Nuclear Engineering Handbook

Heavy Water Reactor

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Heavy Water Reactors

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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

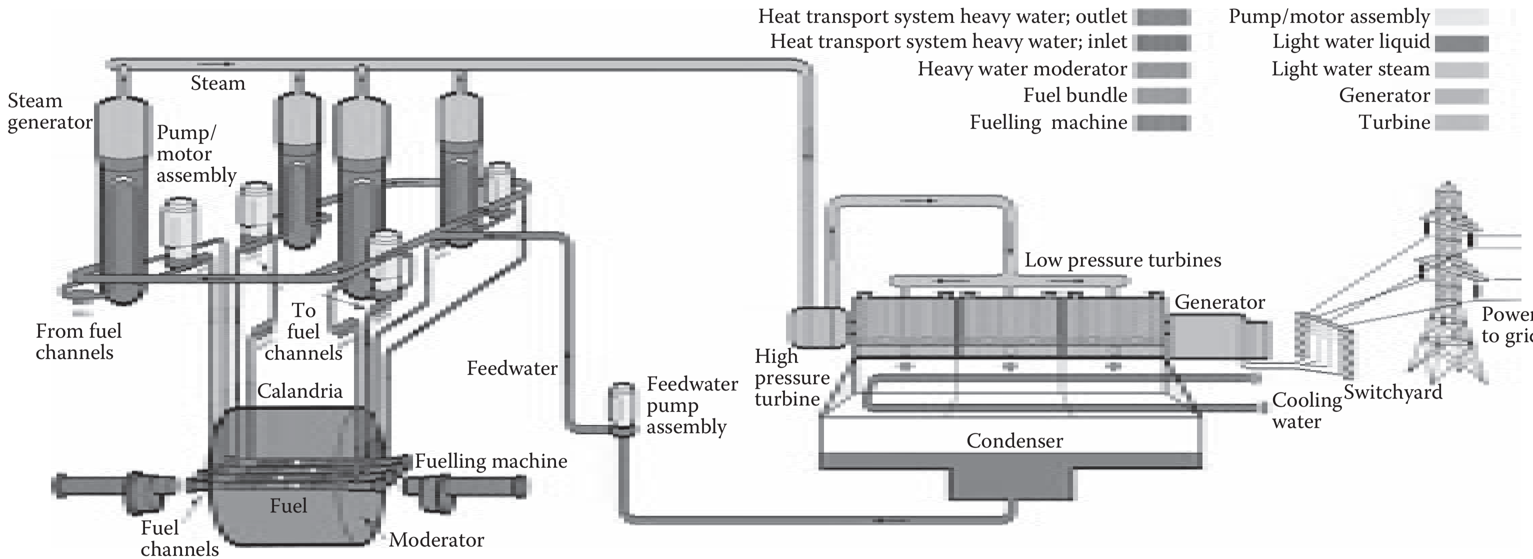


FIGURE 4.1
Nuclear steam supply system.

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

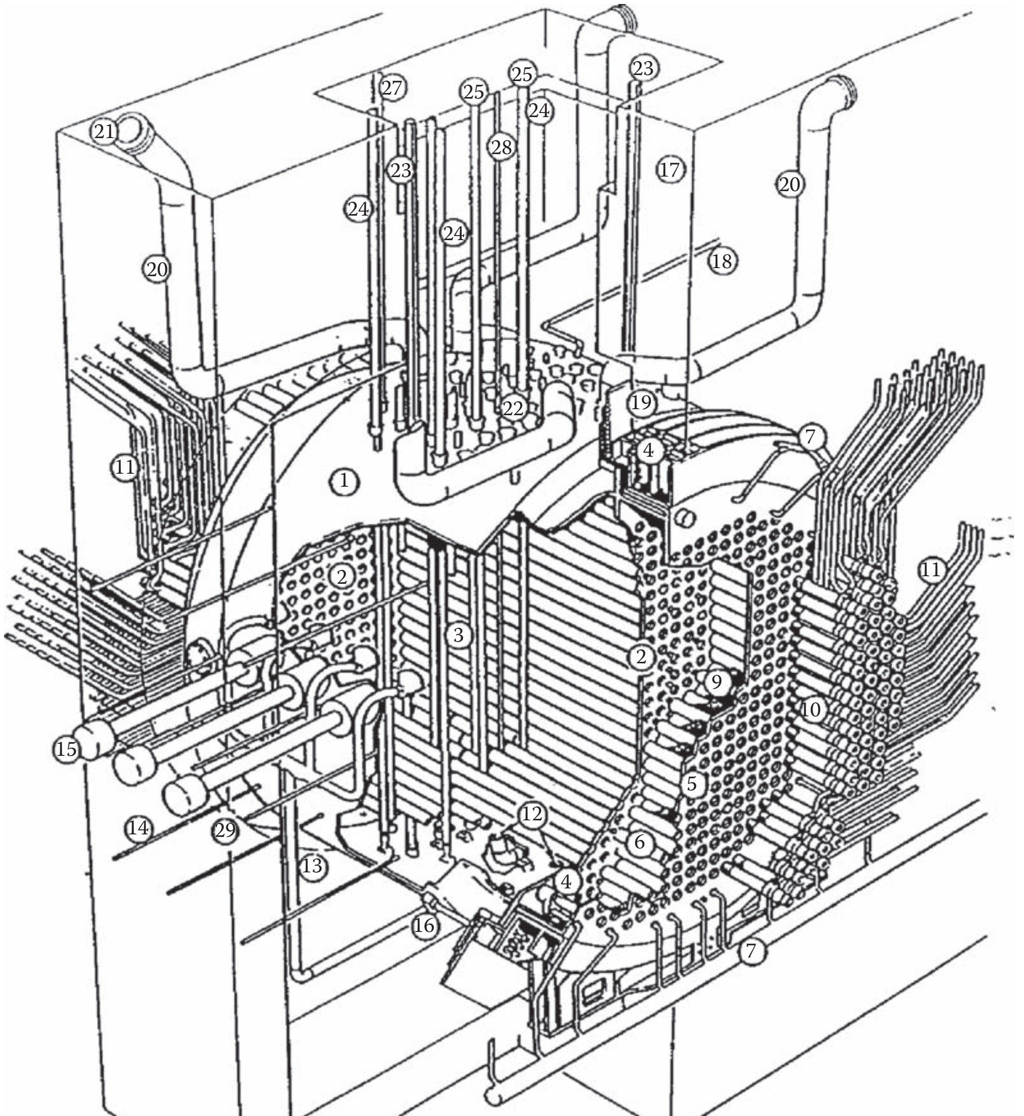
Four countries have evaluated the reactor system in which light water is brought to boiling in vertically oriented pressure tubes, the steam–water mixture being sent to steam drum separators and the steam used directly to drive a turbine. The arrangement simulates the conventional recirculation boiler.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

