR plant that was also economically competitive has since stayed with the HTGR industry. In addition

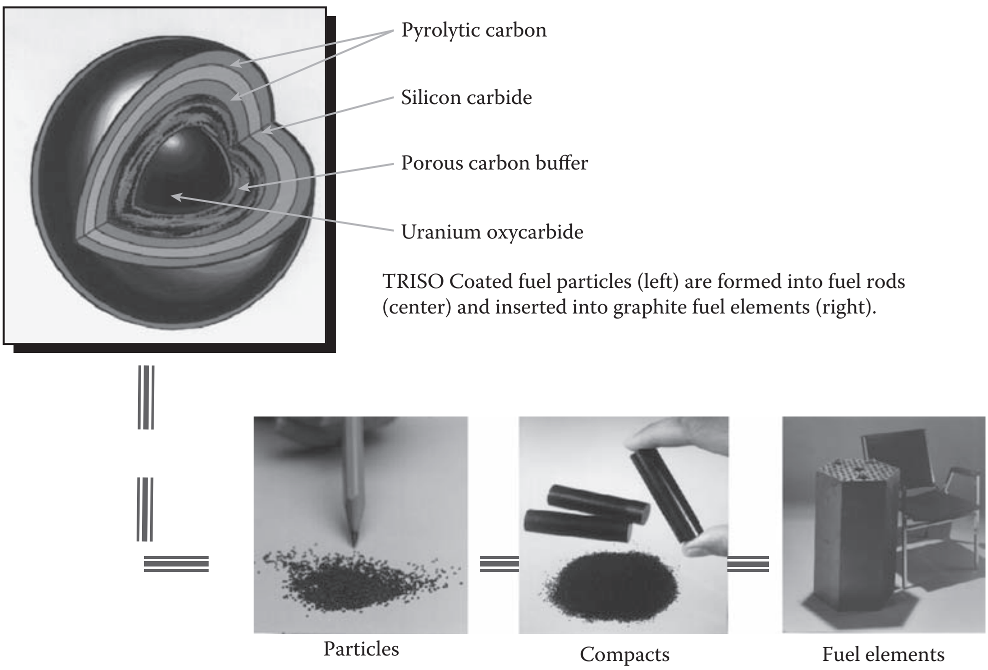


FIGURE 5.6
TRISO coated particle fuel arrangement in hexagonal fuel elements.

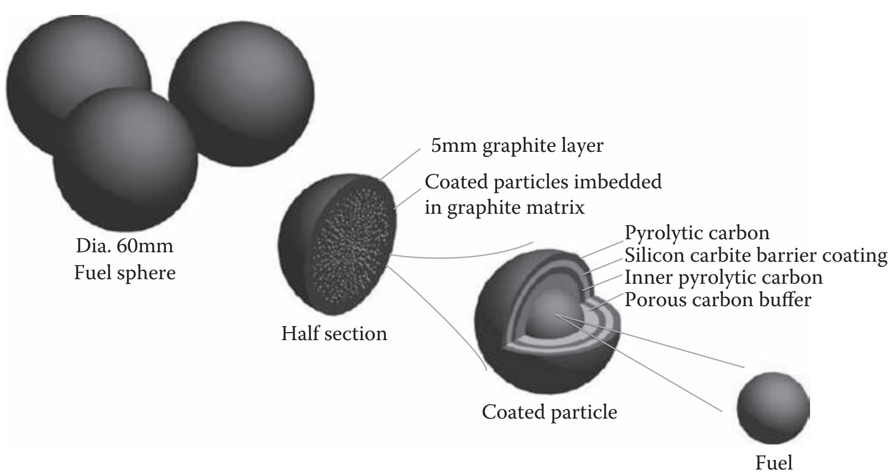


FIGURE 5.7
TRISO coated particle fuel arrangement in spherical fuel elements.

to, and more recently, the Generation IV Forum further challenged HTGR designs to enhance proliferation-resistance and reduce spent fuel inventory. HTGR designers of both types (pebble bed and prismatic block) have responded with a modular reactor approach described in more detail below. As a result, modular HTGR designs started to be considered in the United States in the late 1980s with a thermal rating of 250 MW(t) and a net plant efficiency of ~37% with 4-MPa helium at a top temperature of 690°C. As seen in Table 5.1, because there is limited worldwide experience in HTGR operation, its design evolution and follow-up to construction is even more limited. In fact, the 10-MW(t) China HTR-10 (pebble bed) and 30-MW(t) Japan HTTR (prismatic block) reactors are the only currently operating HTGRs in the world. Although these are only test reactors, having relatively small core thermal power which is completely discharged, they provide critical performance data to help validate a key HTGR evolutionary design change: TRISO fuel particle performance under high burnup and high coolant outlet temperatures. Several modular (prismatic only) plant design descriptions follow in chronological order.

5.3.1 MHTGR Steam Cycle Plant Description (Conceptual Design, Circa 1990)

The reference Modular High Temperature Gas Reactor (MHTGR) steam cycle plant consisted of four identical 350-MW(t) reactor modules with a net electrical output of approximately 550 MW(e) (see Figure 5.8) [Williams 1994]. Each module is housed in a vertical cylindrical concrete silo embedded underground. Each silo serves as an independent

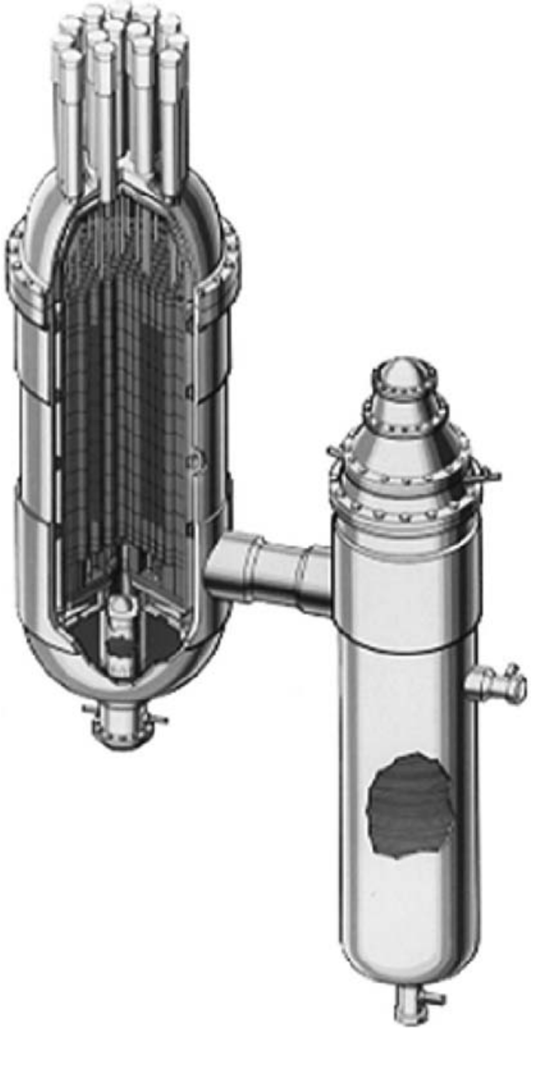


FIGURE 5.8
MHTGR steam cycle plant vessels.

vented containment structure. The four reactor structures form part of the nuclear island (NI) along with other structures which house systems for helium purification, shutdown cooling, hot cell maintenance, power conditioning, and heating, ventilating, and air conditioning. A storage array cooled by natural circulating air is provided to accommodate on-site storage of spent fuel in an adjacent reactor service area. The four reactor structures and the reactor service area are covered by a common enclosure which allows sharing of auxiliary cranes and fuel handling equipment.

