7. ENVIRONMENTAL CONSIDERATIONS

7.1. INTRODUCTION

Concerns over the environment have been recognized at recent United Nations conventions on environmental sustainability and biodiversity. These conventions emphasize that, as humans, plants and animals all depend on a healthy environment, protection of the environment is of paramount importance. All industries, including nuclear power plants, have the potential to impact the environment. This section reviews HWR activities directed towards eliminating or reducing that impact.

A nuclear power plant can impact the environment during all phases of its life-cycle, from construction, commissioning and operation, through to decommissioning. The focus in this section is on operation. Although all operating nuclear power plants release radioactivity, chemicals, metal corrosion products and waste heat into the environment, the emphasis here is on radioactivity because it is usually of primary concern. However, non-radiological releases are discussed briefly as well. In considering radioactivity, both occupational doses received within the nuclear power plant and public doses received outside the exclusion boundary are of interest. This is because these doses are closely related to environmental performance and protection.

This section has two main parts: the first is concerned with the current status of HWR design and operation (Section 7.2), and the second with future improvements (Section 7.3), emphasizing that HWR systems related to environmental performance are being continuously evaluated and improved.

7.1.1. Reactor environments

As with other types of reactor, HWRs are located in a variety of countries and environmental settings. A significant number are located in tropical and subtropical climates. In addition, many sites are in Asia where land is typically far less available than in areas such as North America. The external environment surrounding HWRs therefore varies considerably in terms of climate, vegetation, animal life, population density, cultural practices, size and background radiation. Table LVI summarizes some country specific information on HWRs in operation or under construction.

While the environments surrounding HWRs are diverse, the same basic environmental protection principles apply to all. The environmental performance of any nuclear power plant is closely related to how it is constructed, commissioned, operated and decommissioned. Advances continue to be made in all areas. Sound engineering and operating principles are key factors in ensuring good environmental performance of nuclear power plants.

TABLE LVI. ENVIRONMENTAL CHARACTERISTICS OF HWR SITES

Country	HWR sites (operating or under construction)[a]		Average dose/dose range to an individual from natural background radiation (mSv/a)
	Sea water	Fresh water	
Argentina		3	0.75–0.83[b]
Canada	1	21[c]	2.6
China	2		1.8–5.4[d]
India	4	10	2.3 (1.96–3.22)
Japan	1		0.3–1.2[d]
Republic of Korea	4		2.76
Pakistan	1		1.3–1.75
Romania		2	3.5[e]

[a] Data from the International Nuclear Safety Centre database.
[b] May not include dose from radon gas.
[c] Eight temporarily shut down.
[d] Range for country, not range for reactor sites.
[e] Country average.

7.1.2. Assessing environmental improvements

Although HWRs are designed to prevent releases of radioactivity into the environment, small amounts are released during normal operation with gaseous and liquid effluents. The releases may include traces of actinides or fission products formed in the fuel, or traces of activation products from process systems or structural materials used in the reactor. The releases represent an extremely small addition to the natural radioactive environment. Increases are often so small that they cannot be reliably measured, but they, and the resulting doses to humans and other biota, can be estimated using pathway analysis and related models.

Worker safety and protection have been dominant factors in the design and operation of HWRs. Internal occupational radiation exposure, mostly from tritium (which in HWRs is in the form of tritiated water, not tritium gas), is controlled by system leakage control, drying and ventilation of the reactor air. External occupational exposure is minimized through material control, access control and shielding. Where work needs to be performed in relatively high radiation fields, extensive automation has been introduced. Many of the systems and materials that tend to limit occupational doses also limit releases of radioactivity into the environment — hence, the close relationship between occupational doses and environmental performance.

In addition to radioactivity, traces of non-radioactive materials that may affect workers are discharged into the environment. These include traces of the chemicals added to the process waters in the reactor to condition them for specific applications, cleaning and housekeeping materials, water treatment chemicals and corrosion products. These materials are discharged into the environment, along with the liquid effluents, in very low concentrations.

The environmental performance of HWRs is closely linked to the need to conserve and recycle heavy water. The reactors are designed, built and operated to minimize leakage from heavy water circuits. Systems are also provided to collect and recover leaked water, and include desiccant dehumidifiers (dryers) to remove heavy water moisture from the air. Thus, the residence time of heavy water and tritium in HWR air is short, and most of the heavy water and tritium are recovered and therefore not released into the environment. This dual approach to tritium management has led to low occupational tritium doses and low environmental releases. It has also helped maintain low occupational doses from radioiodines.

Not all HWRs (see Table LVI) use the same systems and materials to enhance environmental performance. The primary differences relate to containment design and the type of annulus gas system. India and Pakistan are building and operating small HWR designs. In Canada, Ontario Power Generation is operating multiunit HWRs with relatively small containment volumes; many of these have a significant fraction of their reactor auxiliary systems located outside containment. The Republic of Korea, Argentina, Romania and Canada are operating 1980 vintage CANDU 6 reactors sold by AECL. The Republic of Korea has recently completed three improved CANDU 6 reactors, and two are under construction in China. A second improved CANDU 6 reactor is being constructed in Romania. The CANDU 6 design has many of the reactor auxiliaries located inside its larger containment volume. AECL is currently marketing a further upgrade, the CANDU 9 reactor, which builds on CANDU 6 improvements.

Of key importance to the environmental performance of nuclear power plants are waste management systems, particularly those for radioactive wastes. All these wastes are usually stored at the plant site. Storage is a temporary measure only. However, in many cases, plans and programmes are in place for the permanent disposal of wastes, which means their effective isolation from the environment and from humans for long time periods.

7.2. STATUS AND EVOLUTION: DESIGN AND OPERATION

7.2.1. Occupational dose

A historical record of average occupational doses from CANDU reactors compared with those from several other types of reactor is given in Fig. 211. As can

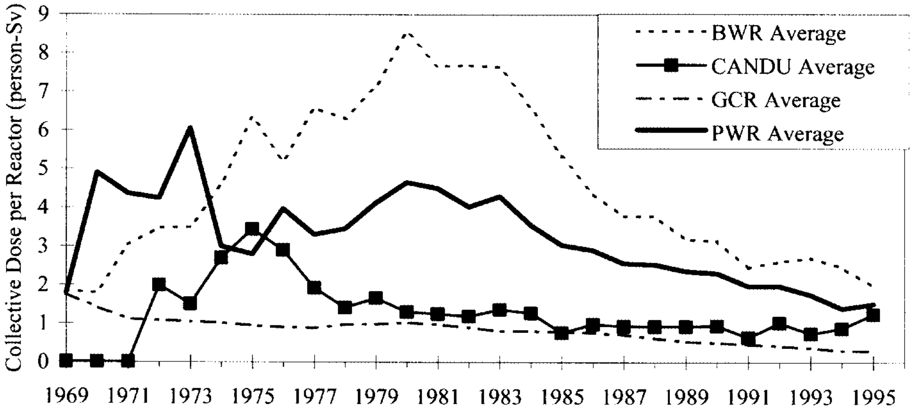


FIG. 211. Average occupational doses for CANDU reactors, boiling light water reactors, gas cooled reactors and pressurized light water reactors. Data taken from Nuclear Power Plant Occupational Exposure in OECD Countries 1969 to 1995, by CEPN for the NEA.

be seen, doses from CANDU reactors are lower or comparable to those from other types. Within an HWR, the concentration of tritium in the process fluids builds up during operation, reaching a plateau sometime after 60 years of operation. Despite this buildup and any probable increase in repair frequencies over time, occupational exposures have steadily declined.

7.2.1.1. Radiation control programmes

Typically, radiation control programmes are based on the International Commission on Radiological Protection (ICRP) philosophy of radiation protection, which draws on the following main principles:

- Each source of exposure should be justified in relation to its benefits or to those of any available alternative;
- Any necessary exposure should be kept as low as reasonably achievable;
- Dose equivalents received should not exceed specified limits;
- Allowance should be made for future development.

The principles employed by the reactor designers and operators are that all applicable dose limits be met, and that exposures be kept as low as reasonably achievable, taking into account economic and social factors.

Occupational doses are due to internal and external radiation exposures. Historically, the external dose has tended to be dominated by activated cobalt (e.g. ^{60}Co), but ^{95}Zr, ^{95}Nb and ^{124}Sb have also contributed, and in some stations, antimony has been the dominant source of external exposure. The internal dose has primarily been due to tritium. In Canada, over two thirds of the annual occupational exposures have occurred during maintenance and shutdown outages, and external radiation has been the major source of exposure during such outages. The experience is similar at Indian HWRs.

In general, there is an increase in radiation fields as a nuclear power plant ages. This is due to a combination of buildup of radionuclides on system surfaces, the need for more inspections and repairs, and (in the case of HWRs) the buildup of tritium in the heavy water. However, data indicate that it is possible to mitigate the potential effects of these field increases through the implementation of a broad range of design and operational changes. Some of the changes cited by HWR designers and operators include the following:

- External dose intensive operations include inspections close to the reactor face, such as fuel channel scraping, inspection of feeder cabinets, maintenance of the fuelling machine bridge, steam generator inspection and repair, and fuel channel replacement. Work planning, remote tools, automation, job specific shielding and self-protection measures have been used to control hazards. Chemical decontamination has also been used very successfully.
- Work undertaken around moderator system components can be associated with higher potential exposures because of the higher tritium content of the moderator water compared with HTS water. Protection equipment, D_2O recovery equipment, dryers and ventilation systems help in controlling the tritium hazard. In addition, improvements in instrumentation for hazard monitoring have been cited as being effective in the control of potential hazards.
- As HWRs age, some owners/operators have found that tritium control (and hence internal exposures) can be reduced by re-optimizing the ventilation system and vapour recovery system (VRS).

7.2.1.2. Design features and operating guidelines

While each HWR design has its own unique features, the following list summarizes the methods generally used for minimizing occupational exposures:

- Placement of shielding between the source and the worker.
- Arrangement of layout to facilitate good access to components and thereby reduce inspection and service times.
- Optimization of the coolant chemistry in order to minimize the deposition of corrosion products in the reactor core.
- Minimization of corrosion of metal surfaces.

- Removal of a portion of the radioactivity present in the water circuits by use of purification systems (removal of particles and dissolved species by filtration and ion exchange), and before the flow enters components in the neutron flux.
- Selective automation of inspection and maintenance work in high radiation areas.
- Use of components with low leakage rates to minimize release of heavy water containing tritium and other radionuclides.
- Use of comprehensive leak detection and repair programmes.
- Use of high efficiency air drying systems to maintain low dew point air in areas where tritium escape can take place in order to minimize tritium in air concentrations.
- Hot conditioning of HTS during commissioning to provide an oxide layer to protect carbon steel surfaces, and thus reduce transport of radioactivity.
- Decontamination of components, such as steam generators, when major repairs are anticipated.
- Full reactor decontamination prior to extended shutdown (this procedure has been used prior to reactor retubing).
- Segregation of moderator and primary heat transport heavy water storage and ventilation. The moderator system requires less maintenance, but the heavy water in the moderator has a much higher tritium content.
- Achievement of increased equipment reliability to minimize maintenance activities and the need for component replacement.
- Subdivision of the reactor into radiological zones on the basis of the level of contamination.[7]
- Reduction in the concentration, in reactor materials, of elements, such as cobalt and antimony, that produce a high yield of activation products in the reactor core.
- Use of fuel management procedures that result in very low incidence of fuel failures.
- Use of fuel failure detection systems to identify the location of a failed fuel element, so that after detection, it can be promptly removed by on-power defuelling.
- Training of reactor staff in radiation protection procedures and the provision of health physics staff to ensure that good radiation protection is practised.
- Development of detailed work plans in areas with high radiation fields, augmented by worker training including the use of mock-ups to practice tasks.

[7] Typically, at least three zones are discussed: zone 3 contains reactor components that are contaminated, zone 2 contains shops where contaminated components are handled, and zone 1 is free from contamination.

- Investigation of all incidents where actual (or potential) radiation exposures higher than normal have (or could have) taken place, followed by a review of practices to identify the need for corrective action.

Not all of the above initiatives have the same impact on dose reduction. The most important initiatives are outlined below in more detail.

(a) Material selection

A major source of radiation fields has been ^{60}Co, the activation product of ^{59}Co. Smelters, refiners and component manufacturers have been able to reduce the cobalt content of alloys during the past few decades. Important materials that require minimum cobalt content are the carbon steel feeders and headers, the boiler tubes and the pressure tubes. Other sources of cobalt are the hard facing alloys used for wear surfaces, of which the best performing ones are high cobalt alloys such as Stellite 6. The substitution of high nickel alloys for Stellite 6 has been partially successful, although Stellite 6 is retained in a few critical applications.

Antimony is commonly used as a filler in seals and gaskets in non-nuclear applications. Its activation product is ^{124}Sb, another high energy gamma emitter. Recognition of the potential significance of this material has led to its virtual elimination in seals and gaskets. In addition, extensive quality assurance procedures, particularly with carbon steel components, are used to ensure that all heat transport components are free of antimony.

To reduce corrosion of the carbon steel feeders, and thus reduce the production of ^{60}Co and increase the service life of the feeders, chromium enriched carbon steels have been specified for use in the newer CANDU reactors.

The causes of reduced service life of pressure tubes in early reactors have been removed through improved manufacture and assembly practices, and improved component design. This is expected to result in lower occupational dose and waste generation if the pressure tubes need little maintenance and minimal inspection until replacement.