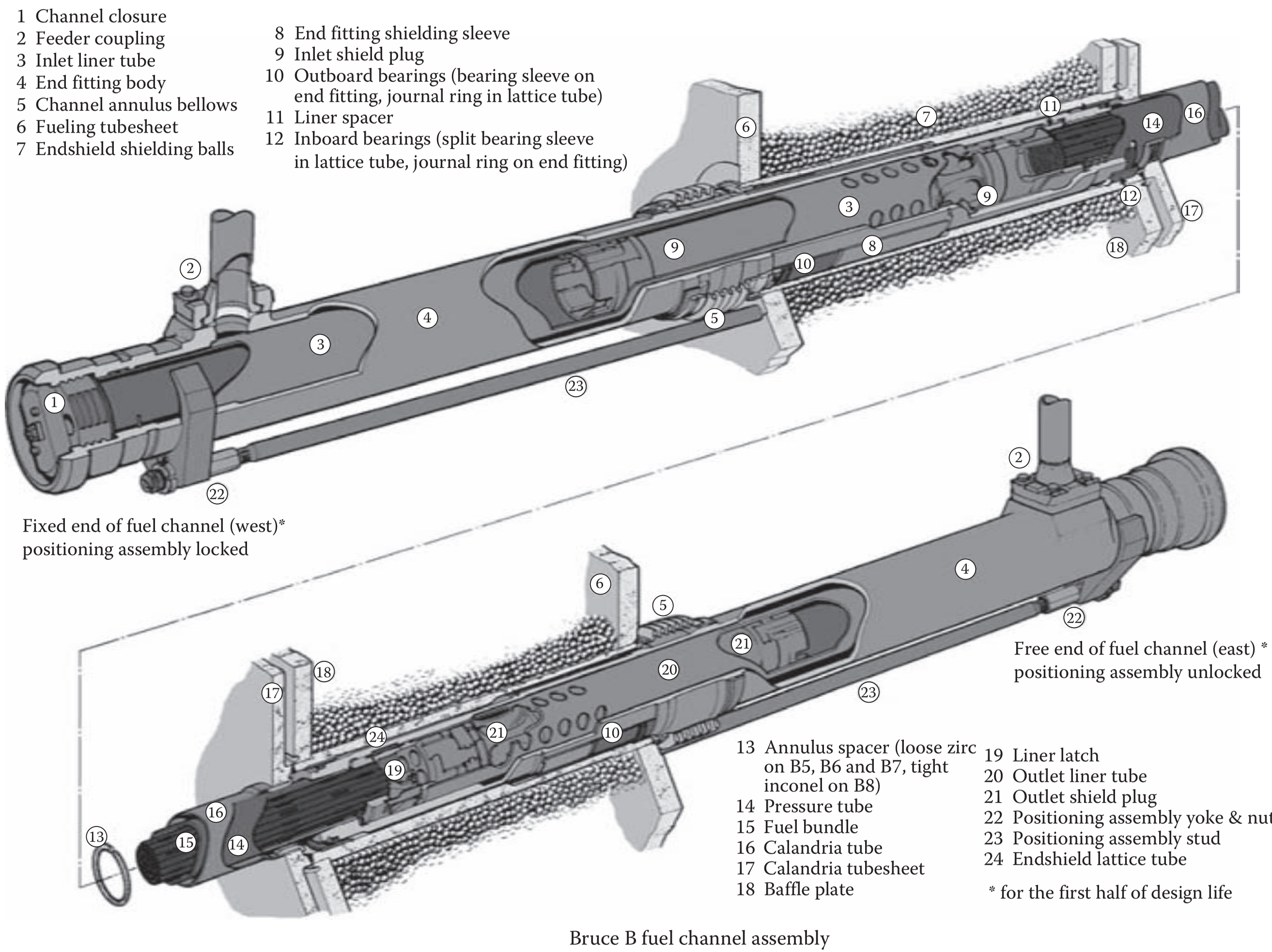


FIGURE 4.3
Bruce B fuel channel assembly.

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

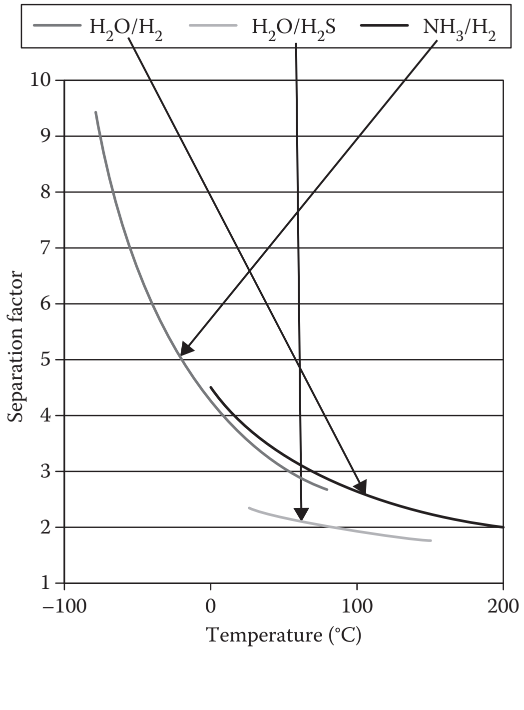


FIGURE 4.4
Equilibrium constants.

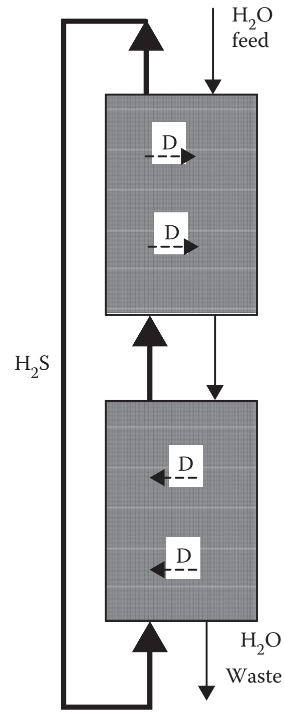


FIGURE 4.5
Schematic of a bithermal water–hydrogen sulfide process.

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.