The MHTGR energy conversion area, or turbine island, is non-safety-related and is separated from the NI so that conventional, fossil-fired equipment and standards can be used in its construction and operation. It is located adjacent to the NI so the main steam and feedwater connections between the turbine building and the individual reactor structures will be as short and direct as possible. The reference energy conversion design incorporates two 275-MW(e) non-reheat turbine generator sets, each connected to a pair of reactors. Four stages of feedwater heating are used to optimize the turbine cycle.

The core incorporated a graded low-enriched uranium and thorium (LEU/Th) fuel cycle with an equilibrium cycle in which fuel exposure reaches 964 effective full power days (EFPDs) (3.3 calendar years at 80% equivalent availability), with one-half of the active core being replaced every 482 EFPDs (1.65 calendar years at 80% equivalent availability).

5.3.2 Process Steam/Cogeneration Modular Helium Reactor (PS/C-MHR) Plant Description (Conceptual Design, Circa 1995)

The PS/SC-MHR was designed to meet the rigorous requirements established by the Nuclear Regulatory Commission (NRC) and the electric utility-user industry for a second-generation power source in the late 1990s [Shenoy 1995]. The plant was expected to be equally attractive for deployment and operation in the United States, other major industrialized nations, and the "developing" nations of the world.

At the time, the most economic PS/C-MHR plant configuration included an arrangement of several identical modular reactor units, each located in a single reactor building. The plant was divided into two major areas: the NI, containing several reactor modules, an energy conversion area (ECA), containing turbine generators and other balance of plant equipment. Each reactor module was designed to be connected independently to steam turbine in or other steam utilizing systems.

The reactor module components are contained within three steel pressure vessels; the reactor vessel, a steam generator vessel, and connecting cross vessel. The uninsulated steel reactor pressure vessel is approximately the same size as that of a large boiling-water reactor (BWR) and contains the core, reflector, and associated supports. The annular reactor core and the surrounding graphite reflectors are supported on a steel core support plate at the lower end of the reactor vessel. Top-mounted penetrations house the control-rod drive mechanisms and the hoppers containing boron carbide pellets for reserve shutdown.

The heat transport system (HTS) provides heat transfer during normal operation or under normal shutdown operation using high pressure, compressor-driven helium that is heated as it flows down through the core. The coolant flows through the coaxial hot duct inside the cross vessel and downward over the once-through helical bundle steam generator. Helium then flows upward, in an annulus, between the steam generator vessel and a shroud leading to the main circulator inlet. The main circulator is a helium-submerged, electric-motor-driven, two-stage axial compressor with active magnetic bearings. The circulator discharges helium through the annulus of the cross vessel and hot duct and then upward past the reactor vessel walls to the top plenum over the core.

Major cogeneration applications are highly energy intensive and diverse, including such processes as those associated with heavy oil recovery, tar sands oil recovery, coal liquification, h-coal liquefaction, coal gasification, steel mill and aluminum mill processes. Several process heat applications were also considered in the design too, which are discussed in more detail later.

5.3.3 GT-MHR Plant Description (Preliminary Design, Circa 2000)

Like most nuclear power plants up to that time, HTGR plants had been designed with reactor core length-to-diameter (L/D) ratios of about 1 for neutron economy. Detailed evaluations showed that low power density HTGR cores with L/Ds of 2 or 3, or more, were effective for rejecting decay heat passively. In the long slender, low power density HTGR cores, it was found that decay heat could be transferred passively by natural means (conduction, convection and thermal radiation) to a steel reactor vessel wall and then thermally radiated (passively) from the vessel wall to surrounding reactor cavity walls for conduction to a naturally circulating cooling system or to ground itself [Labar 2003].

To maintain the coated particle fuel temperatures below damage limits during passive decay heat removal, the core physical size had to be limited, and the maximum reactor power capacity was found to be about 200 MW(t) for a solid cylindrical core geometry. However, a 200-MW(t) power plant was not projected to be economically competitive. This led to the development of an annular core concept to enable larger cores and therefore, higher reactor powers. The first MHTGR designed with an annular core had a power of 350 MW(t). When coupled with a steam cycle power conversion system (PCS), the plant had a net thermal efficiency of 38% and was economically competitive (marginally) at that time (late 1980s). To improve economics while maintaining passive safety, the core power was subsequently raised to 450 MW(t) and then to the current reference core power of 600 MW(t). The resultant modular HTGR design, now known as the Modular Helium Reactor (MHR), represents a fundamental change in reactor design and safety philosophy, and shown in Figure 5.9.

The latest evolution made for the purpose of economics has been replacement of the Rankine steam cycle PCS with a high-efficiency Brayton (gas turbine) cycle PCS to boost

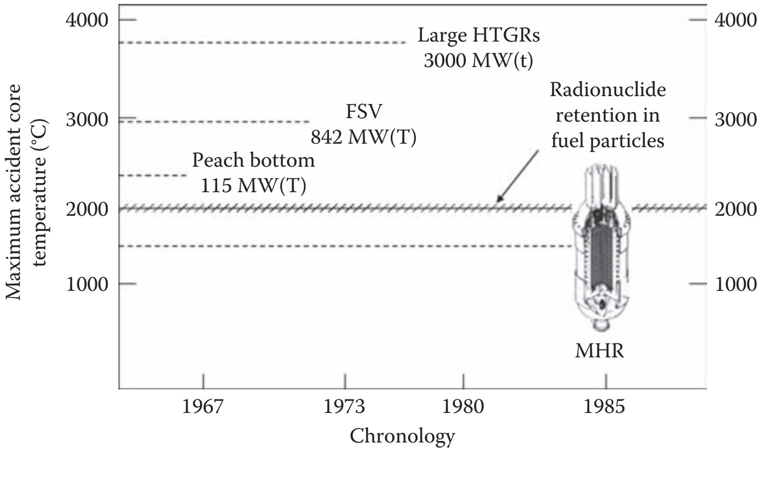


FIGURE 5.9
HTGR prismatic block evolution into the MHR.

the thermal conversion efficiency to ~48%. The coupling of the MHR with the gas turbine cycle forms the GT-MHR. The GT-MHR retains all of the MHR passive safety characteristics but is projected to have more attractive economics than any other generation alternative [Shenoy 1996]. The organization behind the MHR and GT-MHR designs is General Atomics (GA).

The GT-MHR, seen in Figure 5.10, couples a gas-cooled MHR, contained in one pressure vessel, with a high efficiency Brayton cycle gas turbine PCS contained in an adjacent pressure vessel. The reactor and power conversion vessels are interconnected with a short cross-vessel and are located in a below-grade concrete silo. The below-grade silo arrangement provides high resistance to sabotage—a requirement in a post 9/11 world. The GT-MHR share the same Gen-IV goals relating to safety, economics, environmental impact and proliferation resistance [USDOE 2002], summarized as follows:

- Safety: The safety design objective is to provide the capability to reject core decay heat relying only on passive (natural) means of heat transfer (conduction, convection, and radiation) without the use of any active safety systems.
- Economics: The economics design objective is a busbar generation cost (20 year levelized) less than the least cost generation alternative.