(b) Activity transport

The heat transport chemistry parameters are optimized to minimize corrosion of the carbon steel piping. AECL has developed an activity transport model that simulates the corrosion of system components, the transport of particulate and dissolved corrosion products, and their redeposition in the heat transport circuit. The model indicates that under non-optimum conditions, corrosion products, containing ^{60}Co and other nuclides produced by neutron activation, are deposited on the fuel and the pressure tubes. These corrosion products may be released from the fuel as particulates or in solution, and together with the radionuclides are redeposited in the reactor circuit. This process is the major contributor to radiation fields occurring around HTS components. Maintaining the optimum pH of the HTS has been found to be important for minimizing corrosion of the carbon steel piping and for reducing the spread of radioactivity.

(c) Failed fuel management

A fuel element failure and its location can be identified using the gaseous fission product monitors and the delayed neutron monitors included with most HWRs. Gaseous fission product monitors detect the characteristic gamma rays of certain fission products that are released into the heat transport coolant from failed fuel. A delayed neutron monitoring system identifies the fuel channel that contains the failed fuel and can confirm that the failed fuel has been removed by a subsequent refuelling operation. Early detection and prompt removal of failed fuel limits deterioration of the failed fuel element and the quantity of fuel particles released into the HTS. In Indian HWRs, the use of gaseous fission product monitors has been discontinued, and delayed neutron monitoring adopted for all units.

(d) Automation

Inspections of pressure and boiler tubes have been automated, along with pressure tube maintenance, resulting in significant reductions in worker exposure during outages. In addition, the replacement procedures for pressure tubes have, whenever possible, been organized and automated to reduce the cost, outage time and occupational dose during replacement activities.

(e) Scaffolding

When components need to undergo repair or maintenance, scaffolding is often erected to provide access. The erection and disassembly of scaffolding can be time consuming and these activities have been identified as significant components of the worker dose in some HWRs. Whenever possible, designers and owners/operators have been replacing scaffolding with permanent or mobile platforms.

(f) Decontamination

Chemical and mechanical decontamination of radioactive components have been used in HWRs. Both full-reactor decontamination and the decontamination of specific components have been practised. To perform chemical decontamination, the reactor is shut down, the lithium hydroxide is removed by ion exchange, and organic acids and complexing agents are added to the circuit heavy water. The reagents dissolve the corrosion products. The reagents are regenerated by passing the solution through ion exchange resins, and at the end of the decontamination the reagents are removed. To date, ten full-reactor decontaminations have been completed in Canada using the CANDU decontamination (CANDECON) process. Decontamination factors averaged 4.1 at the reactor face, 3.4 at the boilers and 6.2 for the piping. The decontamination factor is defined as the "radiation field before decontamination divided by the radiation field following decontamination."

Partial decontaminations can be used when the repair and maintenance work is concentrated on a specific equipment type. Boilers, for example, can be isolated from the heat transport circuit and decontaminated separately. Both grit blasting and CANDECON chemical decontamination have been applied to decontaminate boilers. Decontamination is beneficial not only from the dose reduction perspective, but also because it improves heat transfer through boiler tubes and flow in the HTS circuit.

In Argentina, the high efficiency removal of oxides (HERON) decontamination process has been developed and applied to the main pumps of the HTS. One advantage of this process is that it operates at lower potentials than more widely applied electrochemical methods. The decontamination factors achieved with the HERON process have been greater than 10.

Gentilly 2 is the only Canadian HWR reporting significant antimony fields. Prior to maintenance outages, several antimony decontaminations were applied to this reactor. In the decontamination process, the reactor is shut down, the normally reducing water chemistry conditions are changed to oxidizing conditions, and the antimony released by dissolution is removed by ion exchange resins. Optimization of this decontamination procedure has resulted in more efficient antimony removal, shortened duration of the decontamination process, and less ion exchange resin being used.

(g) Controlling the spread of contamination

The following design features have been cited as ensuring that the spread of radioactive contamination inside an HWR is minimized:

- *Zoning*: Zoning is the classification of an area according to its level of potential contamination. Depending on the zone classification, different requirements for protective clothing are specified. The geographical layout of zones is kept simple, minimizing the number of zone interchanges and hence the amount of interzonal traffic. Physical barriers direct the movement of persons and material from the most contaminated to the clean zones. To assist in traffic control, a number of contamination monitoring locations are typically provided at the zone interfaces used by workers.

- *Rubber area*: The use of rubber shoes is an extension of the zoning system. The objective is to ensure that an area of high contamination does not contribute to the spread of contamination. This is done by 'roping off' a contaminated area and requiring the wearing of protective clothing within it. There is usually a rubber shoe change station and, as a minimum, everyone wears a white coat or overalls and rubber shoes. When a particular job has been completed, the area is decontaminated and the protective clothing removed.
- *Change rooms*: The type of protective clothing worn in each of the zones is regulated and facilities are provided for changing clothes and washing. Typically, change rooms contain storage for personal and work clothing, and radiation monitoring facilities.
- *Ventilation system and VRS*: The spread of radioactive contamination is further controlled by controlled ventilation systems and by the VRS, in addition to the controls described above.
- *Protective equipment*: Protective clothing is used to stop contamination spreading to clean areas and to protect the workers when working under contaminated conditions. Whenever airborne contamination reaches critical levels, workers protect themselves from inhalation hazards by wearing suitable respiratory protection. Various respirator designs are used to provide the level of protection required.
- *Decontamination*: When equipment such as pumps and valves become contaminated and are removed for repair, they are decontaminated by using chemical or physical cleaning processes. Special techniques are also used for the decontamination of skin and clothing.

(h) Radiation exposure control programmes

In some HWRs, the containment volume is accessible when the reactor is on-power. These HWRs usually have larger containments that house many of the auxiliary systems. However, within such containments not all the rooms or areas are classified as accessible. In HWRs with inaccessible containments, auxiliary systems are distributed in so-called 'confinement areas', which also have controlled access.

Detailed records are kept on radiation exposure associated with each nuclear power plant task. The data can then be studied to identify individual systems, components and tasks that cause significant radiation exposure. Design and operational changes can then be introduced to reduce exposures due to the largest contributors. In CANDU 6 HWRs, most of the occupational exposures occur during annual plant outages, when maintenance work is being undertaken on radioactive system components. In general, the deposited radioactivity in the HTS and tritium in the heavy water systems are the principal radiation hazards.

The following design features have been cited as the means of ensuring that equipment with potentially high radiation fields is shielded from the work areas:

- Normal or heavy concrete walls shield large components, such as heat exchangers, from accessible areas.
- Filters are surrounded by lead based shielding material.
- Whenever practicable, pipes that are radioactive are run through areas that are inaccessible during operation, shielded behind walls, or located inside protective trenches.
- Shielding of pumps is accomplished by separating the motor from the pump bowl by means of an internal barrier or a thick concrete floor.
- Valves are located in valve galleries or behind shielded walls that have holes for valve manipulation. Shielding from nearby components may be required.

7.2.1.3. Controlling tritium exposure

Any escape of heavy water from reactor systems or from auxiliary systems involves the release of tritium. Many HWR design features are aimed at reducing heavy water escape, and at removing heavy water vapour from the inside air in order to minimize worker exposure and tritium emissions.

(a) HTS

Mechanical joints are potential pathways for the escape of heavy water. In newer HWRs, however, most of the mechanical joints used in older designs have been eliminated, resulting in most of the components in the HTS being connected by an all-welded piping system. The newer HWRs do, however, retain some components with mechanical joints. These include the rolled joints between the pressure tubes and the end fittings and the end fitting closures. Some designs also retain mechanical graylocs between the end fittings and the feeders, but these are high integrity, metal to metal joints. There are no valves in the main HTS circuit of newer reactors but several of the auxiliary systems contain valves. Typically, there is a gland seal system on each HTS pump drive shaft and a leakage collection system on the stem of each major valve. Mechanical joints remain, however, potential pathways for the escape of heavy water.

In most HWRs, the fuelling machine vault receives leakage from the graylocs and/or end fitting closures. Some heavy water may also escape into the vault through joints in the fuelling machine and when the fuelling machine disengages from an end fitting. The air in the fuelling machine vault is circulated through a VRS, where desiccant dehumidifiers (dryers) adsorb water vapour from the atmosphere, to prevent its escape from the vault. The desiccant is regenerated at intervals to maintain drying capacity, and the regeneration condensate is routed to the D_2O management system. Various dryer technologies have been used. Operating experience shows that these dryers are extremely effective at recovering heavy water.

(b) Moderator system

The moderator system is a low pressure, low temperature system, and therefore the opportunities for heavy water escape are small. However, even in newer reactor designs, the main circuit has a number of valves. These valves and some components in the external circuits are connected using mechanical or flanged joints. To reduce leakage, newer reactors have live loaded, double packed stem valves on large valves, and bellows valve stem seals on small valves. Nevertheless, small, detectable amounts of heavy water may escape from the moderator systems.

The major components of the moderator system and its auxiliaries are contained in either a moderator enclosure or confinement rooms. The air from such areas is typically circulated through a dedicated VRS, similar to that used for the fuelling machine vaults. The opportunities for releases of tritium to occur from equipment located in these rooms is therefore small. However, in older HWRs, some tritium released in these areas passes to other areas. Newer HWR designs include additional improvements intended to further reduce this migration.

(c) Process systems' auxiliaries

In addition to the fuelling machine vaults, newer reactor designs isolate the process systems' auxiliaries and provide a VRS for them. The number of such areas and the fraction of the auxiliary equipment contained in them vary between designs. Most HWRs have some equipment located outside the VRS areas, and leaks from this equipment can lead to airborne tritium emissions and hazards. Emissions are closely scrutinized during design and operation, and, when warranted, areas are connected to a VRS.

(d) Ventilation and vapour recovery

The ventilation system and the VRS work together to maintain low tritium in air concentrations in all parts of an HWR and to recover heavy water that has escaped. In areas where the potential for escape is very limited, low tritium in air concentrations are maintained using the ventilation system. In areas where there is a higher potential for escape, the air is circulated through an efficient VRS. The areas serviced by each system and the tritium levels maintained by them vary between HWR designs. The trend has, however, been towards connecting more areas to the

VRS, to increase the capacity of this system, and to reduce tritium air concentrations and releases to the environment.

7.2.1.4. Design changes

A variety of design changes have been discussed and/or implemented in new HWRs to minimize occupational hazards and doses. As the primary vendor of HWRs, AECL has been one of the more active organizations, updating its CANDU 6 design and applying the experience gained to produce the improved CANDU 9 design. Relative to the initial four CANDU 6 reactors built in the early 1980s at Pickering, Ontario, several key improvements have been made to subsequent CANDU 6 reactors. The main ones cited are as follows:

- *Expansion of the moderator subsystem of the VRS:* In the case of the newest CANDU 6 plants in the Republic of Korea, the moderator purification area has been connected to the VRS serving the moderator equipment enclosure. The capacity of the VRS has been doubled to maintain low tritium levels in this area. Provisions have been made for improving ventilation air flow from the non-accessible areas to the moderator equipment enclosure, thereby reducing the spread of tritium from this enclosure to other areas. The CANDU 9 design is even further advanced because the purification system has been moved into containment. Both of these changes are consistent with the improvements adopted by Ontario Power Generation in its newest HWRs.
- *Integration of the reactor building ventilation system (RBVS) with the VRS:* To reduce tritium emissions and releases to the environment, the RBVS and the VRS have been integrated in the designs of proposed CANDU 6 and CANDU 9 HWRs. All the air leaving the reactor building (containment) is treated by the VRS before discharge. Provisions have also been made for improving ventilation air flow from the accessible areas to the non-accessible areas, ensuring that the tritium does not spread to the accessible areas.
- *Delayed neutron sensing tube vibration:* In the case of the newest CANDU 6 plants in the Republic of Korea, changes in the layout of the delayed neutron sensing tubes under the feeder cabinet were made to eliminate fretting and allow for ease of visual inspection of the tubes.
- *Improvements to the D_2O and to the tritium management system:* In the most recent CANDU 6 plants in the Republic of Korea, a dryer has been installed at the air intake filter to service the reactor building. This reduces D_2O vapour downgrading and thereby improves the efficiency of the VRS. This change was pioneered by the New Brunswick Power Corporation as a retrofit in its Point Lepreau CANDU 6 plant. It has been proposed by KEPCO for the Wolsong 1 CANDU 6 plant, and has been included in the CANDU 9 design. The

CANDU 9 design has, however, taken the concept further, by eliminating the dousing tank and thereby further reducing downgrading of collected D_2O vapour. Both of these changes reduce the volume of water present for upgrading and thereby the amount of tritium discharge to the environment with liquid effluents.

- *Hard facing alloys:* Where possible, the hard surfacing cobalt alloy Stellite 6 used in older designs has been replaced in new HWRs. As a result, a reduction in external occupational exposures resulting from the deposited radioactivity in the HTS components is expected.
- *Outlet feeder material:* In new HWRs, the outlet feeders have been changed to include a low concentration of chromium in the carbon steel. The chromium reduces the corrosion rate of the steel and thereby reduces the accumulation of corrosion products and the associated radiation fields at shutdown. As a result, occupational doses to workers carrying out maintenance on the HTS will be reduced.
- *Welded feeders:* In the CANDU 9 design, the feeders are welded to the end fittings rather than being connected by graylocs. This will eliminate leakage and escape of D_2O into the fuelling machine vault.
- *Improved reactor layout:* The layout of the CANDU 9 reactor components has been improved, thereby reducing the need for workers to access areas with elevated radiation fields.