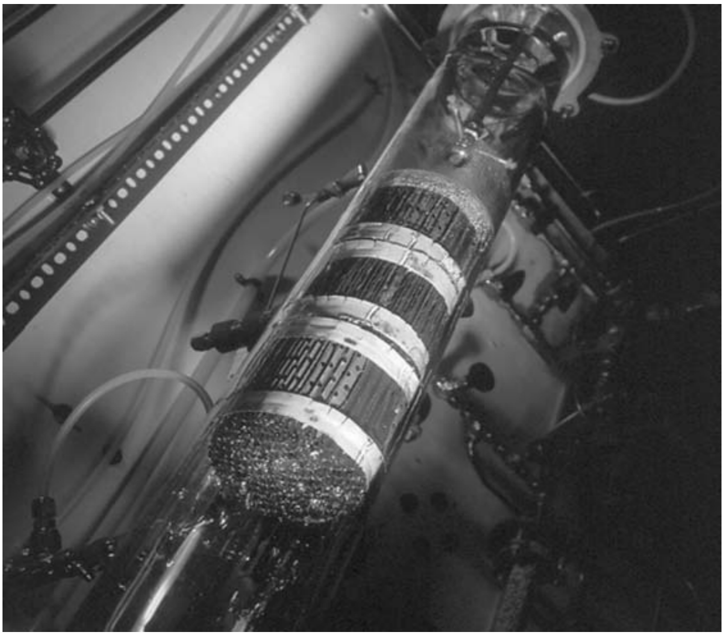


FIGURE 4.7
Structured wetproofed catalyst for water–hydrogen exchange.

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.

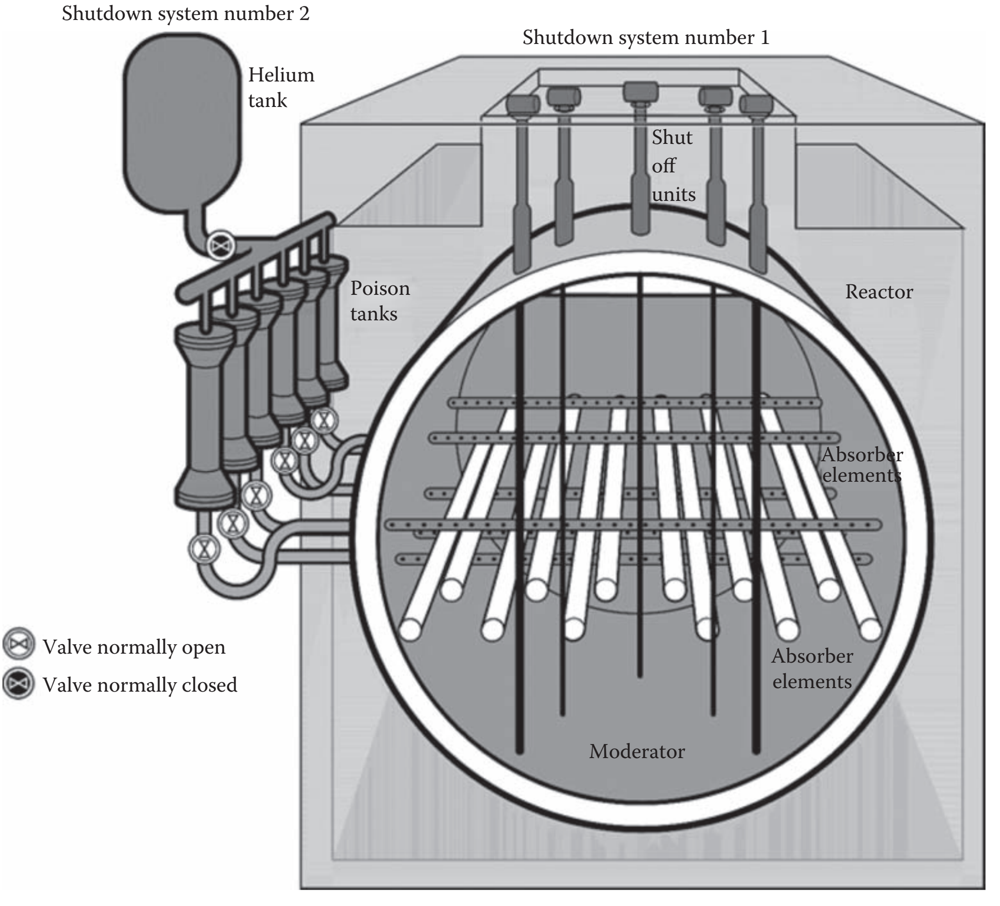


FIGURE 4.9
Shutdown systems.

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

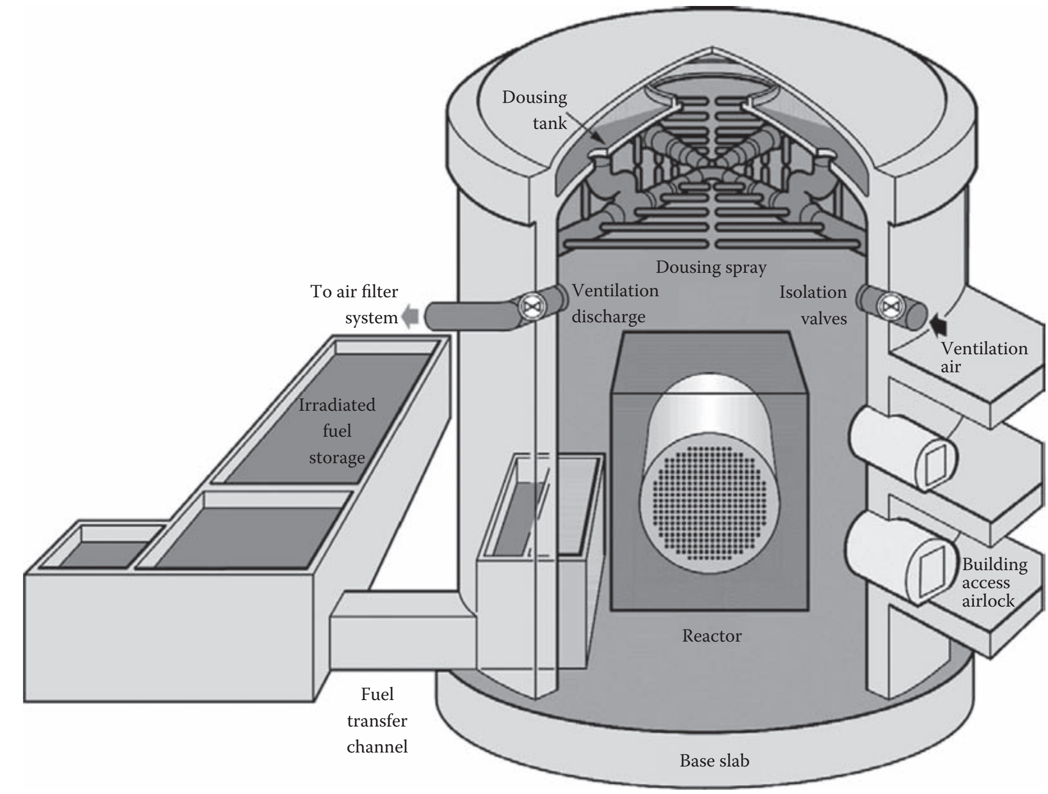


FIGURE 4.10
Single unit containment.