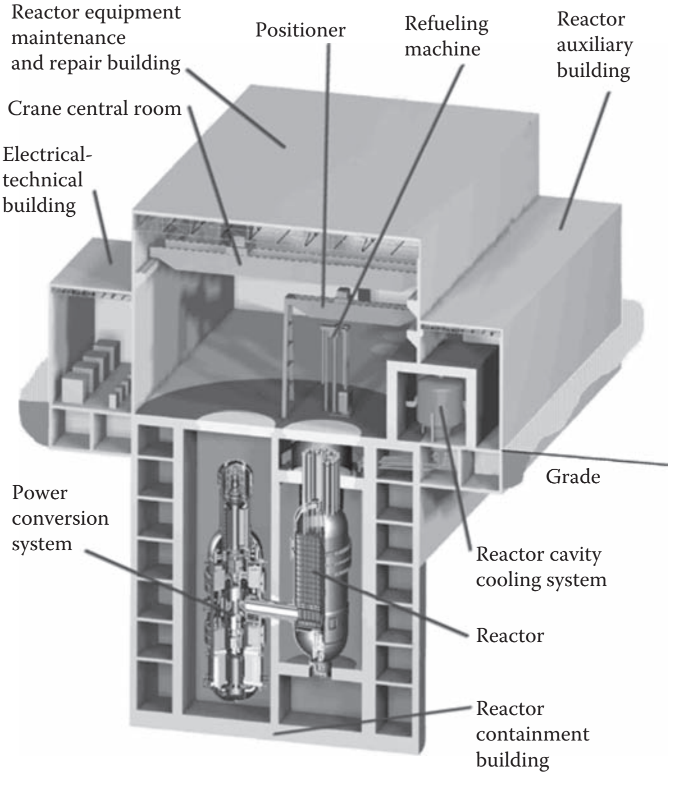


FIGURE 5.10
GT-MHR module.

- Environmental Impacts: The environmental impact design objectives, relative to the impacts of LWRs, are:
  - Reduced thermal discharge
  - Reduced heavy metal wastes
  - Reduced risk of repository spent fuel radionuclide migration to the biosphere.
- Proliferation Resistance: The proliferation resistance design objective is a plant and fuel system that has high resistance to sabotage and to diversion of either weapons usable special nuclear materials or radioactive materials.

The safety design objective is achieved through a combination of inherent safety characteristics and design selections that take maximum advantage of the inherent characteristics. The inherent characteristics and design selections include:

- Helium coolant, which is single phase, inert, has only minute reactivity effects and does not become radioactive.
- Graphite core, which provides high heat capacity, slow thermal response, and structural stability to very high temperatures.
- Refractory coated particle fuel, which retains fission products at temperatures much higher than normal operation and postulated accident conditions.
- Negative temperature coefficient of reactivity, which inherently shuts down the core above normal operating temperatures.
- A low power density core.

5.4 GT-MHR Design

5.4.1 GT-MHR Reactor System

Figure 5.11 shows a cross sectional view of the GT-MHR Reactor System, which includes the reactor core, the Neutron Control System, and other equipment within the reactor vessel. The core design consists of an array of hexagonal fuel elements surrounded by identically sized solid graphite reflector elements vertically supported at the bottom by a core support grid plate structure and laterally supported by a core barrel. The fuel elements are stacked 10 high in an annular arrangement of 102 columns (Figure 5.12) to form the active core. The core is enclosed in a steel reactor pressure vessel. Control rod mechanisms are located in the reactor vessel top head, and a shutdown cooling system (SCS) provided for maintenance purposes only is contained in the bottom head.

The mixed mean helium outlet temperature is 850°C. The hot outlet helium flows from the reactor core to the PCS through a hot duct located in the center of the cross-vessel; helium is cooled to 490°C in the PCS and returns to the reactor through the annulus formed between the cross-vessel outer shell and the central hot duct. The cooled helium flows up to an inlet plenum at the top of the core through the annulus between the reactor vessel and the core barrel. From the top inlet plenum, the helium is heated by flowing downward through coolant channels in the fuel elements, collected in a bottom outlet plenum and guided into the cross-vessel hot duct. All the core components exposed to the heated helium are either graphite or thermally insulated from exposure to the high temperature

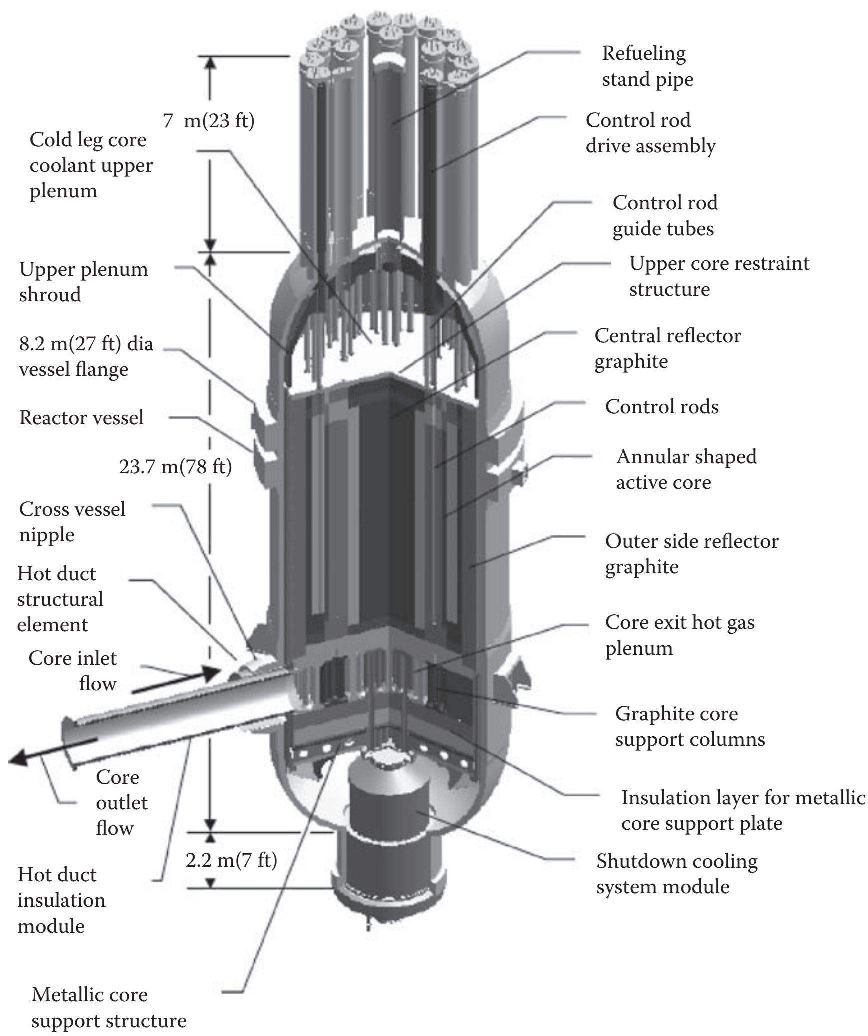


FIGURE 5.11
GT-MHR reactor system.

helium. Graphite has high strength, does not readily combust and has dimensional stability to very high temperatures (~2300°C).