In addition to the above measures, tritium can be extracted from heavy water using a number of technologies, which produce a tritium reduced D_2O product and a tritium enriched hydrogen stream. While the correlation between tritium concentrations and either tritium emissions or occupational doses is not clear, tritium extraction (detritiation) does offer the capability to cap tritium concentrations at levels below the equilibrium concentrations. It can also offer heavy water management advantages, as it simplifies the movement of heavy water between reactors or systems having different tritium concentrations. Detritiation may also simplify the decommissioning of HWRs because it may improve the economic value of the D_2O assets. To date, only Ontario Power Generation, the owner/operator of 20 CANDU reactors, has implemented large scale detritiation (Table LVI). Developments in this area have focused on reducing the costs of detritiation technology and on establishing the optimal time for introducing detritiation into a reactor life cycle. It is expected that future HWRs will continue to be designed to operate for their entire design lifetimes without implementing detritiation. For both existing and future HWRs, there will, however, be the option of considering detritiation relatively early in their life as part of their heavy water and tritium management programmes.

7.2.2. Radioactive emissions

Radioactive emissions leave nuclear power plants as gaseous, liquid or solid wastes. The principal gaseous radioactive wastes — tritium and ^{14}C — are associated with heavy water. Tritium is primarily formed in the moderator and, to a lesser degree, in the HTS, and is released as tritiated heavy water vapour, i.e. the molecules of heavy water vapour include some DTO molecules. Carbon-14 is principally formed in the moderator system and is released as carbon dioxide. Three other gaseous radioactive wastes are also released: noble gases, radioiodines and particulates. Waterborne emissions from HWRs are typically categorized into tritium (as DTO), ^{14}C and gross β–γ radiations. All radiological emissions are monitored before release from an HWR into the environment. Releases contribute to radionuclide concentrations in the environment, and to doses to plants, animals and humans. The doses involved are typically very low. In addition to the amount of radioactivity released, they depend on the specific environmental features of each site. Figure 212 summarizes average maximum public doses from Canadian HWRs over time.[8] As can be seen, these doses are a small percentage of the natural background dose, and, furthermore, have been declining.

7.2.2.1. Tritium release

The measures outlined in Sections 7.2.1.3 and 7.2.1.4 on tritium dose control for workers also result in parallel reductions in heavy water losses and tritium emissions into the environment.

7.2.2.2. Carbon-14 emissions

Over 95% of the ^{14}C generated by the CANDU reactor is produced in the moderator. The remainder is generated in the HTS, with minor quantities generated by the annulus gas system. The ^{14}C generated in the moderator system is present as carbonate in the moderator water. Once generated, it is transferred either to the ion exchange columns in the moderator purification system or to the moderator cover gas. Normally, over 95% of the ^{14}C is removed by the ion exchange columns, the remainder being emitted through moderator cover gas venting and leakage. Low ^{14}C emissions are maintained by ensuring that the ion exchange columns are not saturated, and thus can continuously and effectively remove ^{14}C from the moderator

[8] There is no central registry of public doses covering all PHWRs. The Canadian Government, however, publishes summaries of emissions and emission limits for reactors in Canada, from which dose estimates can be made.

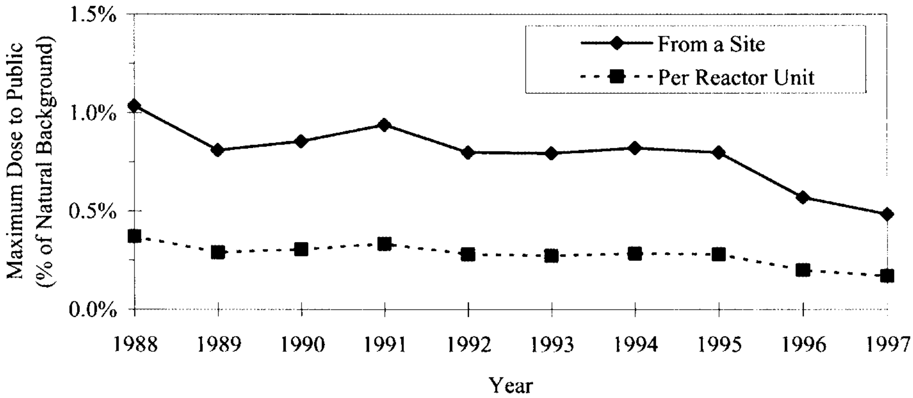


FIG. 212. Maximum average dose to a member of the public from a CANDU reactor in Canada. The figures are calculated using data obtained from Radioactive Emissions Data from Canadian Nuclear Generating Stations 1997 to 1998, Report INFO-210/Revision 8, AECB, Ottawa.

water. Owners/operators have pursued reductions in ^{14}C emissions by improving their control of moderator chemistry.

In early HWR designs, nitrogen or air was circulated through the reactor annuli (between the pressure tube and the calandria tube). Carbon-14 is the neutron activation product of ^{14}N, and so was a major source of ^{14}C production in the annuli. Newer reactors use CO_2 as the annulus gas, and some of the older reactors have been converted to use this gas. Therefore, ^{14}C is produced in trace quantities only and releases into the environment are correspondingly reduced.

7.2.2.3. Noble gas emissions

The noble gases released from HWRs are ^{41}Ar and various fission products. The fission products are mainly noble gas radionuclides from the natural uranium fuel, whereas the ^{41}Ar is derived from activation of the ^{40}Ar present in air and other gases.

Fission products collect in the free space between the pellets of uranium dioxide fuel and the Zircaloy 4 fuel sheath. Normally, these fission gases are contained by the fuel cladding. However, defects do occasionally occur in the cladding and the fission gases are then released. These gases are released into the HTS and the spent fuel handling system. Several HWRs have an off-gas management system to collect fission produced noble gases and thereby reduce emissions. The off-gas management system contains an activated carbon bed that adsorbs the noble gases and delays their emission into the environment until these gases have decayed to insignificant levels.

Argon-41 emissions are controlled by excluding air from the reactor core. In operating reactors this is achieved through careful maintenance of reactor components, the venting of gaseous systems, such as the moderator cover gas and annulus gas systems, and, if air ingress is detected, employment of quality control on the source of the gas supply. In newer HWRs, the trend has been to blanket most heavy water storage tanks with helium to minimize the dissolution of ^{40}Ar into heavy water. In newer Indian HWRs, argon is eliminated from the reactor core area by using a water filled (instead of an air filled) calandria vault, by the absence of an air-cooled thermal shield, and by replacing air in the annular space between the pressure tube and calandria tube with carbon dioxide. These features are also used in all newer reactors of Canadian origin.

7.2.2.4. Other radionuclides

In addition to tritium, ^{14}C and noble gases, HWRs report releases of radioiodines, airborne particulates and waterborne gross β–γ radiations into the environment. The production and release pathways of these materials are similar to those from other reactor types. Typically, HWRs release very low levels of these materials, often less than other reactor types. As with other reactor types, air filters and liquid decontamination systems are employed to control potential and actual releases.

7.2.3. Non-radiological emissions

7.2.3.1. Sources of water and water conditioning

The water required to support the operation of an HWR is drawn either from a lake, a river or the sea, depending on the location and available supply. The intake water is distributed to different HWR systems with or without treatment. Typically, the majority (>90%) of the intake water after screening is used as once through cooling water for the condensers. A relatively small amount of the intake water (about 9%), after screening, is used as service water. A small portion of the intake water (about 1%) is subjected to extensive physical and chemical treatment and is used as boiler feedwater. Depending on the location of a given HWR and the details of its design, the proportion and the extent of water treatment varies. HWRs located on the coast use sea water for once through cooling purposes, whereas the rest of the water needed is taken from freshwater sources.

7.2.3.2. Effluent water sources

The typical major systems and the types of water effluent produced by HWRs are summarized as follows:

• Water treatment plant:	Filter backwash Clarifier effluent Rinse water Ion exchange regeneration solution.
• Steam generator (boiler blowdown):	Cleaning waste solution (inactive and active, depending on cleaning method employed).
• Turbine building and auxiliaries:	Oily waste solution Drain and sump effluents.
• Reactor auxiliaries:	Laundry waste solution Shower and sanitary drains effluents Active solutions from chemistry laboratories.
• Condenser:	Once through coolant water.
• Service water system:	Service water effluents.
• Site:	Storm water runoff, construction and commissioning waste solutions.

The dominant sources of chemicals in the effluent water of existing HWRs are the water treatment plant and boiler blowdown:

- *Water treatment plant effluents:* Typically, water treatment plants consist of a clarifier followed by filters and ion exchange columns. In the clarifier, polyelectrolytes are often used to assist the settling of suspended organic and mineral particles. They are discharged through the clarifier drains. Sulphuric acid and sodium hydroxide are used to regenerate the ion exchange resins, and they result in increases in the sulphate and sodium contents of the effluent released into the environment.
- *Boiler effluents:* The content and volume of the boiler blowdown depends on the type of chemistry control regime being applied. In Canada, HWR boiler blowdown typically contains morpholine which is used as a boiler conditioning chemical, some residual hydrazine which is used as an oxygen scavenger, and ammonia which is used for pH adjustment. The dissolved species and suspended solids in this stream released into the environment are in the fraction of parts per million range.

- *Active liquid waste stream:* This stream is collected from locations where the water may be contaminated. The collected liquid includes waste solutions resulting from: laundry operation, chemical cleaning, laboratory operation, floor sump cleaning, active oil–water separation and deuteration/dedeuteration of ion exchange resins. The chemical constituents of the active liquid waste stream include soaps, detergents, laboratory reagents, suspended solids and traces of oil. Typically, a portion of this effluent is treated by filtration and ion exchange to remove or reduce radioactivity. This also results in some reduction in the chemical content of the effluent stream released into the environment.

More concentrated chemical solutions are generated when non-radioactive components are subjected to chemical cleaning. Examples are boiler and heat exchanger cleaning solutions.

All of these effluent streams are discharged into the environment along with the main cooling water effluent. In Canada, Ontario Power Generation HWR effluents have been subjected to standard environmental toxicity tests under Ontario's Municipal Industrial Strategic Abatement Program. The diluted effluents were not found to be toxic to a species of small crustacean or to rainbow trout, and therefore they have been assessed as not being harmful to the environment.

7.2.4. Waste storage

7.2.4.1. Process waste

HWRs tend to produce lower volumes of solid waste than other reactor types. Generally, most HWRs are equipped with facilities for the interim storage of solid wastes. The equipment and facilities are flexible enough to cope with the anticipated increase in waste volumes and activities during periods of major maintenance or adverse reactor operation. The exact details of these systems, including their capacities, are specific to the nuclear power plant. The origins of the solid radioactive wastes can be classified into three main groups: fuel fission products, system material activation products and system fluid activation products.

Although the majority of the radionuclides remain at their place of origin, some ultimately reach one or more parts of the active waste management system. For example, fuel fission products are contained within the fuel sheath and only those fission products in defected fuel elements can escape. The majority of the fission products that do escape from fuel defects while in the core or in the fuel handling equipment, are filtered, trapped or removed in the HTS and its auxiliary systems. This leads to a requirement for disposal of the fission products collected, either in spent resins or in filter elements, as solid wastes. Radionuclides that escape by leakage, or otherwise, from the HTS boundary, reach the building atmosphere. Most of these are collected by the active ventilation system.

A typical solid radioactive waste management system includes the facilities to handle the following types of waste:

- Spent fuel (high level waste, or spent fuel waste, should the decision be taken not to reprocess it (see Section 6)).
- Spent resins (intermediate level waste).
- Spent filter cartridges (intermediate level waste).
- Low activity solid wastes (low level waste), consisting of:
  — Non-processible/non-combustible waste (metal, glass);
  — Processible/combustible waste (paper, rags).
- Organic fluids, oils and chemicals.

Radioactive solid wastes are produced on a continuous basis. The wastes (other than spent fuel) can be assigned to one of the following five categories:

- Spent resins (from both light and heavy water radioactive circuits).
- Spent filter cartridges (from both light and heavy water radioactive circuits).
- Low activity solid wastes; these may be classified as combustible and non-combustible, or as compactible and non-compactible.
- Organic fluids, oils and chemicals.
- Small volumes of liquids with too high a radioactivity to allow discharge (non-aqueous contaminated liquid wastes).

All of these materials are collected in the nuclear power plant and prepared for storage. Each type of waste is handled and stored differently, as described below. The final disposal methods used depend on the particular site, the regulatory requirements, and also on the policies of the plant owner/operator.

(a) Spent resins

Spent resins originate from various systems, including the HTS and the moderator purification and cleanup system, the liquid waste management system and the decontamination facility. The total spent resin volume derived from a CANDU 6 reactor averages around 7 m^3/a. Resin is usually stored temporarily in storage vaults (tanks) inside the nuclear power plant. Spent ion exchange resin in the form of slurry is discharged into these vaults. Water from these tanks is transferred to the radioactive liquid waste management system. The residence time of resin in these tanks varies between nuclear power plants.

(b) Spent filter cartridges

Spent filter cartridges originate as a result of heat transport purification, moderator purification, spent fuel bay cooling and purification, liquid waste management and active drainage, as well as from the heat transport pump gland seal, fuelling machine and heavy water supply. The radioactivity on these filters is caused mainly by active particles collected in filter elements. The filter cartridge waste produced averages around 2 m^3/a for a CANDU 6 reactor.

(c) Non-processible and processible wastes

Non-processible/non-combustible and processible/combustible wastes originate from normal day-to-day nuclear power plant operation. They consist of cleaning materials, protective clothing, contaminated metal parts and miscellaneous items. Waste originating from certain radiological areas is often automatically considered radioactive even though it may contain no radioactivity. Typically, about 90% of the waste from a CANDU 6 reactor has contact radiation fields that correspond to dose rates of less than 5 μSv/h. These wastes may also include a small volume of solidified liquid waste. Several nuclear power plants have instituted waste sorting, monitoring and diversion programmes to segregate inactive from active wastes at source. The inactive waste bags are monitored for contamination, and if found clean, are shipped to a local landfill site. This programme has resulted in a significant reduction in the volume of radioactive waste requiring storage and disposal.