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,

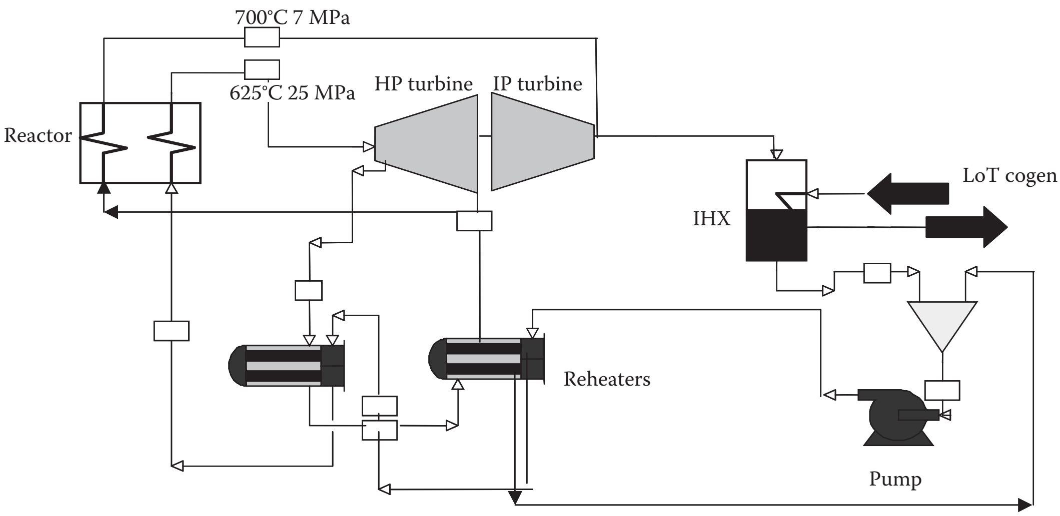


FIGURE 4.11
CANDU CWR (schematic).

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

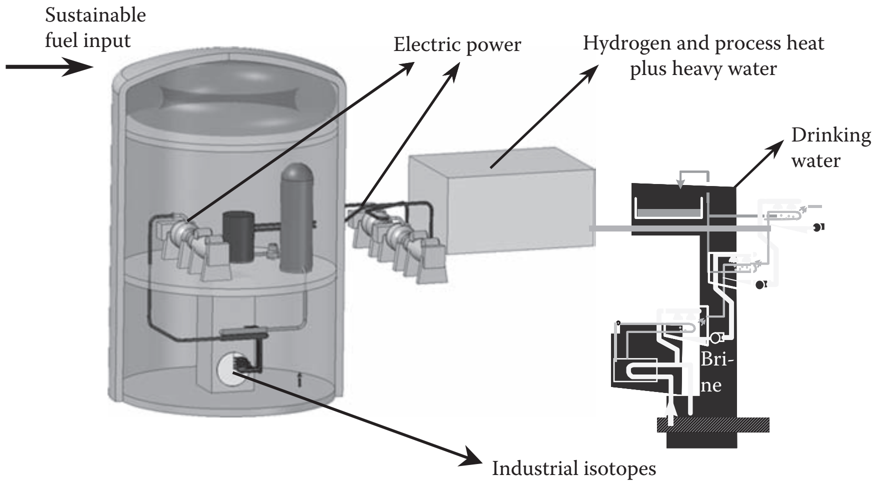


FIGURE 4.12
CANDU X concept co-producing hydrogen, process heat and distilled water alongside electricity.

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

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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.

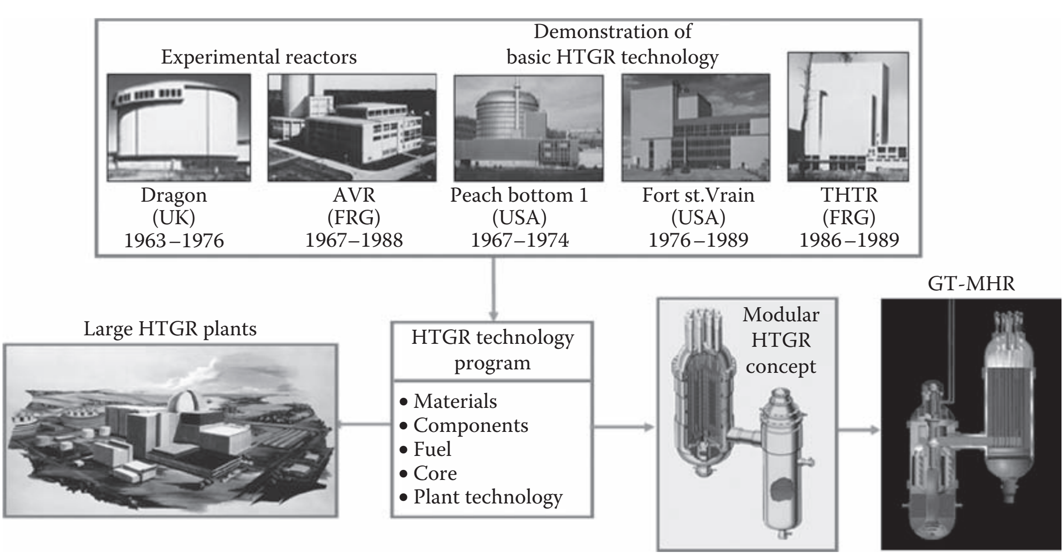

FIGURE 5.1
Broad global foundation of helium reactor technology.