Because of the accident at Chernobyl in 1986, the role of graphite in reactor safety has received increased attention. However, the consequences of the Chernobyl accident were caused by massive fuel failure and not by graphite oxidation that occurred during the accident. Decay heat from the nuclear fuel was sufficient to maintain relatively high graphite temperatures for an extended period of time, causing the graphite to radiate the "red glow" that was observed during the accident. High-purity, nuclear-grade graphite reacts very slowly with oxygen and would be classified as noncombustible by conventional standards. In fact, graphite powder is a class D fire extinguishing material for combustible metals, including zirconium. For the GT-MHR (and the PBMR for that matter), the oxidation resistance and heat capacity of graphite serves to mitigate, not exacerbate

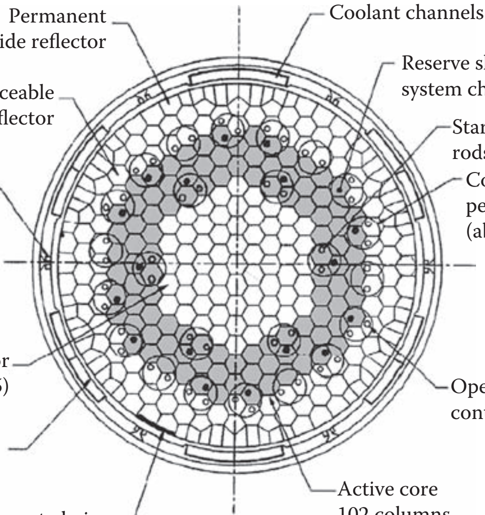


FIGURE 5.12
GT-MHR annular core cross section at vessel mid-plane.

the radiological consequences of a hypothetical severe accident that allows air into the reactor vessel.

Tables 5.2 and 5.3 show an outline of GT-MHR nominal plant design parameters and GT-MHR coated particle fuel design parameters, respectively.

5.4.2 GT-MHR PCS

The GT-MHR direct Brayton cycle (gas turbine) PCS contains a gas turbine, an electric generator, and gas compressors located on a common, ~29 m long vertically orientated shaft supported by magnetic bearings. The PCS also includes recuperator, precooler and intercooler heat exchangers. Heated helium flows (Figure 5.13) directly from the MHR into a gas turbine to drive the generator and gas compressors. From the turbine exhaust, the helium flows through the hot side of the recuperator, through the precooler and then passes through low and high-pressure compressors with intercooling. From the high-pressure compressor outlet, the helium flows through the cold, high-pressure side of the recuperator where it is heated for return to the reactor.

The use of the direct Brayton cycle to produce electricity results in a net plant efficiency of approximately 48% is shown in Figure 5.14. This efficiency is ~50% higher than that in current LWR nuclear power plants.

The GT-MHR gas turbine PCS has been made possible by key technology developments during the last several years in large aircraft and industrial gas turbines; large active magnetic bearings; compact, highly effective gas-to-gas heat exchangers; and high strength, high temperature steel alloy vessels. The selection of (1) the direct cycle PCS and (2) integrated vertical shaft PCS arrangement was made on the basis of achieving optimum economics from consideration of several alternatives. There are several alternative high efficiency Brayton cycle PCS and arrangements that could be used. Some of these would require less development effort but would have higher capital cost and electricity generation cost.

TABLE 5.2
GT-MHR Nominal Plant Design Parameters

MHR System	
Power rating, MW(t)	600
Core inlet/outlet temperatures, °C	491/850
Peak fuel temperature – normal operation, °C	1250
Peak fuel temperature – accident conditions, °C	<1600
Helium mass flow rate, kg/s	320
Core inlet/outlet pressures, MPa	7.07/7.02
Power Conversion System	
Helium mass flow rate, kg/s	320
Turbine inlet/outlet temperatures, °C	848/511
Turbine inlet/outlet pressures, MPa	7.01/2.64
Recuperator hot side inlet/outlet, °C	511/125
Recuperator cold side inlet/outlet, °C	105/491
Net plant efficiency, %	48
Net electrical output, 1 module, MW(e)	286

TABLE 5.3
GT-MHR Coated Particle Design Parameters

	Fissile Particle	**Fertile Particle**
Composition	$UC_{0.5}O_{1.5}$	$UC_{0.5}O_{1.5}$
Uranium enrichment, %	19.8	0.7 (Natural Uranium)
Dimensions (μm)		
Kernel diameter	350	500
Buffer thickness	100	65
IPyC thickness	35	35
SiC thickness	35	35
OPyC thickness	40	40
Particle diameter	770	850
Material Densities (g/cm³)		
Kernel	10.5	10.5
Buffer	1.0	1.0
IPyC	1.87	1.87
SiC	3.2	3.2
OPyC	1.83	1.83
Elemental Content Per Particle (μg)		
Carbon	305.7	379.9
Oxygen	25.7	61.6
Silicon	104.5	133.2
Uranium	254.1	610.2
Total particle mass (μg)	690.0	1184.9
Design burnup (% FIMA)[a]	26	7

[a] FIMA is an acronym for Fissions per Initial Metal Atom.

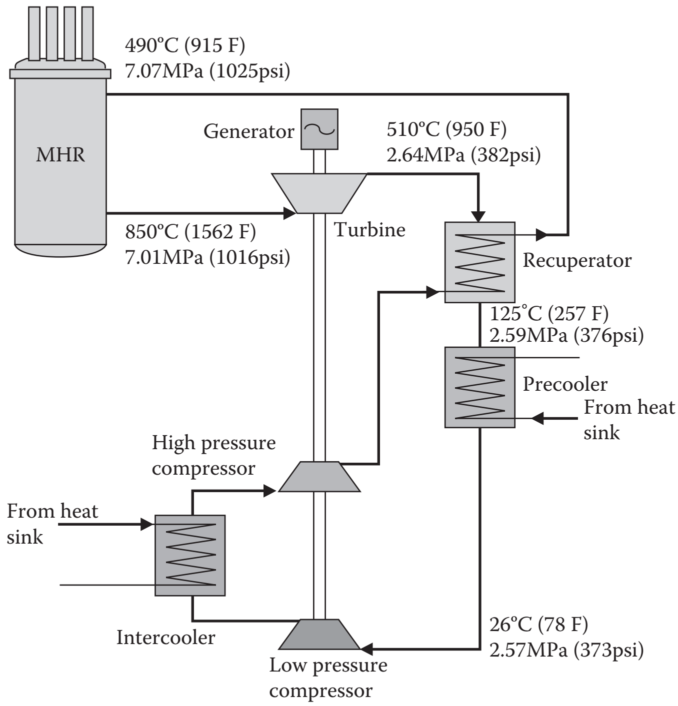


FIGURE 5.13
GT-MHR coolant flow schematic.

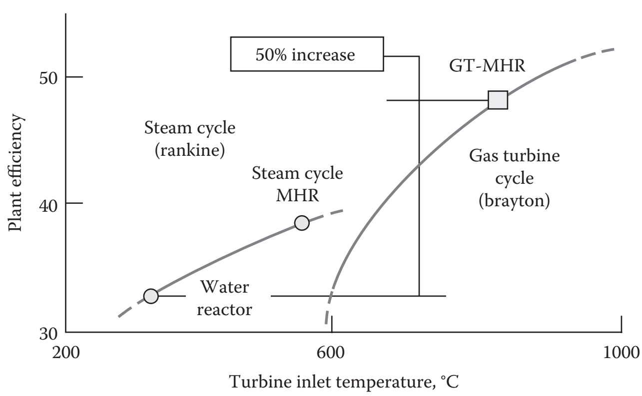


FIGURE 5.14
Comparison of thermal efficiencies.