Typically, non-processible and processible wastes are first deposited separately in plastic bags at locations on the boundary between the active and inactive zones. These bags are subsequently sealed and checked externally for their level of radioactivity with β–γ monitors. The contamination criteria employed for treating the waste as inactive are based on local regulatory requirements. Inactive wastes can be disposed of cheaply as ordinary non-active solid waste.

Solid radioactive wastes may be compacted (if the waste is relatively soft) or compressed (if the waste contains hard materials). About 80% of the maintenance wastes generated at a CANDU 6 station are usually compactable. A typical volume reduction ratio for compaction is 5:1, with a waste density of about 100 kg/m^3 before compaction. Another volume reduction technology practised at Ontario Power Generation is incineration. Following such volume reduction processes, the wastes are usually put into standard 200 L (0.2 m^3) drums. On average, a full drum weighs about 100 kg. The drums are generally stored in solid radioactive waste storage facilities. Typically, drums from CANDU 6 nuclear power plants contain maintenance wastes that correspond to a radiation dose of less than 0.002 Sv/h on contact.

(d) Organic fluids, oils and chemicals

These wastes arise from decontamination areas, and include lubricating oils from pumps and organic solvents from various sources. Since their volumes are relatively small, it is current practice at most Canadian nuclear power plants to store them in 200 L drums at the plant to await future treatment and disposal. The treatment for such radioactive wastes may include special packaging, neutralization and/or solidification, and on-site or commercial incineration. If the oils can be treated to yield a product with a very low contaminant level, the product may be suitable for reuse as fuel for boilers. The Point Lepreau nuclear power plant incinerates oils with very low contamination levels in an on-site boiler.

(e) The solid radioactive waste storage facility

The solid radioactive waste storage facility is generally located at the nuclear power plant site, although there are central facilities that serve several nuclear power plants. The purpose of the facility is to store the waste in a readily retrievable fashion for eventual permanent disposal, to isolate stored waste from the environment, and to minimize radiation exposure to workers, the public and other biota. The capacity and design features of the facility are site specific. However, the facility typically consists of an area located within the exclusion boundary of the nuclear power plant where drainage characteristics are good, and where the lowest points of the concrete storage cells are above the highest anticipated level of the water table. The facility is designed to store both low and intermediate level wastes, while at the same time allowing for retrieval for future waste disposal (Section 7.3.4). The types of waste stored in the solid radioactive waste storage facility include spent filter cartridges (low and intermediate level wastes), and other compacted/compressed and packaged radioactive wastes, such as contaminated tools, piping and reactor components. The average radioactive waste output for a CANDU 6 reactor is presented in Table LVII.

7.2.4.2. Spent fuel

The spent fuel bays are designed to accommodate the spent fuel discharged from the reactor, including emergency discharge of fuel from the reactor core. Once the CANDU spent fuel has spent at least six years cooling in the bay, it is feasible for it to be transferred to an interim dry storage facility. Passively cooled dry storage structures have been constructed at the Gentilly 2, Point Lepreau, Wolsong and Ontario Power Generation sites and are planned for other HWR sites.

TABLE LVII. AVERAGE SOLID RADIOACTIVE WASTE OUTPUT FOR A CANDU 6 UNIT

Waste category	Volume[a] (m^3/a)	Activity[a] (TBq/a)
Spent fuel	110 bundles/week	
Spent resin[b]	7	15.5
Low level compactible wastes[c]	22	0.015
Low level non–compactible wastes[c]	13	0.020
Disposable filters	2	1.85
Other wastes	1.5	<0.3

[a] These quantities are based on 12 years' operational data (1983–1994) from four CANDU 6 HWRs (Point Lepreau, Gentilly 2, Wolsong 1 and Embalse). Owing to a variety differences, both the volume and the radioactivity of the wastes vary between HWRs.

[b] This quantity applies at the point of discharge to the in-station spent resin tanks.

[c] These volumes depend on the waste segregation procedures used by each station, and on the compaction and/or processing technologies employed by each station, prior to waste storage.

7.2.5. Land use

Indian HWRs have a 1.6 km radius exclusion zone, a 5 km radius sterilization zone and a 16 km radius emergency planning zone. In Canada, reactors have been built with reasonably large exclusion boundaries because land prices are relatively low. In the case of jurisdictions having more limited available space, however, it is advantageous to have smaller sites. Site size is primarily a function of the exclusion boundary, but it is also a function of the space required for each reactor. New CANDU designs can be constructed which occupy less land. For example, the CANDU 9 design can accommodate an exclusion boundary with a radius of less than 500 m, and each reactor occupies relatively less space than do the older designs. These represent considerable reductions in land use.

7.3. FUTURE DIRECTIONS AND IMPROVEMENTS

Currently, AECL dominates the development of HWR technology for future nuclear power plants and many owners/operators have extensive programmes pursuing improvement of existing plants. Thus, considerable R&D and engineering efforts are focused on developing and introducing improvements to existing and new HWRs. This is resulting in the continuous improvement of environmental performance, because improvements tend to reduce releases of radioactivity, chemicals and metal corrosion products.

7.3.1. Occupational dose

7.3.1.1. External dose reduction

As a result of improvements in design, monitoring of the work environment, automation of repair and inspection, and decontamination techniques, the occupational dose of the newer HWRs has been reduced. Various improvements have been implemented or are being evaluated. The following improvements are in the development and design stages:

- Zinc addition to the HTS is being assessed. Zinc can reduce the buildup of ^{60}Co on out of core surfaces, leading to lower occupational doses. An additional benefit is reduced corrosion rates for many of the component materials in the HTS. This will assist in minimizing deposit loadings in the steam generators and on the inlet feeder pipes, thereby maintaining thermal efficiency. Thinner oxide deposits are also expected as a result of zinc addition, which will help minimize the waste produced during full or partial decontamination.
- Cobalt based hard faced alloys are still being used in reactors in a few critical applications, such as in valves and ball bearings. Alternative materials are being evaluated as potential replacements.
- Historically, CANDU reactors have controlled the pHa of the HTS between 10.2 and 10.8. Improvements have been made to the chemistry control system to facilitate narrower pHa control. This results in a low, flow assisted corrosion rate for the carbon steel piping and, consequently, reduced production of activated corrosion products in the reactor core.
- The original corrosion inhibitor used in decontamination solutions was a sulphur based compound. Residual sulphur has been implicated in the localized corrosion of steam generator tubes having a high nickel content, e.g. alloy 600. The sulphur based inhibitor has been replaced with a mixture of sulphur free inhibitors that together are at least as effective as the original inhibitor. The decontamination process is being optimized to reduce corrosion of carbon steel surfaces and to improve the decontamination of stainless steel surfaces such as end fittings. This reduces radiation fields during replacement or maintenance operations.

7.3.1.2. Internal dose reduction

Improvements in tritium dose control for workers are expected to be implemented at existing and new nuclear power plants. These include improvements to protective equipment and instrumentation. In the case of new plants, further refinements of the plant layout are expected. For example, the further isolation of some moderator equipment is being evaluated as a retrofit to existing reactors. Detritiation, discussed in Section 7.2.1.4, is also a possibility for future nuclear power plants. Tritium dose control will not only reduce doses but will also reduce emissions into the environment.

7.3.2. Radiological emissions

7.3.2.1. Tritium emissions

The improvements in tritium management outlined in Section 7.2.1 will also reduce doses and emissions into the environment.

7.3.2.2. Carbon-14 emissions

While improved moderator ion exchange resin management can ensure significant reductions in ^{14}C emissions into the environment, opportunities for making further reductions are being considered. Ontario Power Generation briefly used a ^{14}C scrubber on the moderator cover gas of a small demonstration plant, but has not pursued this initiative further. This technology is also being investigated by AECL. The facilities used to store spent ion exchange resins constitute secondary, potential sources of emissions. There is no clear evidence that these facilities are important sources of radionuclide emissions, but several approaches are available that can achieve substantial reductions. Methods for reducing air ingress into reactor systems are also being studied because ingress necessitates the venting of gases from the reactor.

7.3.2.3. Noble gas emissions

Noble gas emissions tend to be so low that they are usually below the detection limit of even the most sensitive gamma spectrometers used for compliance monitoring. Thus, emissions are often reported to be at the detection limit, which is an overestimation of the actual emissions. In some HWRs, monitors have been upgraded and instruments with lower detection limits used. Data from them clearly show that noble gas emissions are indeed very low. It is expected that improved monitors will become a standard feature at those HWRs where there is concern about their reported release.

Methods for reducing the emission of fission gases into the environment are being investigated. Ultimately, the best way of reducing them is to avoid fuel failures. Thus, fuel quality assurance becomes of paramount importance. HWRs may also have a further modified off-gas management system to work towards the same goal.

Releases of ^{41}Ar are related to the ingress of air into the reactor because air contains ^{40}Ar. Methods for reducing air infiltration are being investigated and include the recycling of process gases and the use of storage tank bladders.

7.3.3. Non-radiological emissions and wastes

Ontario Power Generation has undertaken intensive monitoring of its stations under the MISA programme (Section 7.2.3.2). Over 450 000 analyses have been performed for 153 chemical, physical and biological parameters on samples taken from all of its nuclear power plants. Analyses identified contaminants that were found in significant concentrations, in relation to background concentrations and detection limits, although 'significant' contaminants do not necessarily cause a detrimental impact on the environment. The average number of significant contaminants per process effluent was 5, or about 3% of the 153 parameters evaluated.

While only small quantities of chemicals are discharged in the effluent water from nuclear power plants, there is potential for reducing both the volume of water discharged and its chemical content. To maintain healthy fish populations, it is particularly important to reduce the concentration of nitrogen compounds in the effluent water. These toxic compounds include hydrazine, morpholine and ammonia, added for eH and pH control of reactor circuits, such as the boiler and the coolant water circulation systems.

Zebra mussels, which were inadvertently introduced into the Great Lakes, have invaded the water intake pipes of several nuclear power plants in Ontario. The same problem has arisen at the Cernavoda nuclear power plant in Romania. The most common way to control such infestations is to dose the intake pipes with chlorine. However, the use of chlorine is environmentally undesirable and thus several other, more acceptable, approaches are being tested to replace chlorine.

There is also a potential for recycling some of the effluent waters. Many of the discharged streams have a lower chemical content than the feedwaters because they have been through a deionization or other water conditioning process prior to use (Section 7.2.3.1). Thus, the quantity of chemicals required to treat recycled water would be less than that used for fresh water. All that is required is a conditioning process to enable the feedwater specifications to be met. A careful evaluation is required, because if some of the impurities are not removed by the conditioning process, they could accumulate in the recirculation system and have the potential to cause other problems.

Several initiatives are focused on reducing the volume of radioactive waste generated. The diversion to producing landfill waste that is free from radioactivity is being adopted by an increasing number of nuclear power plants (Section 7.2.4.1). This can have a very significant impact on the volume of radioactive waste generated. The optimization of the ion exchange processes has the potential benefit of significantly reducing the volume of ion exchange resin generated. Conditioning technologies for ion exchange resins, such as cementation, are being assessed.

7.3.4. Waste disposal

Significant progress in the transition from waste storage to waste disposal has been made in Canada. A licensing application by AECL for a near surface disposal facility for low level waste is under review by the nuclear regulator. Several conceptual studies have been completed for the disposal of Ontario Power Generation's low level waste in near surface and underground vaults. A co-operative initiative between the Canadian Government and the nuclear electricity utilities for the development of a national low level waste and intermediate level waste disposal facility is at a formative stage. The target date for operation of the disposal facility is 2015.

An R&D programme was funded and managed by AECL and the Canadian CANDU utilities to evaluate the disposal of used nuclear fuel in the crystalline rock of the Canadian Shield. The viability of the disposal concept developed was evaluated by an independent panel which conducted public hearings to address both the technical and social merits of the concept. In this, the panel was assisted by an independent scientific review group. The panel concluded that the proposed concept is technically sound, but voiced concerns regarding its social acceptability. Responsibility for implementing the Government's acceptance and interpretations of the panel's recommendations now rests with the waste owners. Progress is being made by a programme carried out under the direction of Ontario Power Generation.

7.3.5. Other environmental considerations

7.3.5.1. Pathway analysis

Pathway analysis is frequently used to identify and quantify radionuclide migration in process equipment, the work place and the environment. A broad range of information is being developed to improve understanding of the dominant processes, pathways and parameters. This includes measurements, correlation of emission and dose records with events and conditions in the nuclear power plants, laboratory and field experiments, and modelling of radionuclide generation and migration. Information exchange through such international programmes as the Biosphere Model Validation Study is important in this regard.

The environmental impact of the radioactivity released from HWRs is normally far too low to be detected by observation. Thus, reliance is placed on pathway analysis to estimate environmental concentrations and doses. The models and parameter databases used for this purpose are being continually improved, as is model validation to provide more reliable estimates. New approaches for assessing radiological and other effects on the environment are being developed and applied. It is expected that the relative environmental impact of HWRs will continue to decrease as these tools and models evolve.

7.3.5.2. Water usage

Use of the discharged heat from the HWR to produce fresh water from sea water is a possible future use of reactor waste heat. There may also be other applications for waste heat.

7.3.5.3. Reduction in thermal effluents

Future CANDU designs may use moderator heat for feedwater heating. This would increase efficiency (equating to about a 1% increase in electrical output), and lower temperatures in the effluent. Lowering of this temperature could prove to be important because many aquatic organisms are sensitive to temperature changes and high temperatures.