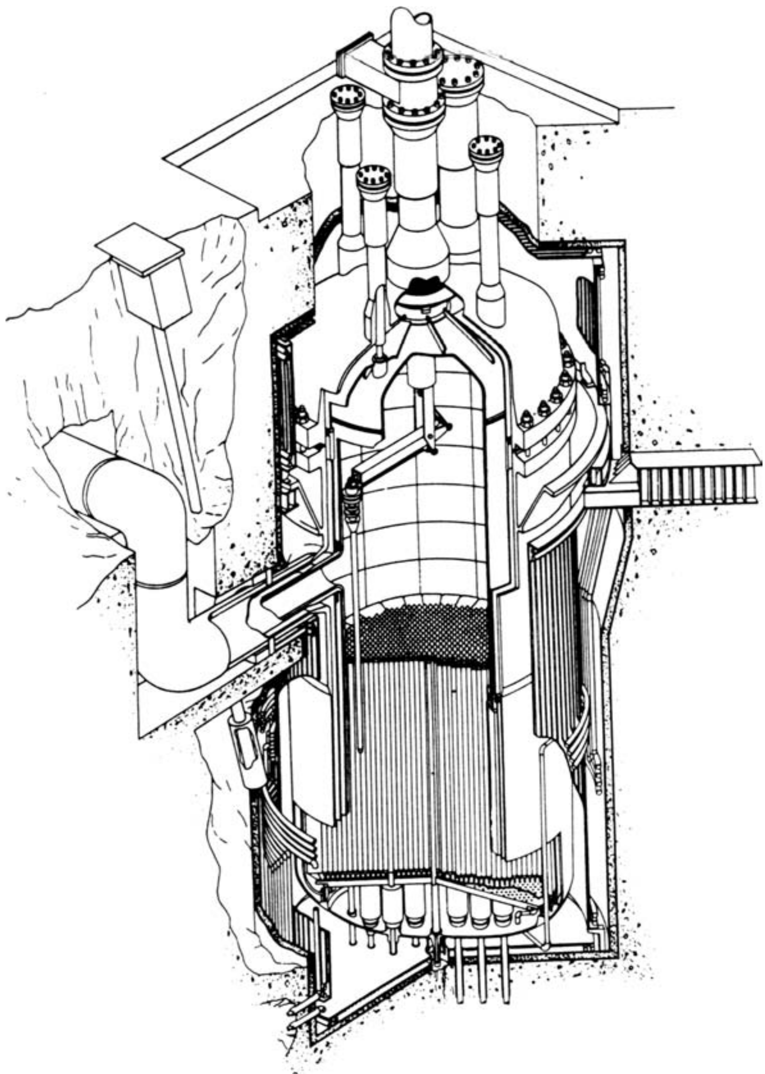

FIGURE 5.2
Isometric view of the peach bottom reactor.

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.

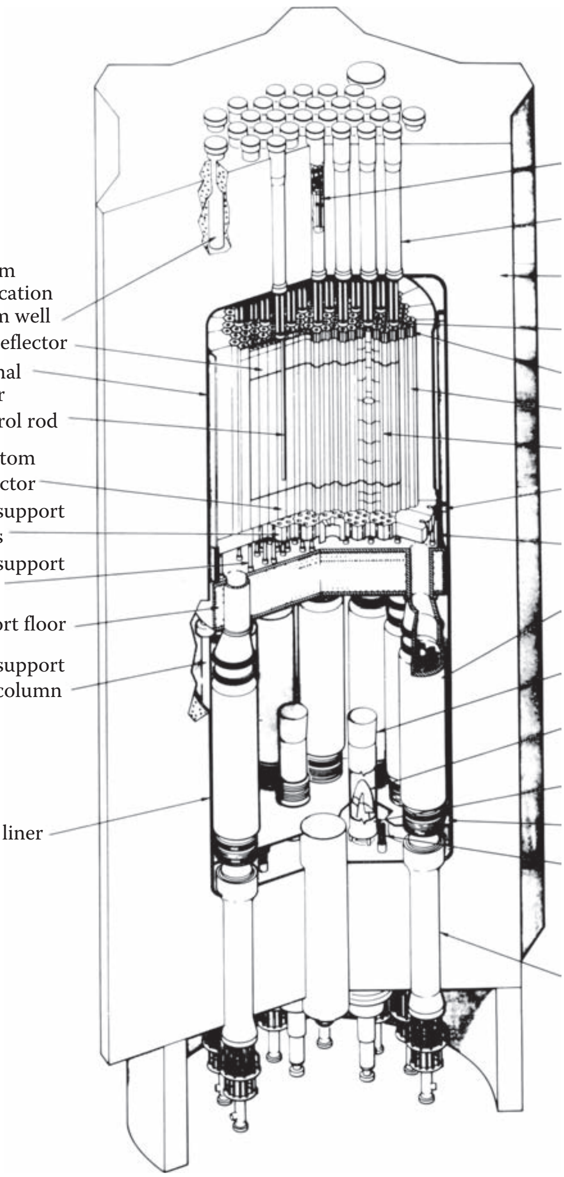

FIGURE 5.3
Isometric view of FSV PCRV, core and primary system.

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.

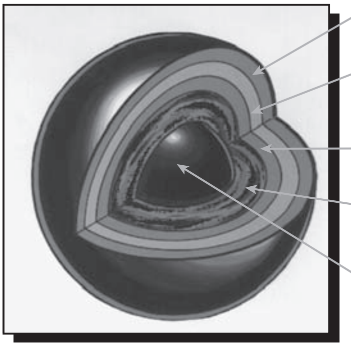

FIGURE 5.4
TRISO coated fuel particle.

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].

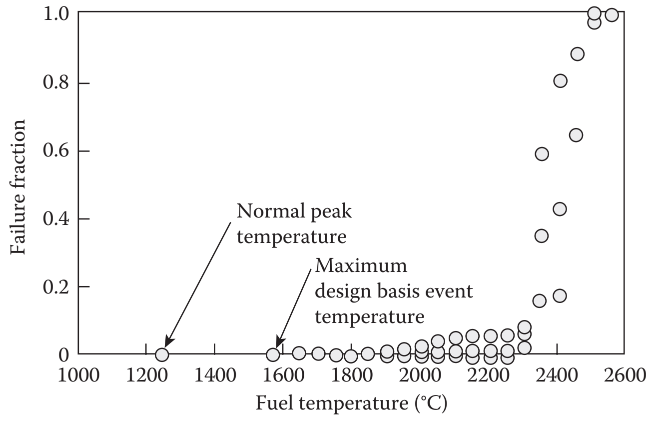

FIGURE 5.5
TRISO coated particle fuel temperature capability.