5.4.3 GT-MHR Heat Removal System

The GT-MHR has two active, diverse active heat removal systems, the PCS and the SCS that can be used for the removal of decay heat. In the event that neither of these active systems is available, an independent passive means is provided for the removal of core decay heat. This is the reactor cavity cooling system (RCCS) that surrounds the reactor vessel (Figure 5.15). For passive removal of decay heat, the core power density and the annular core configuration have been designed such that the decay heat can be removed by conduction to

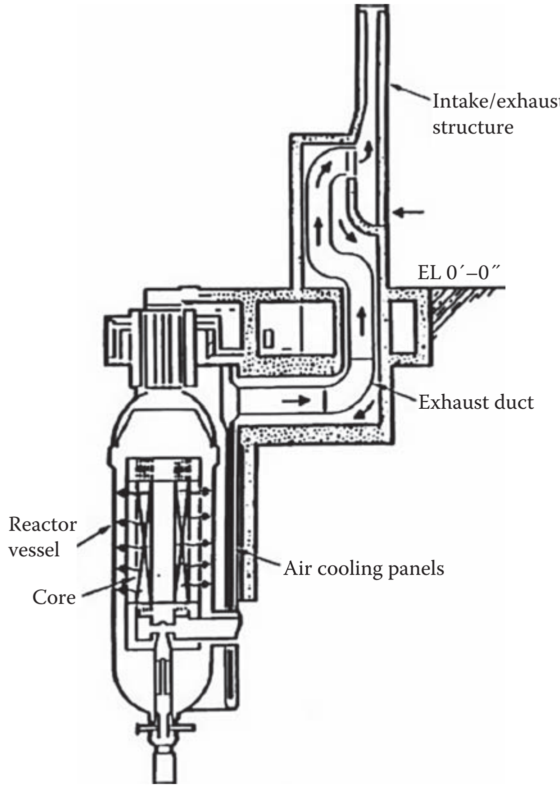


FIGURE 5.15
GT-MHR passive reactor cavity cooling system.

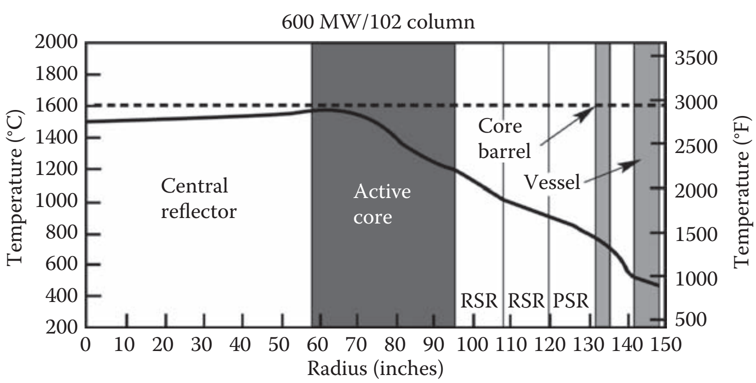


FIGURE 5.16
GT-MHR radial temperature gradient during after-heat rejection to RCCS.

the pressure vessel (Figure 5.16) and transferred by radiation from the vessel to the natural circulation RCCS without exceeding the fuel particle temperature limit (Figure 5.17).

Even if the RCCS is assumed to fail, passive heat conduction from the core, thermal radiation from the vessel, and conduction into the silo walls and surrounding earth (Figure 5.18) is sufficient to maintain peak core temperatures to below the design limit.

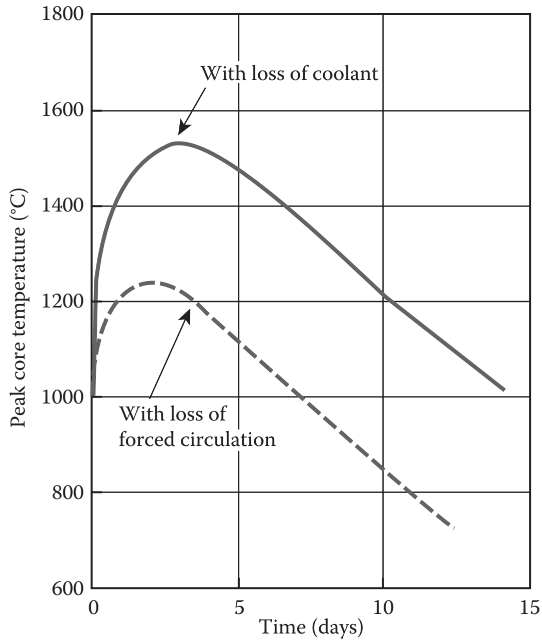


FIGURE 5.17
GT-MHR Core heat-up temperatures with passive after heat rejection.

As a result, radionuclides are retained within the refractory coated fuel particles without the need for active systems or operator action. These safety characteristics and design features result in a reactor that can withstand loss of coolant circulation, or even loss of coolant inventory, and maintain fuel temperatures below damage limits (i.e., the system is meltdown proof). The core graphite heat capacity is sufficiently large that any heatup, or cooldown, takes place very slowly. A substantial time (of the order of days vs. minutes for other reactors) is available to take corrective actions to mitigate abnormal events and to restore the reactor to normal operations.

5.4.4 GT-MHR Environmental Characteristics

The GT-MHR has significant environmental impact advantages relative to light water reactor plants (Table 5.4) between a 4-module GT-MHR plant and a large PWR. The thermal discharge (waste heat) from the GT-MHR is significantly less than the PWR plant because of its greater thermal efficiency. If this waste heat is discharged using conventional power plant water heat rejection systems, the GT-MHR requires <60% of the water coolant per unit of electricity produced. Alternatively, because of its significantly lesser waste heat, the GT-MHR waste heat can be rejected directly to the atmosphere using air-cooled heat rejection systems such that no water coolant resources are needed. Because of this capability, the use of the GT-MHR in arid regions is possible.

The GT-MHR produces less heavy metal radioactive waste per unit energy produced because of the plant's high thermal efficiency and high fuel burnup. Similarly, The GT-MHR produces less total plutonium and Pu^{239} (materials of proliferation concern) per unit of energy produced.

The deep-burn capability and high radionuclide containment integrity of TRISO particles offer potential for improvements in nuclear spent fuel management. A high degree of

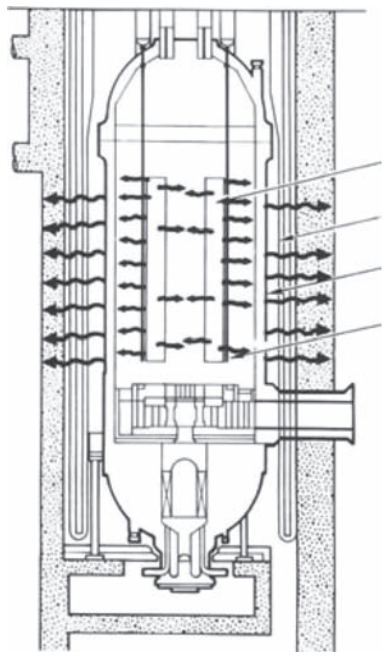


FIGURE 5.18
GT-MHR passive radiation and conduction of after heat to silo containment.