7.3.5.4. Environmental monitoring systems for non-human biota

The Canadian nuclear regulator has introduced a new, expanded environmental protection programme, which includes addressing both radioactive and chemical contaminants in a comprehensive ecological context. Nuclear power plant owners/operators will be required to demonstrate, through performance assessment, monitoring, or other evidence, that their provisions for protecting the biophysical environment are adequate. The expanded environmental protection programme will be implemented through eleven sets of activities, which include: development of regulations, standards, guides and procedures; performance of assessments and analysis; consultation and co-operation with other Government agencies; compliance with external legislation and policies; participation of stakeholders; training of regulatory staff; review and evaluation of the programme; and efficient management of the programme. It is expected that this will increase the incentives for Canadian nuclear power plants to minimize emissions and thereby reduce potential impacts on the environment.

7.4. CONCLUSIONS

All operating nuclear power plants release small quantities of radioactivity, chemicals and metal corrosion products with liquid and/or gaseous effluents.

Furthermore, waste heat and both radioactive and non-radioactive solid wastes are produced.

Existing HWR systems, materials and operating procedures are designed to enhance reactor efficiency, and to limit radiological doses to workers and to the public. This, in turn, results in low radionuclide concentrations in the environment and good environmental performance. Many changes have been, or are being, implemented to improve reactors and to further reduce doses and discharges into the environment. Some of the existing HWRs have been retrofitted to include these changes, and they have been included in the new CANDU 6 and 9 designs. Many of these improvements not only result in lower doses and reduced discharges of radioactivity, but also in reduced discharges of chemical and metal corrosion products. Improved reactor efficiency, dose reduction and environmental performance go ‘hand in hand’, and they are continuous processes with many new initiatives in the planning stage.

HWRs have sophisticated waste management systems, which involve waste reduction, classification and sorting. Generation of radioactive wastes is minimized and wastes are stored safely, according to the level of radioactivity, usually at nuclear power plant sites. Allowances for the eventual disposal of these wastes are being made, and disposal facilities are under development or being planned. The safe disposal of radioactive wastes is of key importance in environmental protection. Non-radioactive wastes from nuclear power plants are being disposed of in landfills and other approved facilities.

In Canada, the nuclear regulator is increasing the emphasis placed on protection of the overall environment. The programme involves a comprehensive ecological approach and includes both radionuclide and chemical releases from nuclear power plants. Ecological risk assessment tools are being developed and applied by AECL in response to this programme in order to evaluate the environmental performance of both existing and proposed nuclear facilities. This effort will continue to enhance the environmental performance of HWRs.

It is clear that through design, selection of materials and adoption of procedures, the environmental effects due to nuclear power plants can be mitigated to very low levels. It is also clear that HWRs represent an environmentally sound method of electricity generation.

8. VISION OF ADVANCED HWR DESIGNS

8.1. INTRODUCTION

HWRs, and the nuclear industry as a whole, share the same basic challenge for the next millennium, i.e. to remain a reliable, cost effective and accepted source of energy. Global increases in energy supply are needed to provide increasing populations with an improving standard of living. HWRs can make a major contribution to meeting the increasing global needs by providing environmentally benign, sustainable and economic energy sources. The future energy market could obtain energy from a variety of sources, ranging from different nuclear generation schemes to fossil fuel burning to the application of emerging technologies, such as H_2 fuel cells. Each of these will have its place in both the electricity and transportation sectors. Thus, the challenge for HWRs will not only be to retain, but to expand their share of these markets.

To ensure that HWRs make the required contribution, a vision of the future must address the three drivers listed in the introduction to this report, i.e. improved economics, enhanced safety and sustainable development.

In addition to these three drivers, there are two key developmental factors that need to be exploited, which, if satisfactorily addressed, will facilitate renewed development of the industry and result in a greater share of the energy (electricity) market being gained. These are environmental protection and the broader applications of nuclear energy.

8.1.1. Environmental protection

Nuclear power used for electricity generation contributes a negligible amount of greenhouse gas, particulates and other polluting gases to the environment. The radioactive releases from nuclear power plants during normal operation are extremely small and pose no hazard to the health of the surrounding population. The substitution of nuclear stations for coal burning stations represents an important option for reducing the load on the atmosphere, which is contributing to poor health in densely populated areas as well as to climatic instability.

8.1.2. Broader applications of nuclear energy

While about 20% of total energy is currently used to generate electricity in Western economies, there is potential for the increased use of nuclear energy in process heat production. HWRs are already used as a heat source for heavy water plants, and there are a number of potential applications of nuclear energy to supply process heat for various purposes, such as:

- High temperature process heat industries such as metal processing require large heat inputs for melting and heating operations or electricity for electrolytic operations. In the case of HWRs, this may mean converting electrical energy in some way to high temperature process heat, possibly by plasma arc techniques.
- Relatively low temperature (<250°C) steam applications for processes such as desalination and oil extraction from tar sands or oil shale. As mentioned, CANDU HWRs have already been used to provide process steam to heavy water production plants. Very little modification of the basic process system design would be required to provide heat input to desalination plants. The in situ extraction of oil from tar sands is now done by injection of steam from horizontal steam lines drilled into the strata which causes oil to drain to parallel and lower collector pipes. The steam conditions required (temperature flow rate, etc.) are compatible with the steam temperatures generated by HWRs.
- District heating, which is commonly used in Eastern Europe and where nuclear power is a demonstrated and suitable source of heat in many locations.
- Use of hydrogen in fuel cells; the hydrogen being derived from the electrolysis of water. In order to achieve maximum benefits from its use, this requires the energy to be generated without contributing to greenhouse gas emissions. Nuclear is the most cost effective generating technology of the alternatives to fossil fuels. Another potential application is the generation of hydrogen for the production of methanol for automobile fuel.

The remaining sections will emphasize the components and expected developments in the three drivers described previously. The development 'thrusts' will then be illustrated with outlines of work being undertaken in three countries in relation to three different HWR concepts. These concepts are:

- The short and long term visions of an HWR as an evolutionary design distant from the current reference HWR;
- The Indian perspective of a long term lower cost HWR based on the boiling light water design;
- The ultra-safe concept developed by the Russian organization ITEP, which is based on the KS150 reactor, extends the line of heavy water moderated gas cooled reactors.

The national needs will vary and will depend on the circumstances and the resources available. In summary, the flexibility of the heavy water design permits a variety of concepts to be developed that satisfy the three drivers mentioned in Section 8.1.

8.2. ECONOMIC VISION

The economics of HWRs (and all water cooled reactors) is dominated by capital costs, the interest charges on which account for a major portion of the LUEC of the electricity generated. Some possible ways to reduce the capital cost contribution are discussed in the following sections.

8.2.1. Increased plant size

The HWR designs currently available on the market are rated at ~200 MW(e), 500–700 MW(e) and 900 MW(e). The size of these plants has evolved naturally and results from the energy requirements of existing customers and from a survey of market needs. As much of the electricity demand for the next century is projected to occur in areas with a high population density and relatively low land availability, larger HWRs of ~1200 MW(e) and greater are under consideration. Assuming no change in the essential channel parameters (i.e. channel power, core mean coolant (CMC) temperature), an increase in plant size alone leads to an economy of scale that will reduce levelized capital and O&M costs. As described in Section 4, the magnitude of the reduction depends on the ratio of the plant outputs to an exponent of 0.4–0.6.

Any of the proposed evolutionary changes to the reference design that are discussed in subsequent sections must compare favourably to an appropriately scaled existing design. It is also expected that future developments will build on the key features of the HWR design shown in Fig. 213.

It is noted that if the future market demands a smaller reactor, an appropriately sized HWR can be developed relatively quickly, by matching the required number of channels with the corresponding process equipment. The proven CANDU HWR fuel channel technology allows a range of reactor sizes, each utilizing the identical operational and licensing environments.

8.2.2. Thermal efficiency

To a first approximation, fuelling costs and specific capital costs are inversely proportional to the thermodynamic efficiency, which is governed by the CMC temperature [290]. Figure 214 illustrates the percentage cost reduction as a function of CMC temperature. In this example, the reference is a conventional 700 MW(e) plant with a CMC temperature of ~290°C. The 57% asymptote represents the limit set by Carnot efficiency.

It is recognized that the benefits portrayed in Fig. 214 are oversimplified because there will obviously be additional costs associated with increasing the coolant temperature and/or pressure, not to mention added costs for the secondary

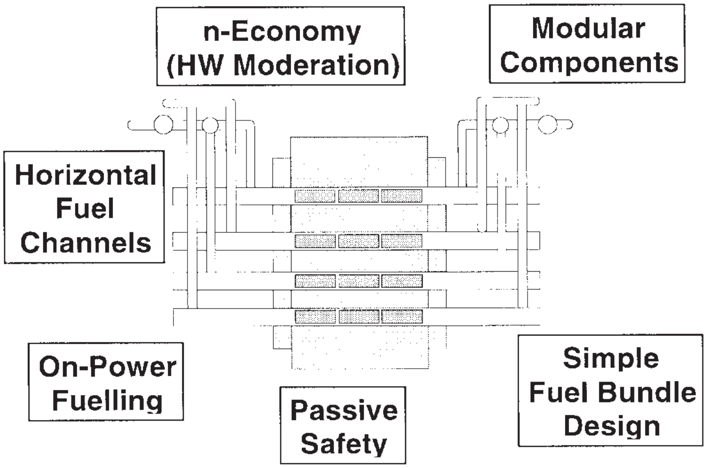


FIG. 213. HWR reactor features.

circuit and the balance of plant. The extent to which these factors would offset the savings provided by the increased efficiency has to be evaluated for any proposed change in CMC temperature, and hence, reactor design.

For example, the logical evolution of the HWR design would be to increase the CMC temperature of the heavy water coolant. In this case, the maximum outlet temperature will be limited to ~360°C, resulting in a cost reduction of about 8% in terms of thermal efficiency. However, as the temperature is increased towards this limit, the efficiency savings will likely be offset by increased costs associated with the

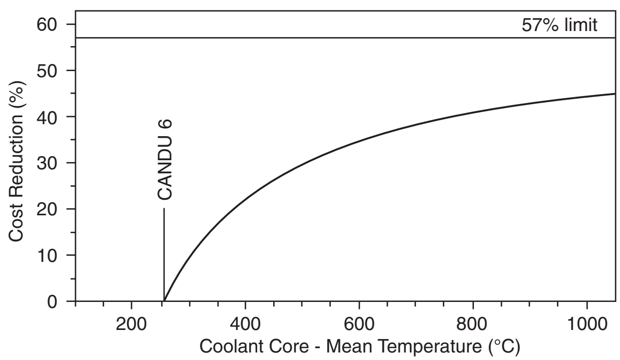


FIG. 214. HWR reactor efficiency.

fuel channel. That is, the increase in temperature and pressure may require either an additional pressure tube replacement over the reactor lifetime, or a new type of fuel channel, which may not require replacement, but would probably cost more to fabricate initially. It is also possible that higher fueling costs would be associated with a new channel design. The actual reduction in LUEC for a liquid heavy water cooled design will, therefore, be a few per cent, depending on how the above considerations can be optimized. However, development is proceeding along the lines described, where the effects of an outlet temperature of 325–330°C are being examined. To enable larger cost reductions to be realized, a higher temperature coolant is needed, where the difference between the efficiency gains and the capital and O&M costs is maximized.

8.2.2.1. Alternative coolants

Direct cycle HWR reactor concepts where the coolant was perfluorocarbon have been suggested and evaluated because this fluid exhibits good, high temperature stability at a fairly low operating pressure. However, the concept was abandoned after it was demonstrated that there is a marked incompatibility between the fluorine based chemicals and zirconium alloys. A concept proposing the use of N_2O_4 as a coolant was also abandoned because the compound is highly toxic, and the expense of developing an appropriate turbine set was deemed to be prohibitive.

The coolant choices for long term CANDU development have subsequently been reduced to superheated steam and supercritical water (SCW) with T_c >373°C and p_c >22 MPa. Even though the latter coolant requires a high operating pressure and a change in fuel channel design, it has attracted the most attention in recent years for numerous reasons:

- Much higher thermal efficiencies can be obtained with SCW compared with liquid (subcritical) water.
- The absence of a two phase region in SCW alleviates concerns of fuel dry out and flow instabilities.
- The mean coolant density is reduced by a factor of three or four, depending on the temperature, thus leading to a reduction in heavy water inventory.
- The reduced density of SCW leads to a reduced void reactivity, sufficient that light water becomes a coolant option.
- SCW has a high specific heat near the critical point, which for the same channel power and core temperature rise leads to an increase in the core enthalpy change and a commensurate reduction in channel flow.
- The large expansion coefficient (including through the critical point) opens up the possibility of natural circulation primary flow.

Current fossil fuel boilers can operate under supercritical conditions, and turbines have been designed to meet inlet temperatures of >400°C. Moreover, studies on LWRs operating with SCW are currently under way in Japan and the Russian Federation and these have shown that supercritical reactors are feasible with modest extrapolations of current technology [291–293].

In representng the SCW cooled reactor as the epitome of HWR evolution, a logical development path from a conventional HWR emerges. The development path and the technological requirements needing to be satisfied in order to achieve such a reactor are discussed in the following sections.

8.2.3. Simplified design and construction efficiency

Costs can be favourably influenced by matching design specifications to operating conditions. This is particularly important for the design of auxiliary systems and those which have an insignificant effect on safety. Such designs can potentially use PSAs or risk informed analyses as the basis of the amended design or design specifications.

Vendors are now in the position of having built a number of reactors, within predicted cost and schedule, using improved construction techniques, particularly computerized project control. Continued improvements in construction methods, for example, the use of modularized design, will improve schedules further. Costs in terms of LUEC can be reduced by standardizing designs.