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

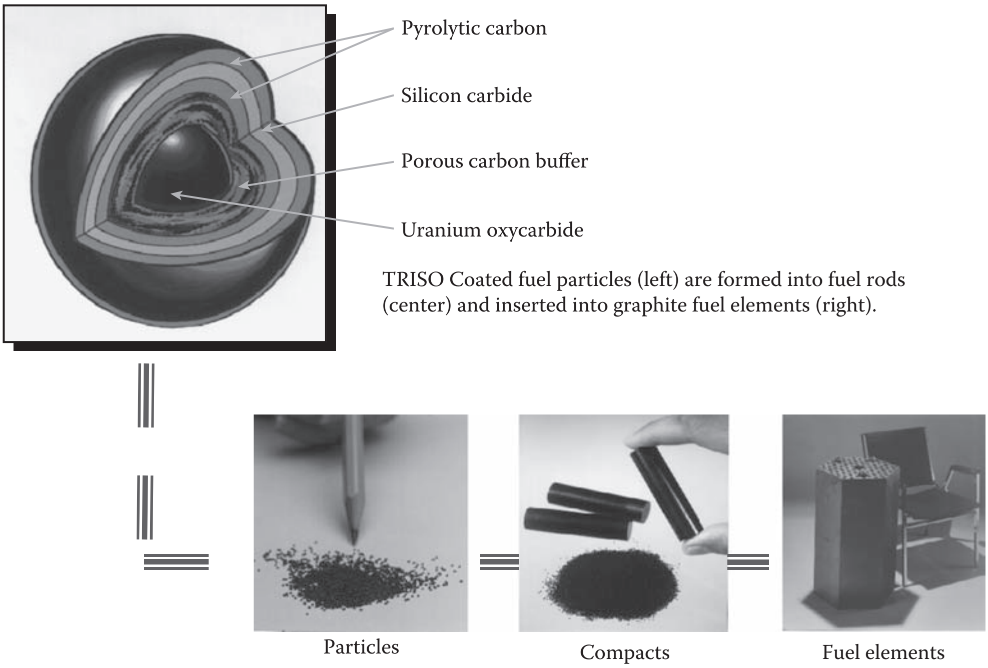

FIGURE 5.6
TRISO coated particle fuel arrangement in hexagonal fuel elements.

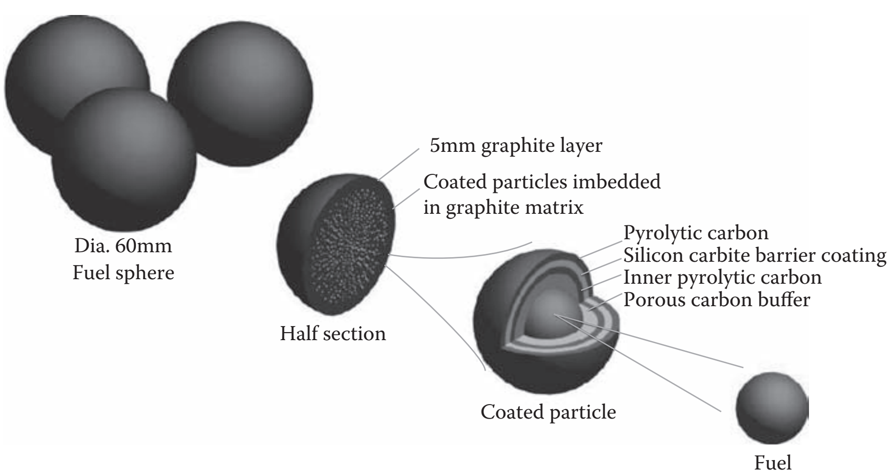

FIGURE 5.7
TRISO coated particle fuel arrangement in spherical fuel elements.

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

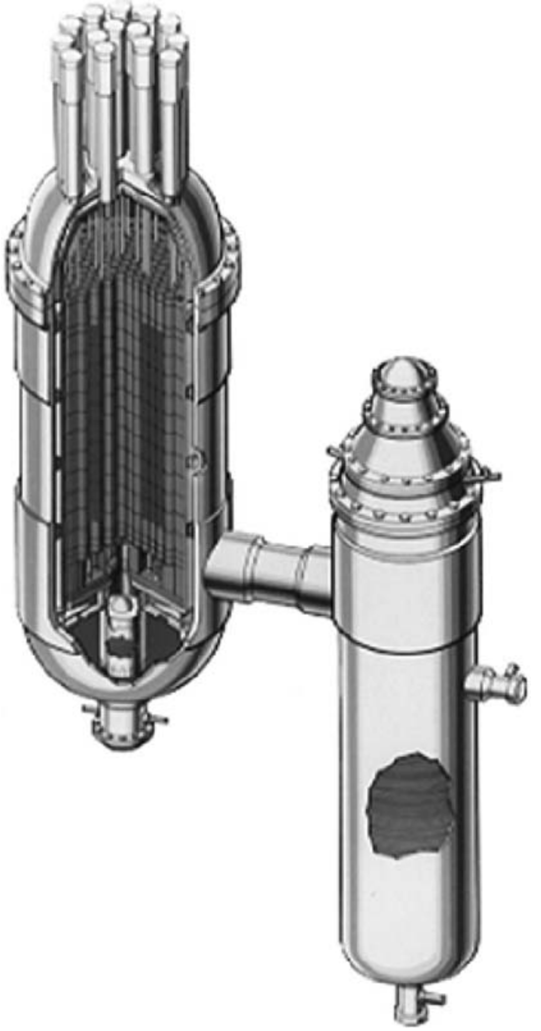

FIGURE 5.8
MHTGR steam cycle plant vessels.

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

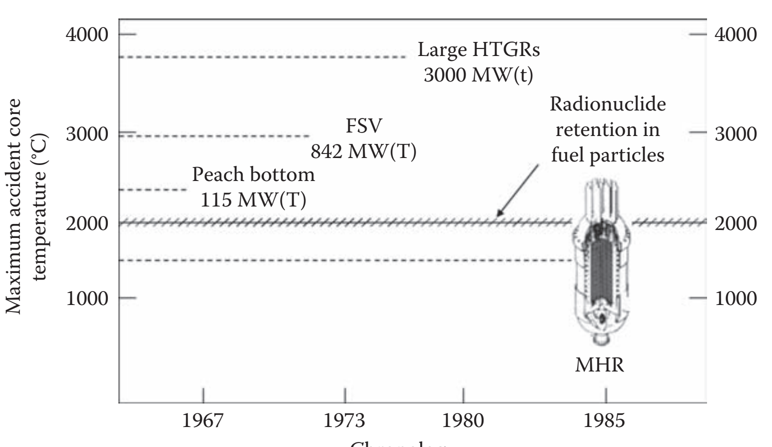

FIGURE 5.9
HTGR prismatic block evolution into the MHR.

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.

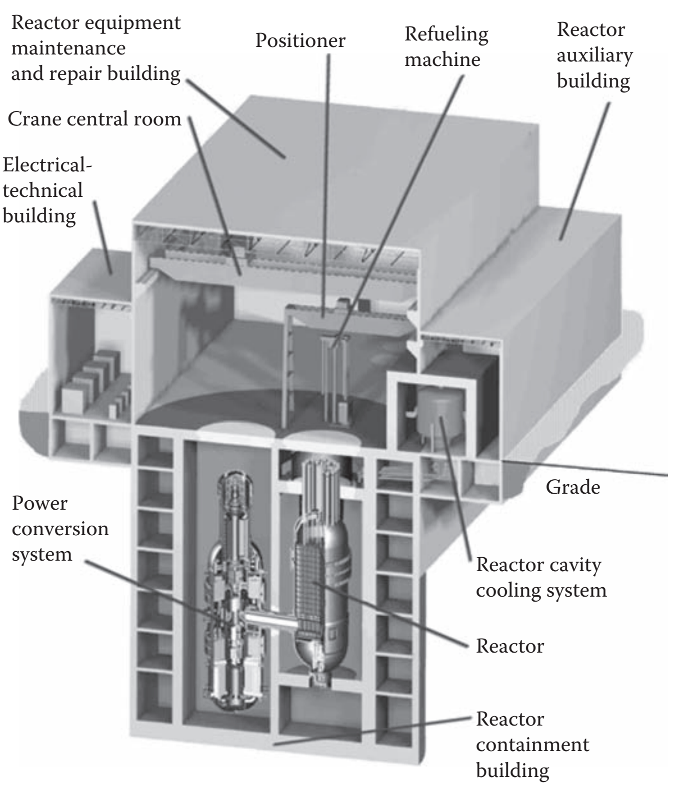

FIGURE 5.10
GT-MHR module.