TABLE 5.4
Resource Consumption and Environmental Impact Comparison

Plant Parameters	Large PWR	GT-MHR
• Thermal power (MW(t))	3914	4×600
• Electric power (MWe)	1385	1145
• 60 year power generation (GWY)	72.3	59.8
Thermal Discharge		
• Heat rejection (GWt/GWe)	1.8	1.1
• Cooling water req'd (10^4 Acre-Ft/GWY)	2.4	1.4
Equilibrium Fuel Cycle		
• Heavy metal loading (MT/GWt)	26.8	7.5
• U Enrichment (%)	4.2	15.5 (Avg)
• SWU Demand (103 kg-SWU/GWY)	135	221
• U_3O_8 Consumption (MT/GWY)	181	246
• Full power days per cycle	432	460
Spent Fuel Discharge		
• Discharged heavy metal (MT/GWY)	21.4	5.4
• Discharged Pu (kg/GWY)	235	109
• Discharged Pu^{239} (kg/GWY)	171	43

degradation of plutonium and other long-life fissile actinides can be achieved by the deep-burn capability. Nuclear design analyses of the MHR deep-burn concept indicate that, in one pass through the reactor, virtually complete destruction can be accomplished of weapons-usable materials (plutonium-239), and up to 90% of all transuranic waste, including near total destruction of neptunium-237 (the most mobile actinide in a repository environment) and its precursor, americium-241. The resultant particles contain significantly reduced quantities of long-life radionuclides and very degraded fissile materials that can then be placed in a geologic repository with high assurance the residual products have insufficient interest for intentional retrieval and will not migrate into the biosphere by natural processes before decay renders them benign.

5.5 GT-MHR Fuel Cycles

5.5.1 GT-MHR Uranium Fuel Cycle

A Commercialization option of the GT-MHR has been in development at GA since 1993 to produce electricity at competitive generation costs, and is a promising candidate for near term commercial deployment in the United States. Two different types of fuel TRISO particles are used for power profiling purposes: 19.9% low-enriched (LEU) particles and natural uranium (NU) particles. The current design uses a once-through fuel cycle, refueling half of the core at every reload interval [Shenoy 1996].

5.5.2 GT-MHR Plutonium Fuel Cycle

When fueled with weapon-grade plutonium (94% enriched Pu-239), the GT-MHR can provide the capability to consume >90% of the initially charged PU-239, and >65% of the initially charged total plutonium, in a single pass through the reactor. This option is referred to as a “Plutonium Consumption MHR” (PC-MHR), and is currently under development in a joint United States–Russian Federation program to provide capacity for disposition of surplus weapons plutonium. The current design is also a once-through fuel cycle type, however only one-third of the core is replaced during refueling.

5.5.3 GT-MHR Thorium Fuel Cycles

5.5.3.1 GT-MHR HEU/Th Fuel Cycle

This fuel cycle is based upon Fort St. Vrain type fuel, which operated from 1976 through 1989. Fuel composition consists again of two separate TRISO particles, 93% highly enriched uranium (HEU) particles and fertile Th-232 particles to achieve maximum U-233 conversion ratios and therefore limit the amount of plutonium produced. Although HEU-fueled reactors would not be considered for commercial use in the United States, the interest here is historical in nature. This design also uses a once-through fuel cycle, refueling half of the core at every reload interval.

5.5.3.2 GT-MHR LEU/Th (single particle) Fuel Cycle

This fuel cycle concept was initially conceived at GA in 1977 and promoted as a “non-proliferation” design option because fissile and fertile fuels co-exist in the same TRISO fuel particle. This design effectively denatures the U-233 produced from the fertile Th-232 fuel by mixing it with non-fissile plutonium nuclides generated from the 19.9% LEU. A significant quantity of Pu-238 is also produced so that the plutonium would also generate a considerable amount of decay heat, thereby making the depleted fuel less attractive as bomb material. This design would also use a once-through fuel cycle, refueling half of the core at every reload interval.

5.5.3.3 GT-MHR LEU/Th (dual particle) Fuel Cycle

This fuel cycle is currently being studied as a method for achieving much longer fuel cycle lengths and extended burnup due to an expected higher conversion ratio from the thorium breeding. By separating the 10.9% LEU TRISO particles from the Th-232 TRISO particles, the MHR can simulate the fertile "blanket effect" utilized in fast breeder reactors. Because the bred U-233 would not be denatured here, this may likely be a closed fuel cycle to recycle the bred fissile uranium.

5.5.4 GT-MHR Mixed Actinide Fuel Cycles

5.5.4.1 Deep-Burn MHR (DB-MHR)

This fuel cycle's sole fuel-source uses reprocessed transuranic waste discharged from Light Water Reactors (LWRs). The fissile plutonium (obtained after a AFCI-UREX or other similar process) becomes the main driver fuel for this cycle. Fortunately, the core neutron spectrum allows for significant neutron capture in the resonance region by several minor actinides mixed in with the plutonium. As a result, this design provides its own negative reactivity control without the need for burnable poisons. Over 96% of the initial Pu-239, including over 60% of the initial actinide nuclides can be destroyed in this cycle.

5.5.4.2 Self-Cleaning MHR (SC-MHR)

This fuel cycle combined discharged and recycled TRU waste from an Low Enriched Uranium (LEU) or mixed Low Enriched Uranium/Thorium (LEU/Th) fuel cycle with the fresh fuel. The mixed-core fuel is composed of 80% fresh fuel and 20% discharged and recycled TRU waste. This is therefore essentially a closed-cycle LEU or LEU/Th fuel cycle. Through recycling bred fissile and minor actinide nuclides from a cycle discharge, very high actinide destruction is possible, approximately >80%.

Table 5.5 shows some MHR parameters of all presented fuel cycle options [Ellis 2004].

5.6 MHR Next Generation Potential Applications

5.6.1 Non-Electric Applications

5.6.1.1 Alumina Plant

Aluminum refining uses two major energy-intensive processes:

(1) Aluminum oxide or alumina is obtained from bauxite via the Bayer chemical process. This process uses a significant amount of steam to react with bauxite and for mechanical drive. It also requires electric power.

TABLE 5.5

MHR Parameters of all Fuel Cycle Options

MHR Fuel Cycle Option	LEU/NU	LEU/Th	PC-MHR	DB-MHR	SC-MHR
Number of TRiso particle types	2	2	1	2	2
Fuel description	$UC_{0.5}O_{1.5}$	$UC_{0.5}O_{1.5}/ThO_2$	$PuO_{1.7}$	TRU Oxides	LEU/Th TRU Oxides
Reactor thermal power (MW(t))	600	600	600	600	600
Fuel cycle length (EFPD)	477	950[a]	317	540	540[a]
Net thermal efficiency (%)	47.7	47.7	47.7	47.7	47.7
Average EC Uranium enrichment (%)	15.5	19.9	0	0	19.9
Average EC plutonium enrichment (%)	0	0	94	60	60
EC Uranium loading (kg)	2,262	1,552[a]	0	0	1,242[a]
EC Thorium loading (kg)	0	710[a]	0	0	568[a]
EC Plutonium loading (kg)	0	0	369	237	47[a]

Notes: EC = Equilibrium Cycle, EFPD = Effective Full Power Days.
[a] Best Estimate.

(2) Alumina is reduced to aluminum by electrolysis. This process requires large amounts of electric power.

Most existing commercial aluminum plants use energy from natural gas power plants. Hydroelectric power supplies a very small fraction of the total aluminum electric power requirements. An MHR could be utilized for producing alumina from bauxite. For the size alumina plant considered, a two module 600-MW(t) PS/C-MHR can supply 100% of the process steam and electrical power requirements and produce surplus electrical power and/or process steam, which can be used for other process users or electrical power production. Presently, the bauxite ore is reduced to alumina in plant geographically separated from the electrolysis plant. However, with the integration of 2 × 600 MW(t) PS/C-MHR units in a commercial alumina plant, the excess electric power available, 233 MW(e), could be used for alumina electrolysis. It has been shown the steam and electrical energy requirements for a typical commercial alumina plant processing 726,680 tonnes (800,000 tons) per year of alumina (Al_20_3) can be satisfied by a two module PS/C-MHR [Shenoy 1995].