8.2.4. Ease of O&M

The higher the capacity factor, the lower the LUEC. Capacity factors are determined by the effectiveness of plant management in developing training programmes and operating procedures, and in ease of plant O&M. Thus, a design layout that facilitates necessary maintenance under a well planned life management programme, will inevitably increase capacity factors. The value to the owner of a nuclear plant performing at a 90–95% capacity factor compared with one at 70–75% capacity is easily calculated. HWRs with on-power fuelling make possible high capacity factors and it is within the limits of current technology to expect limited shutdowns for planned maintenance only and capacity factors of ~94%.

8.3. SAFETY VISION

As described in Section 5.5.1, it is expected that within the next ten years the development of safety enhancements will follow a path determined by the need to be cost effective. The items that will receive attention are:

- Features that reduce the likelihood of initiation of postulated severe accidents and that mitigate the consequences. This will take advantage of the safety features inherent in the heavy water design that can be further enhanced, such as:
  - — The ability to add or recirculate water to the moderator, end shields or shield tank to remove heat and contain core damage, with additional heat removal from containment achieved by air coolers;
  - — The control of hydrogen within containment by the addition of hydrogen recombiners and hydrogen igniters;
  - — The addition of emergency feedwater, either pressurized or unpressurized, to the secondary side of the steam generators;
  - — The modelling of severe accident sequences to increase understanding and guide designers, and the development of severe accident management programmes for operators to ensure that no evacuation is necessary.
- Reduction of the exclusion area boundary by establishment (by calculation and the provision of a low leak rate from containment) that small exclusion area boundary limits can be justified.
- Action taken to address the potential for failure of containment isolation for design basis accidents.
- Introduction of a passive system for containment pressure suppression following a postulated high pressure system break within containment.
- Systematic introduction of human factors engineering into the design process, particularly for the control room, in aspects such as layout, the human–machine interface, large overhead ‘mimics’ of system status and computer assisted event diagnosis. The objectives are to achieve:
  - — Increase in allowable operator action time to at least 8 h for ‘design basis’ events. This will probably be achieved by the automation of existing systems.
  - — Better layout of the plant for compactness (MW(e)/m^2), orientation of the turbine generator and stronger separation of Group 1 and Group 2 safety systems.
  - — Changes to process systems and design to:
    - (a) Reduce the likelihood of loss of inventory of the HTS sufficiently to allow thermosyphoning to operate as a heat removal mechanism,
    - (b) Increase moderator system margins by preventing prolonged dry out of the calandria tube,
    - (c) Prevent end fitting ejection for a postulated guillotine failure of the pressure tube,
    - (d) Increase separation of Group 1 and Group 2 electrical services to the main control room so that operation can continue in the aftermath of a severe earthquake.

- Simplification of the ECCS; some simplification having already been achieved.
- Reduction in public and operator doses as a result of reduced activation from corrosion, more efficient dryers and containment ventilation, segregation of high tritium and low tritium areas, welding of feeders to end fittings, and more economic extraction of tritium.
- Optimization of protection against external events using a probabilistic approach.
- Replacement of conservative analytical codes for safety analyses of multiple failures as part of the design basis, by realistic analyses coupled to a bounding uncertainty analysis.

For the next generation HWRs (next ten to twenty years), an evolutionary development path will be followed, based on cost benefit and risk reduction concerns. It is expected that advanced designs will incorporate a number of passive features, where they are simple and reliable, but that active systems will be retained where safety and reliability are adequate or demonstrably sufficient. The requirement of having postulated accidents evolve over longer time-frames will be satisfied without the need for evacuation of the public. Combinations of passive and active safety systems, or hybrid systems, invoke diversity and redundancy in safety system principles of operation and are highly attractive ways of meeting safety goals. These next generation reactors will likely have aspects of the hybrid safety systems incorporated into current evolutionary designs. In this way the need for prototype plants is avoided.

The use of passive safety systems in the period ten to twenty years hence will be related to the most effective application of passive features to the three safety functions — shutdown, cool and contain. Of these, heat removal (cooling) is anticipated to show the more effective use of passive features and is suited to low energy density systems. Hence, the most likely use of passive heat removal is in decay heat removal systems from the reactor, the steam generators, the moderator, the shield tank or the containment. The large inventories of water in the moderator, end shields and shield tank lend themselves easily to passive heat removal. The near ultimate heat sink would comprise an elevated water storage tank, large enough to absorb decay heat for several days.

The other safety functions, shutdown and contain, are near passive in execution and perhaps limited in further development potential. Shutdown margins can be increased by reducing the positive void coefficient with, for example, enriched fuel. Containment designs could develop more towards double containment concepts, but at some increase in cost.

An important passive safety feature under development is PEWS, which forms a heat sink as mentioned above. The ultimate heat sink is the atmosphere and local water bodies. Movement of heat to outside containment can be effected by conduction through walls, by steaming or by air-cooled heat exchangers. Closed loop heat rejection is preferred for longer time-frames.

Beyond twenty years, it is expected that for HWRs to sustain a large market share, the plants will have to operate at higher thermal efficiencies. This implies the use of high temperature coolant or supercritical water as coolant. Such reactors may use passive safety in the following ways:

- Use of a passive high temperature channel,
- Elimination of the consequences of channel flow blockage,
- Use of natural circulation heat removal wherever possible,
- Use of passive containment heat removal.

Only the passive high temperature channel has not already been discussed. This channel concept is an insulated channel design. It has no calandria tube and the cold pressure tube is in contact with the heavy water moderator. The objective is to transfer sufficient heat through the insulating material and the pressure tube to the moderator so that the fuel is not damaged in an accident, even for a loss of coolant to the channel. In normal operation, heat losses to the moderator are acceptable. The channel design CANTHERM opens up the possibility of passive heat rejection from the fuel to the moderator, either with or without heat removal from the coolant and with little fuel damage.

8.4. VISION OF SUSTAINABILITY

It is likely that there is no unique fuel cycle path appropriate for all countries using HWRs (the particular fuel cycle chosen will depend on a range of local and global factors). Advanced fuel technology provides a means of reducing capital and fuel cycle costs, and spent fuel volumes; enhancing safety; extending plant life; increasing operating margins and extending uranium reserves, and acts as a vehicle for dispositioning weapons grade isotopes.

Over the next ten years, advanced fuel designs will be used increasingly by operators. Depending on the cost involved, these fuel designs will likely be accompanied by the gradual introduction of enriched fuel or recycled uranium from spent PWRs. The use of SEU/recycled uranium is expected to show a significant cost advantage, improved operating margins, improved power uprating capability and produce reduced quantities of spent fuel. Thus, the next few years will see demonstrations of the use of SEU in power reactors, and eventually, of recycled uranium. The use of SEU in new reactors will enable more power to be derived from a given size of reactor. The use of higher enrichment, together with a tighter lattice pitch and light water coolant, are features that allow significant capital cost reductions to be achieved. R&D will continue to provide the technological base for more advanced fuel cycles. Experience will continue to be gained in thoria based fuel cycles and in

DUPIC fuel cycles, although both will require investment in processing and fabrication technology.

In the ten to twenty year time-frame, the use of SEU/recycled uranium in HWRs will have become widespread. Enrichment levels may increase to 1.2–1.5%, driven by lower enrichment costs and the need to achieve capital cost reduction in plants.

Further testing will be done on advanced fuels and on fuel bundles reaching higher burnups, higher sheath temperatures, etc. DUPIC or MOX from recycled LWR fuel is likely to be commercialized (see Section 6).

The period beyond twenty years should witness much lower capital cost HWRs exploiting the advanced fuel cycles and operating with higher sheath temperatures, higher coolant temperatures achieving higher efficiency, and negative or low positive void reactivity enhancing passive safety. The objective will be to achieve higher power output with increased operating and safety margins. The high fuel conversion ratio of the HWR means that it can be used to advantage in a combined fuel cycle, including FBRs, to extend both uranium and thorium utilization to very low grade ores.

Specifically, high burnup MOX (benefiting from the PWR/HWR synergism) and high burnup SEU will be exploited, together with thorium cycles. Probably the first demonstrations of HWR/FBR synergism would occur in this time-frame. Inert matrix fuel for actinide burning in HWRs could be implemented if conventional processing of PWR fuel continues.

8.5. CONCEPTS UNDER DEVELOPMENT

8.5.1. PHWR

In the case of the PHWR with horizontal channels, there are three concepts under development (one with two variants) with different time-scales for completion.

In the first, the CANDU HWR concept is under continual evolution, with the evolutionary CANDU 6 (700 MW(e)) and the designed single unit CANDU 9 (900 MW(e)) based upon earlier, integrated multiunit stations. The design of the CANDU 9 is complete.

The second concept envisages a lower capital cost CANDU producing electricity competitively in comparison with the combined cycle gas turbines. This next generation CANDU (described in more detail in Section 8.8) will reduce capital costs by (a) increasing the outlet temperature to ~330°C and the outlet end pressure to ~13 MPa, (b) decreasing the size of the calandria to reduce heavy water cost, (c) increasing the power output of each channel from the use of enriched fuel (~1.5%), (d) decreasing the lattice pitch, and (e) reducing the capital cost by the use of light water coolant. The power output of this reactor is nominally 600 MW(e). The time-scale for completion of development is 2005–2007. The higher outlet temperature and the tighter lattice pitch both introduce technical complexity which will be the subject of significant design and experimental development.

The third concept envisages a supercritical water cooled heavy water moderated reactor employing an indirect cycle with either light water or heavy water cooling. Two variants of the concept have been examined, each at a nominal operating pressure of 25 MPa [290]. The enthalpies, as functions of temperature for the two cases (termed Mark 1 and Mark 2, respectively), are compared with a standard 700 MW(e) CANDU 6 in Fig. 121.

Mark 1 is considered to be achievable with modest improvements to current materials and equipment designs, and, hence, has received most of the development effort. Mark 2, on the other hand, is more revolutionary, and requires more development, particularly since the challenges facing the materials in meeting the proposed temperatures are expected to be significant.

In the Mark 1 concept, the coolant temperature increases from 380°C to 430°C (CMC ~400°C) in the primary system. This temperature range was chosen because it roughly reflects the maximum that could be endured by a conventional, or near conventional, fuel cladding made from a zirconium alloy. However, it also takes maximum advantage of the high specific heat of water near the critical point. Compared with CANDU 6, the higher specific heat results in a core enthalpy that is increased by a factor of three. Thus, for a given channel power and core temperature rise, there is a threefold reduction in mass flow. The pressure drop across a channel would be similarly reduced. These factors, and the relatively high coolant density at the pumps (0.45 g/mL), result in a significant reduction in the primary pumping power.

The high specific heat at the critical point also leads to high heat transfer coefficients from fuel to coolant, and across the steam generator. In the case of Mark 1, the design concept for the secondary circuit involves the transfer of heat to water in a once through counter current flow steam generator operating at 19 MPa (Fig. 215). Representative temperature profiles through the steam generator are illustrated in Fig. 216. The resultant thermodynamic mean temperature of 360°C yields a Carnot efficiency 21% greater than the CANDU 6, leading to a potential cost reduction of ~18%.

A turbine design capable of operating at these temperatures and pressures has been established, based on existing supercritical designs and experience. The concept involves the use of a very high pressure turbine with the exhaust directed to a conventional LWR turbine set. The addition of a high pressure turbine will increase the capital cost of the plant, but thermal efficiencies near 40% are projected. An alternative concept would be to use a dual reheat scheme. An additional projected 3% gain in efficiency would have to be balanced by the added expense of transferring heat

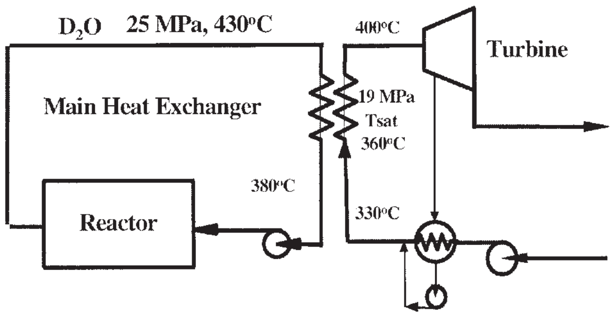


FIG. 215. The Mark 1 concept.

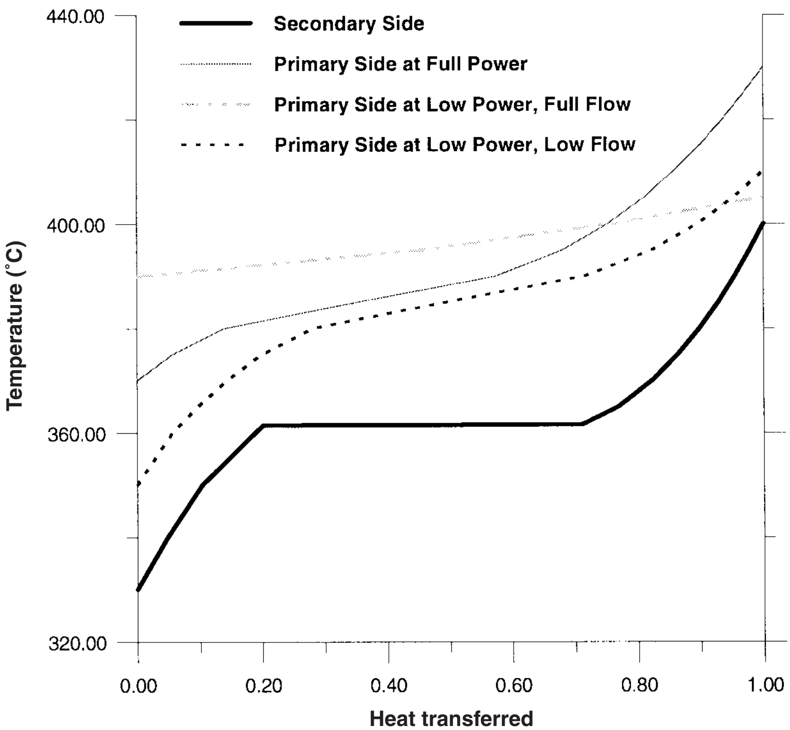


FIG. 216. Steam generator profiles in the Mark 1 design.

from the primary side. In either case, the use of a 1250 MW turbine is consistent with the development of larger plant outputs with reductions in capital cost.