- 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

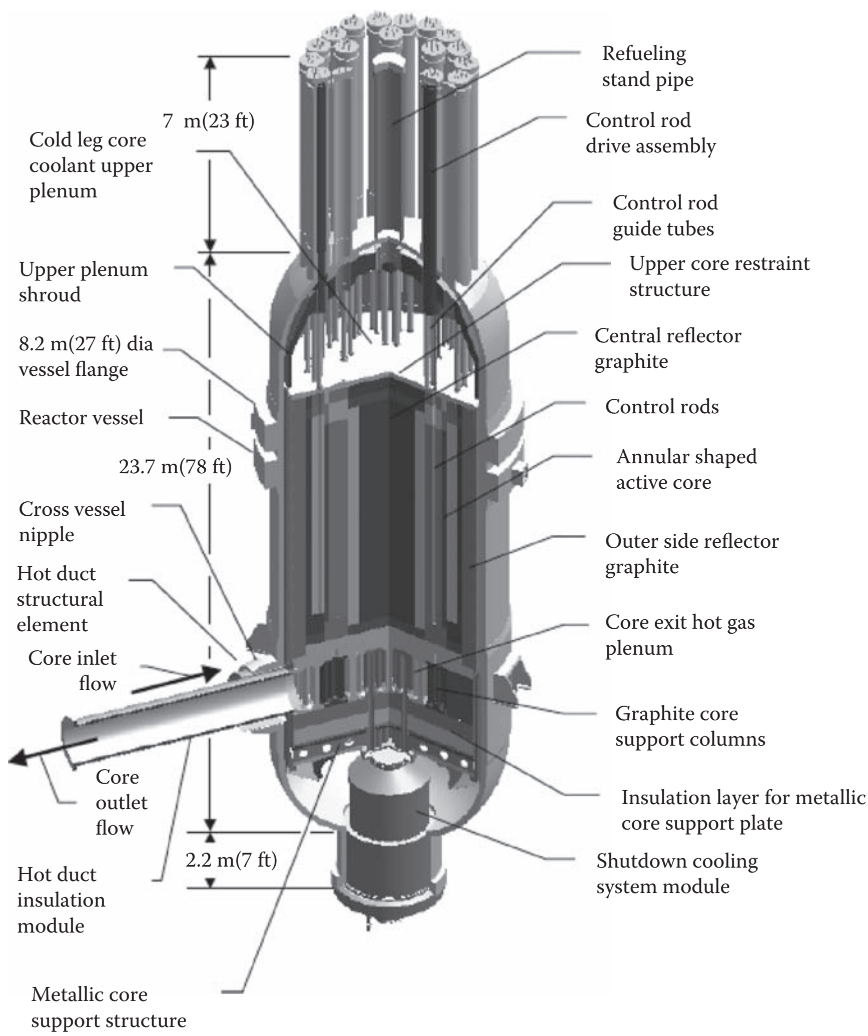

FIGURE 5.11
GT-MHR reactor system.

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

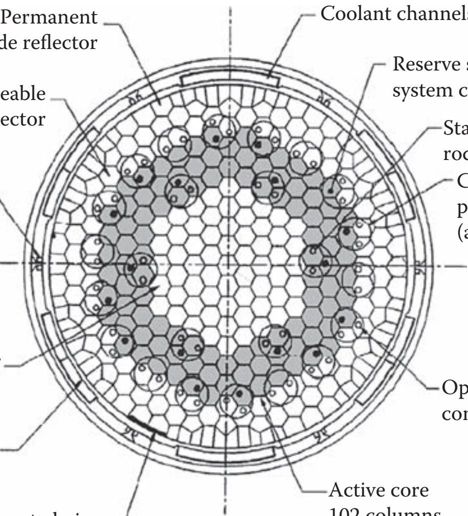

FIGURE 5.12
GT-MHR annular core cross section at vessel mid-plane.

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/cm^3)		
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.

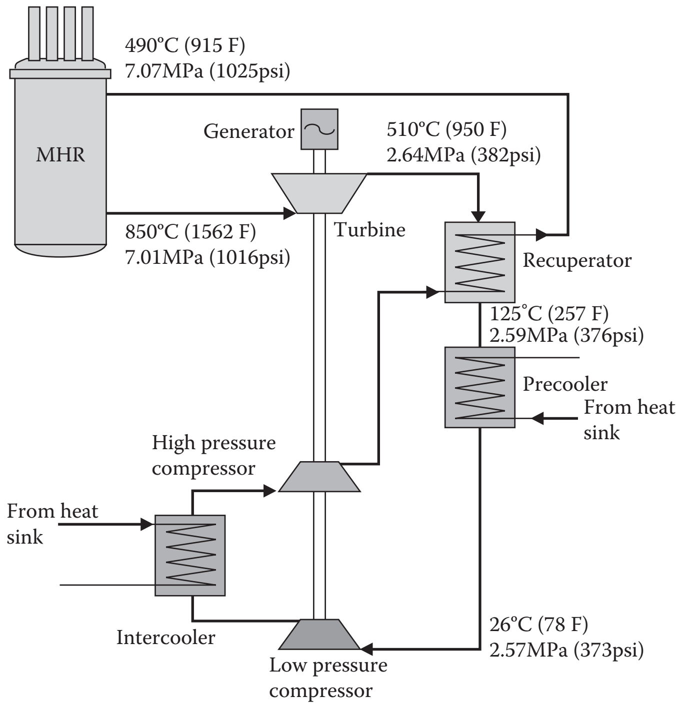

FIGURE 5.13
GT-MHR coolant flow schematic.