5.6.1.2 Coal Gasification

Several countries are interested in developing plants producing gaseous synthetic fuels derived from coal, based on their national objective to reduce foreign oil imports and to use or export the abundant coal. Exxon catalytic coal gasification (ECCG) is one gasification process developed in the United States. Initially, coal gasification plants are expected to obtain thermal power requirements from fossil sources (coal or product liquid and gaseous fuel from the synfuel plant) and to obtain electric power partly from in-plant cogeneration and partly from local utilities. Most processes are estimated to consume 25% to 30% of the feed coal to satisfy the plant energy needs. The ECCG process uses alkali metal salts as a gasification catalyst with a novel processing sequence. Although no net heat is required for the gasification reaction, heat input is required for drying and preheating the feed coal, gasifier heat losses, and catalyst recovery operations. Mechanical drives and plant electrical power also have energy input requirements. An MHR could provide thermal and electrical energy for the ECCG process to benefit worldwide interests by conserving fossil fuel and reducing environmental impact.

5.6.1.3 Coal Liquification

The solvent-refined coal (SRC-II) process is an advanced process developed by Gulf Mineral Resources Ltd. to produce a clean, nonpolluting liquid fuel from high sulfur bituminous coals. Coupling of two module 600 MW(e) PS/C-MHR to the SRC-II process could commercially process 24,300 tonnes (26,800 tons) of feed coal per stream day, producing primarily fuel oil and secondary fuel gases [Shenoy 1995].

In the SRC-II process, the process steam is generated by direct gas-fired boilers, and the process heating by direct gas firing. The fuels utilized are hydrocarbon-rich gas, or CO-rich gas, and purified syngas (i.e., no feed coal is used for fuel). It was shown that a 2 × 600 MW(t) PS/C-MHR can supply these thermal requirements principally by substituting for the fuel gases previously employed [Shenoy 1995]. The displaced gases, which are treated already, may then be marketed.

The 538°C (1000°F) steam supply of the PS/C-MHR provides all system thermal energy requirements in the form of process steam generation, steam superheating, and slurry heating. However, slurry heating by steam will entail the development of a new heat exchanger design. The 2 × 600 MW(t) PS/C-MHR does not generate all the required electrical energy, and a deficit of –38 MW(e) results.

5.6.1.4 H-Coal Liquefaction

In countries with large coal reserves, a strong interest exists to develop and commercialize plants producing liquid and gaseous synthetic fuels derived from coal because of their national objective to reduce foreign oil imports or to export liquid coal. The H-Coal liquefaction is one process which can be used to convert coal into liquid fuel. The H-Coal process has several advantages over other processes, including an isothermal reactor bed, hyrogeneration of the coal with a direct, continuously replaceable catalyst (i.e., no dependence on catalytic effects of coal ash), and the absence of quench injections (which would be required with a series of fixed beds).

5.6.1.5 Hydrogen Production

A significant "Hydrogen Economy" is predicted to limit dependence on petroleum and reduce pollution and greenhouse gas emissions. Hydrogen is an environmentally attractive fuel but contemporary hydrogen production is primarily based on fossil fuels. The United States produces ~11 million tons of hydrogen a year by steam reformation of methane for use in refineries and chemical industries and the use is growing by ~10% per year. This is the thermal energy equivalent of 48 GW(t), and consumes about 5% of our natural gas usage. Use of hydrogen for all the transportation energy needs in the United States would require a factor of 18 more hydrogen than currently used. Clearly, new sources of hydrogen will be needed. Nuclear energy can be one of the sources.

Hydrogen can be produced from nuclear energy by several means. Electricity from nuclear power can separate water into hydrogen and oxygen by electrolysis [Richards 2006b]. The net efficiency is the product of the efficiency of the reactor in producing electricity, times the efficiency of the electrolysis cell, which, at the high pressure needed for distribution and utilization, is about 75–80%. If a GT-MHR with 48% electrical efficiency is used to produce the electricity, the net efficiency of hydrogen production could be about 36–38%. Electrolysis at high temperature, providing some of the energy directly as heat, promises efficiencies of about 50% at 900°C. Thermochemical water-splitting processes similarly offer the promise of heat-to-hydrogen efficiencies of ~50% at high temperatures. Thermochemical water-splitting is the conversion of water into hydrogen and oxygen by a series of thermally driven chemical reactions that could use nuclear energy as the heat source.

The Sulfur-Iodine (S-I) thermochemical water-splitting cycle has been determined to be best suited for coupling to a nuclear reactor [Richards 2006a]. The S-I cycle (Figure 5.19) consists of three chemical reactions, which sum to the dissociation of water. Only water and high temperature process heat are input to the cycle and only hydrogen, oxygen and low temperature heat are output. All the chemical reagents are regenerated and recycled. There are no effluents. An intermediate helium heat transfer loop would be used between the primary coolant loop and the hydrogen production system. With an outlet temperature of 850°C, a maximum temperature of 825°C is estimated for the process heat to the process, which yields 43% efficiency. At a reactor outlet temperature of 950°C and a 50°C temperature drop across an intermediate heat exchanger, an efficiency of 52% is estimated.

5.6.1.6 Steel Mill

The U.S. steel industry is very large and consumes large quantities of energy. It uses 35% of this energy in the form of electricity, fuel oil, or natural gas; the balance is coal. Therefore, the supply of the non-coal energy by an MHR can conserve scarce fossil fuel resources.

A 2×600 MW(t) PS/C-MHR plant can satisfy the energy requirements for a typical commercial steel mill to produce 6.5×10^6 tonnes (7.2×10^6 tons) (liquid) of steel per year [Shenoy 1995]. The surplus energy, which may be generated either as steam at 5.0 MPa (725 psia) and 365°C (689°F) at 125 kg/s (106 Il/hr) or ~ electric power [–100 MW(e)], can be exported

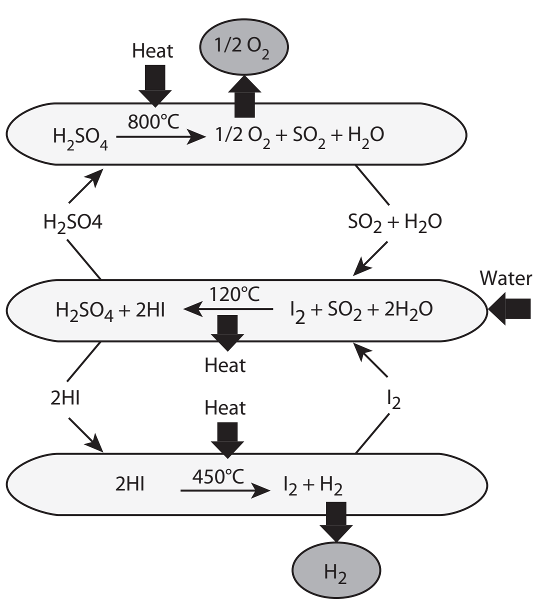


FIGURE 5.19
Sulfur-Iodine thermochemical water-splitting cycle.

outside the plant. Depending on the steel mill location, steam could be supplied to neighboring industries or, alternatively, the electric power can be sold to a utility.