A further cost reduction may be realized because the containment design pressure will be reduced compared with a CANDU 6, for example. Although this appears to be counterintuitive because of the proposed operating temperature and pressure of Mark 1, the actual stored energy released to containment following a LOCA would be less. That is, the reduction in coolant inventory (three to seven times) more than offsets the increase in specific enthalpy (about two times). Hence, for the same design pressure, the containment volume can be reduced by at least 50%.

In the Mark 2 concept, since the specific heat of SCW is much reduced above the critical point, the design concept requires a temperature rise of 100°C across the channel if the inlet temperature is 500°C [290]. These temperatures will necessitate that significant design changes be made to the fuel, cladding and fuel channel. Moreover, the pumping power will be greatly increased because of a larger mass flow requirement (resulting from the reduced specific heat) and a much lower coolant density at the pumps. These factors will be offset somewhat by increased thermodynamic efficiency, and a reduction in D_2O inventory (if heavy water cooling is to be maintained).

A comparison of the salient reactor parameters of Mark 1 and Mark 2 are shown in Table LVIII.

It should be noted that, for both the Mark 1 and Mark 2 designs, these values represent a first iteration towards a design optimization. It will be necessary to compare the cost savings based on thermodynamic efficiency with the cost increases associated with new materials, steam generators, balance of plant, etc. On the basis of such an economic assessment, it will be possible to optimize the values in Table LVIII accordingly.

8.5.1.1. Heavy water inventory and primary pressure control

The CMC density of 0.28 g/cm^3 could lead to a 70% reduction of coolant inventory at full power. However, additional heavy water might be needed to fill the PHTS at reduced power, especially in the cold shutdown condition. Three options are discussed below aimed at avoiding such a need and leading to a 3% additional capital cost reduction.

The first option would use a pressurizer with helium and heavy water. On cold shutdown, helium would enter the large piping and steam generator piping (to accommodate heavy water shrinkage), but there would be sufficient heavy water to fill headers, feeders and fuel channels. Startup, in this case, would be achieved with nuclear heat or with an external heat source. With nuclear heat, after going critical, the temperature would be raised at low power. As the heavy water boils at lower pressures, steam and helium would flow to the pressurizer where the steam would be

TABLE LVIII. SCW COOLED HWRs AND REFERENCE CANDU 6 DESIGN VALUES

Parameter	Reactor		
	Mark 1	Mark 2	CANDU 6
Reactor thermal power (MW)	2280	3400	2159
Reactor electrical power (MW)	910	1500	668
Efficiency (estimated) (%)	43.5	50	30.4
Core inlet temperature (°C)	380	500	266
Core outlet temperature (°C)	430	600	310
Inlet density (kg/m^3)	446	90	780
Outlet density (kg/m^3)	122	70.8	690
Total core flow (kg/s)	2530	10 500	7700
Average channel power (MW)	6	9	5.4
Average channel flow (kg/s)	6.7	27.8	24
Peak channel power (MW)	7.2	11	6.5
Maximum sheath temperature (°C)	465		320

condensed, purging helium from the HTS. When operating pressures and temperatures are reached, pumping would start, enabling power to be increased.

The second option would be to transfer heavy water from the moderator to the PHTS in order to accommodate primary system shrinkage during cooldown. This could present problems with tritium releases from higher PHTS tritium levels, and with the cost of higher PHTS heavy water isotopic concentration. However, the latter could be compensated for by the use of a common heavy water upgrader.

With reduced moderator inventory, the reactor would not be able to be started on nuclear heat. However, the PHTS would be full and would be heated on pump heat, augmented if necessary by external heaters. The excess hot heavy water would be bled off and cooled during transfer to the moderator. Once operating temperatures and densities in the primary circuit were attained, the reactor would go critical.

The third option would be to adopt H_2O as the primary side coolant instead of D_2O, and use the excess H_2O for secondary side and RHR systems.

8.5.1.2. Natural circulation

An existing HWR is capable of removing decay heat by natural circulation. Natural circulation is not only a passive safety feature, it can also reduce the LUEC by reducing the load on HTS pumps. Since the thermal expansion coefficient of SCW is roughly a factor of ten greater than that of ordinary water, it is possible that a SCW cooled HWR could naturally circulate in all operating conditions, including shutdown. This will only be possible for a primary circuit operating around the critical point (i.e. Mark 1) where the density change is greatest. Operating through the critical point has the added advantage of maximizing the enthalpy change, thereby reducing the flow requirement for a given channel power. Even so, a change in the fuel bundle design would likely be required to decrease the pressure loss around the circuit. Preliminary calculations indicate that under these conditions a driving head of approximately 14 m would be sufficient to sustain a naturally circulating reactor. An example of some 'scoping' calculations for a 900 MW unit having about 380 naturally circulating channels is shown in Fig. 217.

8.5.2. Advanced fuel channel designs

In a CANDU 6 reactor, 380 horizontal fuel channels pass through the calandria vessel. Each channel consists of a pressure tube which acts as the pressure vessel, a

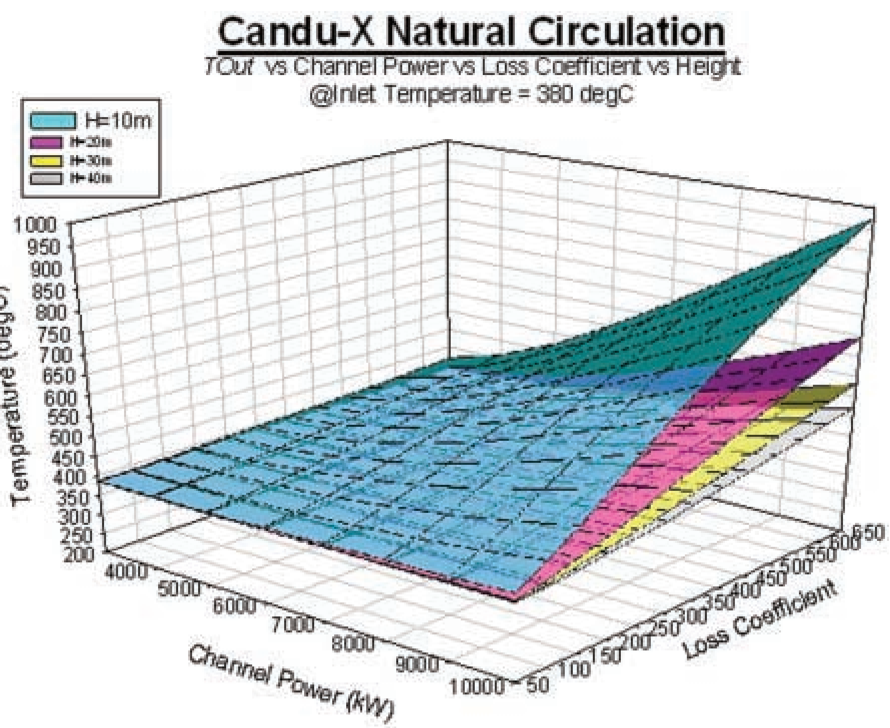


FIG. 217. Natural circulation of water under supercritical conditions.

calandria tube, stainless steel end fittings at the end of each pressure tube, and four spacers which maintain the annular separation between the pressure tube and the calandria tube. It is this separation that provides the thermal barrier between the coolant and the moderator.

The pressure tube is the component that determines the lifetime of the conventional fuel channel. Each tube is fabricated from cold worked Zr–2.5%Nb to a length of approximately 6 m, and with an inner diameter of 103.4 mm. Under the CANDU 6 operating conditions of ~310°C at 10 MPa, a pressure tube manufactured to current requirements has an expected lifetime of 30 years at an 85% capacity factor. The lifetime limit is principally dictated by diametral creep, which decreases the regional overpower protection margin. Another factor affecting the lifetime of the pressure tube is deuterium ingress.[9] The buildup of deuterium results in increased susceptibility to delayed hydride cracking and a reduction in fracture toughness. The deuterium concentration should not exceed the terminal solid solubility at operating temperatures.

In the case of the next generation low capital cost reactor, the channel design will be an extension of existing technologies and intended to meet the higher temperatures and tighter lattice pitch (~220 mm). Thus, the pressure tubes will be considerably thicker to sustain the pressure, and the life of a pressure tube will be approximately twenty years with the capability for fast retubing. The calandria tube will have a larger diameter than the current design and be separated from the pressure tube by a new design of spacer.

In the case of a reactor cooled by SCW, the conventional, or usual, fuel channel described above is not a viable option. Diametral creep and corrosion rates will be exacerbated under supercritical conditions. In addition, a significant increase in the thickness of the pressure tube would be necessary to accommodate the pressure of SCW because the ultimate tensile strength of zirconium alloys falls precipitously with temperature. This increase in thickness may offset some of the issues with regard to creep and corrosion rates, but at the cost of a substantial increase in parasitic neutron absorption; enrichment of ^{235}U or depletion of the ^{91}Zr isotope in zirconium alloys in the core would be necessary to achieve acceptable burnups (perhaps even criticality).

A high temperature channel is, thus, being developed where the pressure tube is thermally insulated from the coolant, allowing the tube to operate at approximately the moderator temperature. This design removes the need for a calandria tube. Experience gained with the EL 4 reactor in France has shown that an insulated channel with a similar configuration is a viable operating concept for a pressure tube reactor [294].

[9] As the coolant in an advanced HWR may be either light or heavy water, hydrogen and deuterium will be used interchangeably.

Given that neutron economy must be preserved, even if an enriched fuel is used, a zirconium alloy is the material of choice for the pressure tube. The use of zirconium alloy, operating at the comparatively low temperature of 70–100°C means that the strength, and the likely creep resistance, will be such that the thickness of an insulated pressure tube will be roughly equal to present HWR designs, therefore a significant neutron penalty is not expected. In addition, the reduction in the corrosion rate at 100°C will reduce the total amount of deuterium incorporated in the pressure tube. As the terminal solid solubility of deuterium is considerably less at low temperature, the manufacturing specification for hydrogen must be kept below ~1 ppm by weight, or, alternatively, the pressure tube fabricated from an alloy resistant to delayed hydride cracking.

Two options are under consideration with regard to the insulation lining the inside of the pressure tube (Fig. 218). The first is a design in which the insulation is provided by a solid insulator of an appropriate thickness. The coolant and fuel pass through a thin liner, or guide tube, which is made from a material compatible with SCW. To preserve neutron economy, the insulator should be as thin as possible, hence, materials with a relatively low thermal conductivity ($<5\ W \cdot m^{-1} \cdot {}^{\circ}C^{-1}$) are under consideration. The capability to transmit the operating pressure to the pressure tube is another consideration.

The second option is a design in which the insulation is provided by a semi-porous solid, where the coolant trapped within the pores provides the thermal insulation. The liner tube is macroporous, and acts only as a guide for the fuel bundles. In this design, some of the constraints associated with the solid insulator are removed or reduced: load transmission is no longer an issue because the porous insulator allows the coolant pressure to act directly on the pressure tube and materials with higher thermal conductivity can also be considered because the thermal conductivity of the insulator is dominated by trapped coolant, not the insulator.

Option 2 is not a panacea, however, since the insulator must show acceptable stability in SCW, which is generally much more corrosive than high temperature liquid water. Radiolysis of the SCW trapped within the pores complicates the corrosion issue, but it should be negligible if proper chemistry control is maintained.

8.5.2.1. Channel safety considerations

The insulated channel, where a robust pressure tube is in direct contact with the moderator, requires different safety methodology than that employed with conventional HWR fuel channels.

One item to be considered is the retention of the leak before break capability. In a conventional channel, a leak in a pressure tube is detected by an increase in the moisture level in the annulus between the pressure tube and the calandria tube. Although a thorough analysis indicates that leak before break would be maintained

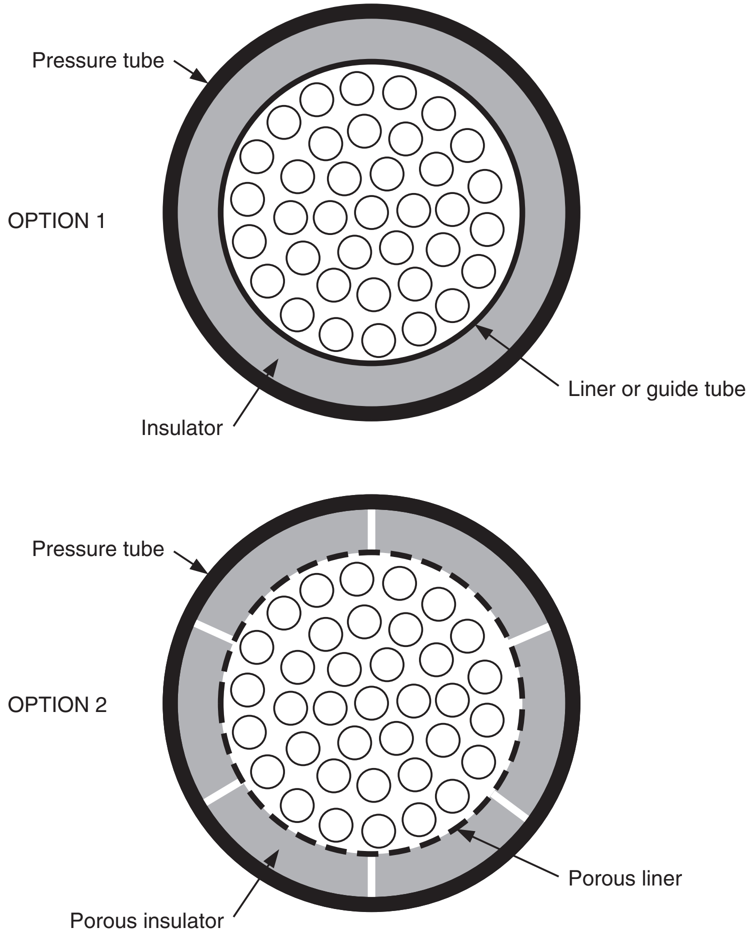


FIG. 218. Option 1 for the insulated pressure tube concept and option 2 for the insulated fuel channel concept.

for zirconium alloys in an insulated channel, the strategy of monitoring the moisture level is not applicable because there is no calandria tube. Instead, a leak would have to be detected within the moderator. As a result of the high coolant pressure, the leak rate into the moderator would be quite high, even for short ‘through wall’ cracks. One approach would be to use the noise associated with the subcooled collapse of steam bubbles in the moderator; this may be sufficient to act as a leak detection signal when received by an acoustic sensor. Alternatively, a chemical or radiochemical tracer (e.g. ^{23}Ne) could be employed within the coolant.