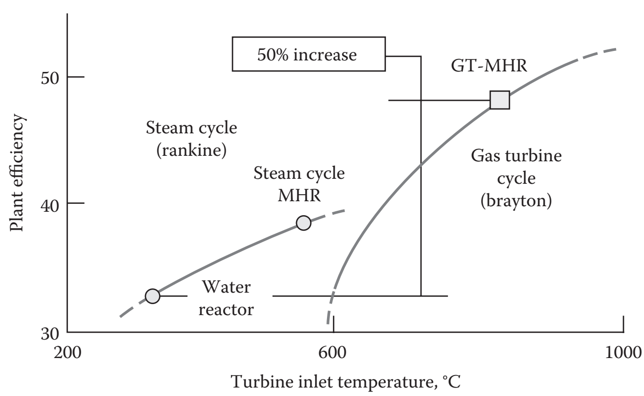

FIGURE 5.14
Comparison of thermal efficiencies.

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

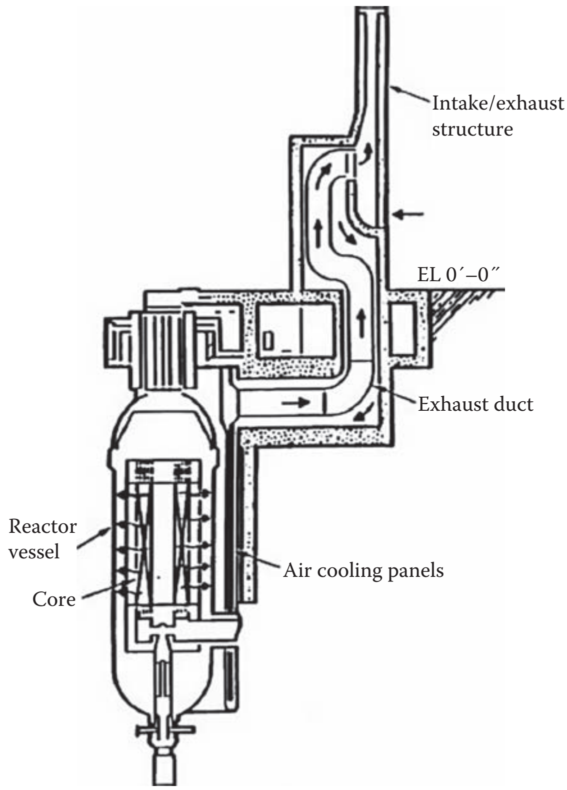

FIGURE 5.15
GT-MHR passive reactor cavity cooling system.

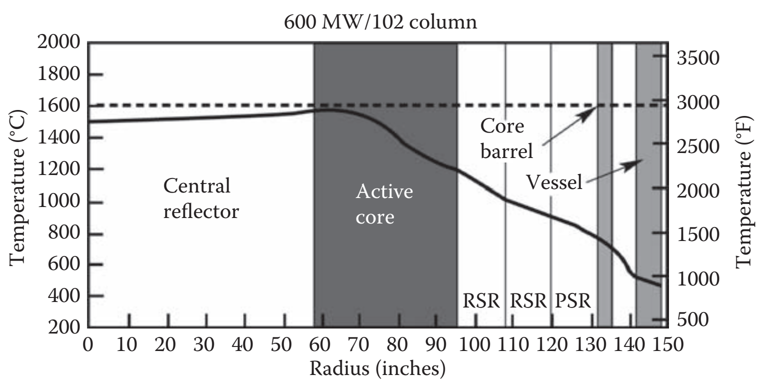

FIGURE 5.16
GT-MHR radial temperature gradient during after-heat rejection to RCCS.

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.

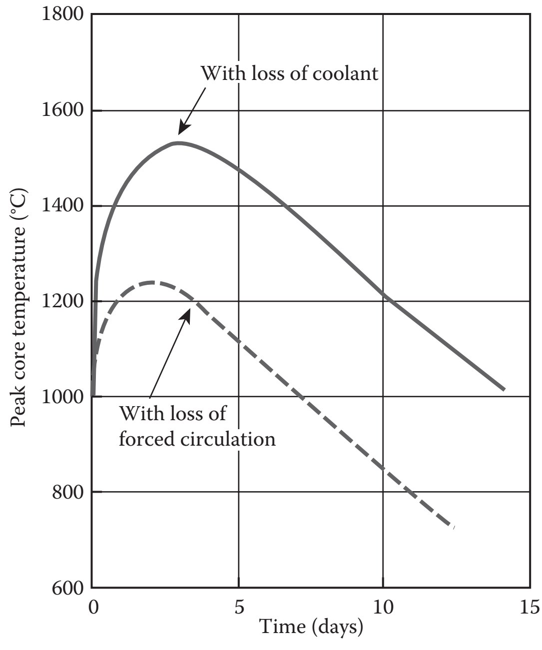

FIGURE 5.17
GT-MHR Core heat-up temperatures with passive after heat rejection.

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

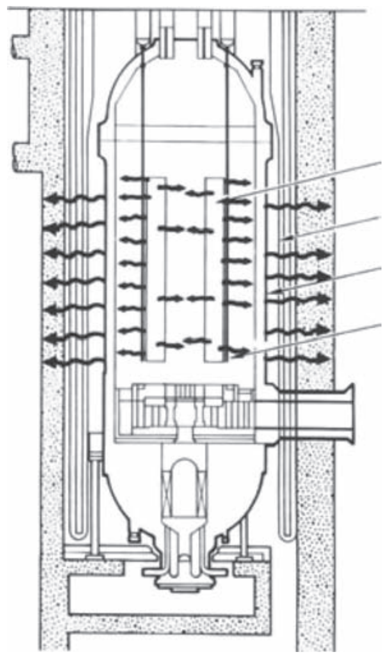

FIGURE 5.18
GT-MHR passive radiation and conduction of after heat to silo containment.

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

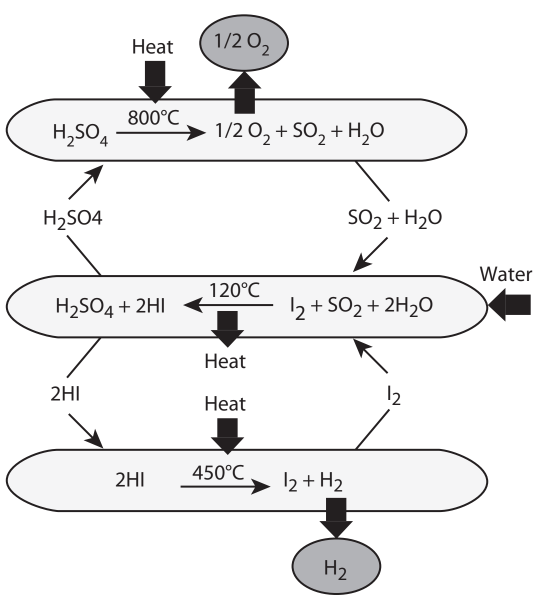

FIGURE 5.19
Sulfur-Iodine thermochemical water-splitting cycle.

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.