5.6.1.7 Synthetic Fuels

Increasing world demand for oil, and the perception that conventional (i.e., liquid) oil reserves have peaked, has caused oil prices to sky rocket and has renewed interest in unconventional oil reserves in the form of oil shale and tar sands, which are estimated to hold several times more oil than the current liquid reserves. It has also spurred interest in synthetic fuel production and in better methods for recovery of heavy oil from operating wells where production is dropping. Economic oil recovery from any of these areas requires high-temperature gas or high-temperature steam. An HTGR is uniquely suited for these applications because it can produce the high-temperature gas and the high-temperature steam at the conditions required for these processes in an environmentally acceptable manner (i.e., without burning natural gas). It can also co-produce electricity for oil field and on-site uses, and the hydrogen needed to convert the hydrogen deficient, heavy crude, into a refinable, syncrude product.

GA's objective for the synthetic fuels program is to develop pre-conceptual MHR designs for each of these applications, including good cost estimates and construction schedules, which can be used to obtain Government agency (DOE, DoD), or Oil company funding for detailed design studies leading to the construction of a demonstration MHR plant. The PS/C-MHR version of the reactor will be used because it is based on Peach Bottom and Fort St. Vrain experience, has been reviewed at the Preliminary Safety Information Document (PSID) level by the Nuclear Regulatory Commission, has realistic cost estimates, and could be put on line in a short time frame without a large technology development program.

5.6.1.8 Research/Test Reactors

In 2006, the University of Texas of the Permian Basin (UTPB) made a partnership with GA, The University of Texas System, and with the participation of local city and county governments as well as with the collaboration of other academic, industrial, and government laboratories, and proposed to construct and operate a High-Temperature Teaching & Test Reactor (HT^3R) as a multifaceted energy research facility. Its proposed location is near the UTPB campus in Andrews County, Texas, and it is projected to be operational by 2012.

The mission of the HT^3R is to be a research and test facility that can support the education and training of the next generation of nuclear scientists and engineers, as well as the performance of high-temperature R&D on materials and processes for the economic production of electricity, hydrogen, synthetic hydrocarbon fuels, and desalinated water. This primary mission will be supported by facilities for research, development and pilot scale testing programs including a radiation laboratory, high temperature materials and process laboratory, and an energy transfer laboratory. The HT^3R will be the cornerstone for a new UTPB research and development "Center of Excellence" that will investigate new frontiers in the applications of high-temperature materials, processes, plus nuclear science and engineering R&D. The HT^3R will be an HTGR with passively safe design features. The HTGR is also a leading candidate for the development of Generation IV reactors meant to provide significant improvement over existing power reactors with regards to safety, economics, proliferation-resistant fuel cycles, and flexibility of applications. Outlet temperatures of 850°C to >950°C will lead to a variety of applications with the potential for significantly higher thermal efficiencies.

The proposed design of the HT^{3}R and its associated facilities are synergistic with the proposed Next Generation Nuclear Plant (NGNP) authorized by Congress for deployment at the Idaho National Laboratory, as well as the Global Nuclear Energy Partnership (GNEP) that has been proposed by the U.S. President. With the planned physical and operating characteristics of the HT3R being very similar to the proposed commercial scale HTGR plants, the HT3R can significantly benefit the NGNP development by reducing key identified risks.

5.6.2 Proliferation Resistance Applications

The GT-MHR has very high proliferation resistance due to low fissile fuel volume fractions and the refractory characteristics of the TRISO fuel particle coating system that forms a containment from which it is difficult to retrieve fissile materials.

GT-MHR fresh fuel and spent fuel have higher resistance to diversion and proliferation than the fuel for any other reactor option. The GT-MHR fresh fuel has high proliferation resistance because the fuel is very diluted by the fuel element graphite (low fuel volume fraction). GT-MHR spent fuel has the self-protecting, proliferation resistance characteristics of other spent fuel (high radiation fields and spent fuel mass and volume). However, GT-MHR spent fuel has higher proliferation resistance than any other power reactor fuel because of the reasons given below.

- The quantity of fissile material (plutonium and uranium) per GT-MHR spent fuel element is low (50 times more volume of spent GT-MHR fuel elements would have to be diverted than spent light water reactor fuel elements to obtain the same quantity of plutonium-239).
- The GT-MHR spent fuel plutonium content, the material of most proliferation concern, is exceedingly low in quantity per spent fuel block and quality because of high fuel burnup. The discharged plutonium isotopic mixture is degraded well beyond light water reactor spent fuel making it particularly unattractive for use in weapons.
- No process has yet been developed to separate the residual fissionable material from GT-MHR spent fuel. While development of such a process is entirely feasible (and potentially desirable sometime in the future) there is no existing, readily available process technology such as for spent light water reactor fuel. Until such time as when the technology becomes readily available, the lack of the technology provides proliferation resistance.

The TRISO fuel particle coating system, which provides containment of fission products under reactor operating conditions, also provides an excellent barrier for containment of the radionuclides for storage and geologic disposal of spent fuel. Experimental studies have shown the corrosion rates of the TRISO coatings are very low under both dry and wet conditions. The coatings are ideal for a multiple-barrier, waste management system. The measured corrosion rates indicate the TRISO coating system should maintain its integrity for a million years or more in a geologic repository environment.

5.6.3 Sustainability Applications

GA is currently performing parametric studies on the LEU/Th (dual particle) fuel cycle in hopes of utilizing the thorium breeding to significantly expand the fuel cycle length while requiring less fuel ore.

References

Baxter, A.M., McEachern, D.W., Hanson, D.L., Vollman, R.E. 1994. FSV experience in support of the GT-MHR reactor physics, fuel performance, and graphite. General Atomics, Document GA-A21925, November 1994.

Ellis, C. and Baxter, A.M. 2004. Modular helium reactor fuel cycle concepts and sustainability. HTR China Proceedings, 2004. Tsinghua University, Beijing, China.

Hanson, D.L. 2004. A review of radionuclide release from HTGR cores during normal operation. EPRI Report 1009382, Electric Power Research Institute, February 2004.

IAEA. 1997. Fuel performance and fission product behavior in gas cooled reactors. IAEA-TECDOC-978, International Atomic Energy Agency, November 1997, Vienna, Austria.

Labar, M., Shenoy, A., Simon, W., and Campbell, M. 2003. The gas turbine modular helium reactor. *Nuclear News* 46, no. 11: 28.

Melese, G. and Katz, R. 1984. Thermal and flow design of helium-cooled reactors. American Nuclear Society, La Grange Park, Illinois.

PBMR. 2005. General overview of the pebble bed modular reactor. 031865/1, October 2005.

Richards, M. 2006a. H2-MHR conceptual design report: SI-Based Plant. GA-A25401, General Atomics, April 2006.

Richards, M. 2006b. H2-MHR conceptual design report: HTE-based plant. GA-A25402, General Atomics, April 2006.

Shenoy, A. 1995. *Modular helium reactor for non-electric applications of nuclear energy*. General Atomics, Document GA-A22701, November 1995.

Shenoy, A. 1996. GT-MHR conceptual design description report. General Atomics, Document 910720, Rev. 1, July 2006.

USDOE. 2002. Generation IV international forum. A technology roadmap for Generation IV nuclear energy systems. USDOE, December 2002.

Williams, P.M., Silady, F.A., Dunn, T.D., Noren, R.C., McEachern, D.W., Dilling, D.A., and Berkoe, J.M. 1994. MHTGR Development in the United States. *Progress in Nuclear Energy* 28, no. 3: 265–346.