In the event of a single channel blockage, passive safety can be achieved with the insulated design. One method proposed to minimize the probability of blockage occurring is the addition of a second feeder, as illustrated in Fig. 219 Flow monitoring within the channel may still be a requirement, however, and, hence, an advanced flow monitoring capability is desirable. A single channel blockage in a conventional channel requires that residual heat from the fuel be rejected through the annulus into the cool moderator. The temperature 'spike' may cause the pressure tube to deform plastically (balloon) and make contact with the calandria tube, after which the heat is removed more readily. In the case of an insulated channel, the pressure tube is designed to be in direct contact with the moderator. Moreover, the thermal conductivity of the insulator will be approximately two orders of mag-nitude higher than the annulus gas system. As a consequence, heat is passively transferred more effectively to the moderator under accident conditions, and the structural integrity of the pressure boundary is preserved. It may also be possible to 'tailor' the structure of the insulator so that it deforms in such a way under accident conditions that its thermal conductivity is further increased. The insulator would have to be replaced as a result, but the pressure boundary would remain intact.

8.5.3. Fuel and fuel cycle

A conventional HWR offers unmatched flexibility in fuel and fuel cycles. It is, therefore, important that future HWR designs possess similar flexibility. As a starting point, the CANFLEX design described in Section 6, which is fast becoming the preferred design for present day HWRs, will be used for the SCW cooled design. Its suitability must be established on the basis of the specifics of the SCW cooled design, for example, whether natural or enriched uranium is to be used, the temperature of the coolant, and possible increases in bundle/channel powers. In the following sections these issues are addressed independently, whereas in reality, they must be evaluated together in order to arrive at a final fuel design.

8.5.3.1. Natural uranium and SEU fuel options

In some future markets, the flexibility of a natural uranium cycle, which has been the hallmark of CANDU, may still be required (cf. Section 6). Natural uranium can remain an option for an SCW cooled reactor only if the neutronic absorption of the eventual insulated channel is approximately equivalent to that of the conventional channel, thus preventing a significant reduction in present value burnups (~7000 MW·d/t U). The burnup will also be affected by possible changes to the fuel sheath in the form of more corrosion resistant stainless steel or nickel based alloy that may be required to accommodate the higher coolant temperatures. Zirconium alloys with ^{91}Zr reduced, that is, the isotope exhibiting the largest absorption cross-section for both fast and thermal neutrons, is one possible solution, but the increase in origi-

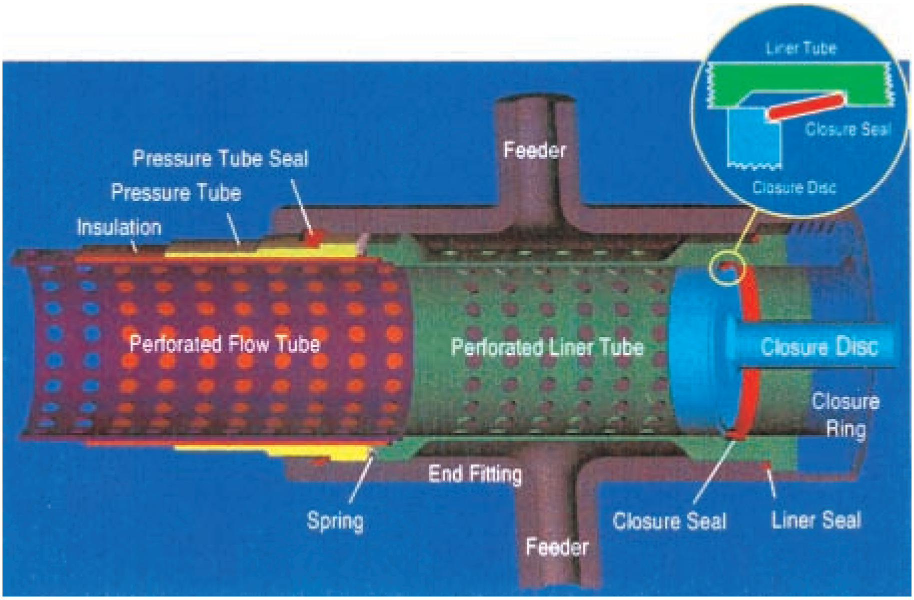


FIG. 219. Possible end fitting design for an insulated fuel channel.

nal manufacturing costs would be substantial using conventional technology. In the final analysis, the benefits of using natural uranium for an SCW concept would have to be evaluated against any increase in fuelling or channel costs.

If sufficient neutron economy cannot be established at an affordable price, SEU will have to be adopted, consistent with the anticipated evolution of the fuel cycle in conventional HWRs. A ^{235}U content of between 0.9% and 1.2% would increase the burnup, and, hence, reduce the quantity of spent fuel produced. Use of SEU would further improve the uranium utilization (the energy derived from the mined uranium). A reduction of about 25% in uranium requirements (per unit energy) is achieved for enrichments between 0.9% and 1.2%. Uranium utilization is an important consideration in countries that have few indigenous uranium resources. Enrichments of between 0.9% and 1.2% also reduce CANDU fuel cycle costs by 20–30% compared with natural uranium fuel cycle costs.

SEU also offers flexibility in reactor design. It can be used to uprate reactor power without exceeding existing limits on bundle or channel power, by flattening the channel power distribution across the reactor core. In any new reactor design, the use of power flattening to obtain more power from a given size of core offers a capital cost advantage over the addition of more channels to the reactor. Alternatively, the power flattening from SEU could be used to lower the peak bundle and element ratings without increasing reactor power. If CANFLEX bundles were used as the carrier for SEU, the peak linear element ratings could be reduced, thus significantly reducing fuel temperature, fission gas release and fuel failure probability [295].

Once a decision is taken to use SEU, some incentive for retaining heavy water cooling is removed. This is true for both conventional and SCW cooled reactor designs, and will lead to a reduction in the LUEC for the reactor. Preliminary calculations indicate that light water cooling within an insulated channel will require enrichment to ~1.2%, at a reduced coolant void reactivity.

8.5.3.2. Fuel sheath

The primary focus of sheath development will be prevention of corrosion, the extent of which is a function of the outlet temperature and the time spent in-reactor (i.e. burnup). Fortunately, the high specific heat near the critical point leads to high heat transfer coefficients and only modest increases in fuel cladding temperature. The first estimates made from a standard 37 element bundle geometry operating at a maximum outer element power of 50.7 kW/m and a coolant temperature of 400°C yielded a nominal cladding maximum temperature of 450°C, even at reduced channel flow. The cladding temperatures would be further reduced with 43 element CANFLEX fuel [290].

The operating conditions for Mark 1 were based, in part, on the maximum temperature that could be endured by a conventional, or near conventional, fuel cladding made from a zirconium alloy. Previous data have demonstrated that a zirconium alloy can successfully operate in superheated steam (500°C) for a period of six months. Under these conditions, a metal loss of 5–10% of wall thickness is anticipated [290]. More recent data, acquired out of flux, indicate that the same alloys will likely be suitable in projected Mark 1 conditions if high burnups are not required and if good chemistry control is maintained. Higher burnups may require that the cladding be coated with a thin, corrosion resistant film. Various application methods for different metallic and ceramic coatings are being investigated. If a satisfactory solution cannot be found using a zirconium alloy, then stainless steel or a nickel based cladding could be used, but the necessity of using enrichment to account for the reduction in neutron economy makes this a less attractive approach.

At temperatures beyond 500°C (i.e. Mark 2), a corrosion resistant coating will have to be employed if a zirconium based alloy is used. A significant fuel design change is another possibility, a change to an all ceramic bundle similar to the SiC matrix composite fuel cladding being developed for high burnup LWR fuel. The concerns regarding ceramic cladding are a lack of toughness and possible failures resulting from tensile forces arising from coolant turbulence, pellet and fuelling machine interactions, etc. A ceramic cladding would have to be free standing, or it would crack and fail as a result of thermal expansion. The compressive stresses arising from the coolant pressure should overcome any such tensile forces, and therefore the use of ceramics should not be precluded on the grounds of brittleness alone. Methods such as fibre reinforcement also exist to improve toughness.

The major technological development required for a ceramic sheath, surprisingly, may be that of corrosion resistance. There are few reports in the literature on the interaction of SCW with ceramics, but many materials normally considered for applications in high temperature water exhibit either poor phase stability or unacceptable corrosion and/or solubility rates in SCW.

8.5.3.3. Bundle power

The average channel power for a 900 MW(e) Mark 1 design is proposed to be ~6 MW, about the same as in a conventional CANDU 6 design. Consequently, CANFLEX will be an acceptable bundle design, as long as an appropriate sheath can be found to account for the higher coolant temperature. Any requirement for enrichment will be largely dictated by the channel design, as discussed above.

If economics dictate that the Mark 1 output should be greater than 900 MW(e) and that the number of channels be fixed, then channel power will have to be increased. This will necessitate a change in fuel design because the consequent increase in central fuel temperatures will result in an increase in fission gas release within the individual fuel elements.

Perhaps the simplest way to achieve higher power is to further subdivide the bundle. For a given bundle power, more elements yield lower peak element ratings, thereby keeping fuel central temperatures to acceptable levels. The 43 element CANFLEX allows more margin for increased fission gas release than do conventional 37 element bundles, and its power limit for this application has to be evaluated. Further subdivision, to, for example, 61 elements, could also be considered. The advantage of subdivision is that a natural uranium fuel cycle can probably be maintained.

A second option would be to use a CANFLEX design, with annular fuel and/or graphite spacer discs: it is possible to use one without the other, but maximum gain is obtained when both are used. The addition of graphite discs between fuel pellets lowers fuel temperatures by providing high thermal conductivity paths from the centre of the rod to the coolant at the surface. Annular fuel removes the hottest part of the fuel (i.e. the centre of the pin), allowing higher powers to be achieved without higher temperatures. The annulus also provides a natural plenum (free volume) in which to house fission gas. The removal of fissile content from within the bundle, however, probably removes the option of using a natural uranium fuel cycle. At AECL, considerable data were obtained on both annular fuel and graphite spacers in the late 1960s and early 1970s for a 37 element bundle.

Another option for consideration is inert matrix fuel. In this design, uranium is mixed, as a solid solution or second phase, into a high conductivity non-fissile (inert) matrix, such as SiC. Such fuel types are being explored by other countries as the means of burning plutonium or annihilating actinide waste. A favoured option is low conductivity ZrO_2. For use in an SCW cooled reactor, an inert matrix fuel design would require the most development effort. Nonetheless, there is a high level of confidence that an acceptable design could be produced. In the very long term, development could be envisaged of a fully ceramic bundle made from an inert matrix fuel that exhibits sufficient chemical and mechanical integrity such that no sheath is required.

8.5.3.4. Fuel cycle

Depending on the enrichment and refuelling strategy, the fuel cycle can be from 60 d to 180 d (800–2400 MW·d/t burnup), potentially reducing fuelling machine usage by >50% and fuel waste volume by a similar amount. Thus, the corrosion lifetime of the fuel cladding, as well as the fission gas capacity of the fuel itself, will effectively determine the optimization of the fuel cycle length, burnup and maximum linear power [295]. In addition, in the longer term, optimization of the mix of CANDU reactors and LWRs will enable the LWR fuel to be reused in the CANDU plants, thus extending the fuel utilization and uranium (energy) resources. The effect of such optimization is illustrated in Fig. 220, where the once through LWR cycle is compared with other options, including MOX recycling. The potentially increased efficiency of the SCW cooled HWR concept then allows for even further reduction in uranium requirements and can significantly extend the useful life of known resources.

8.5.3.5. Refuelling

On-line refuelling has been one of the hallmarks of the HWR design, and this will continue to be the case with the SCW cooled design. To offer even greater flexibility, one strategy would be to use single ended refuelling. Bundles will be withdrawn from the channel into a magazine on the fuelling machine outside the core, allowing the bundles to be shuffled to obtain the desired flux profile. Since there is a large coolant density change across the core, refuelling will probably occur with the flow direction, such that new bundles are positioned where the coolant density is highest. Figure 221 shows the calculated k_{eff} as a function of burnup for 1.2% SEU cooled by light water in an insulated channel for refuelling with, and against, the flow direction.

8.5.4. Alternative approaches

To extend the benefits of nuclear energy beyond electricity generation, cogeneration options should be explored for short term nuclear power plants and their long

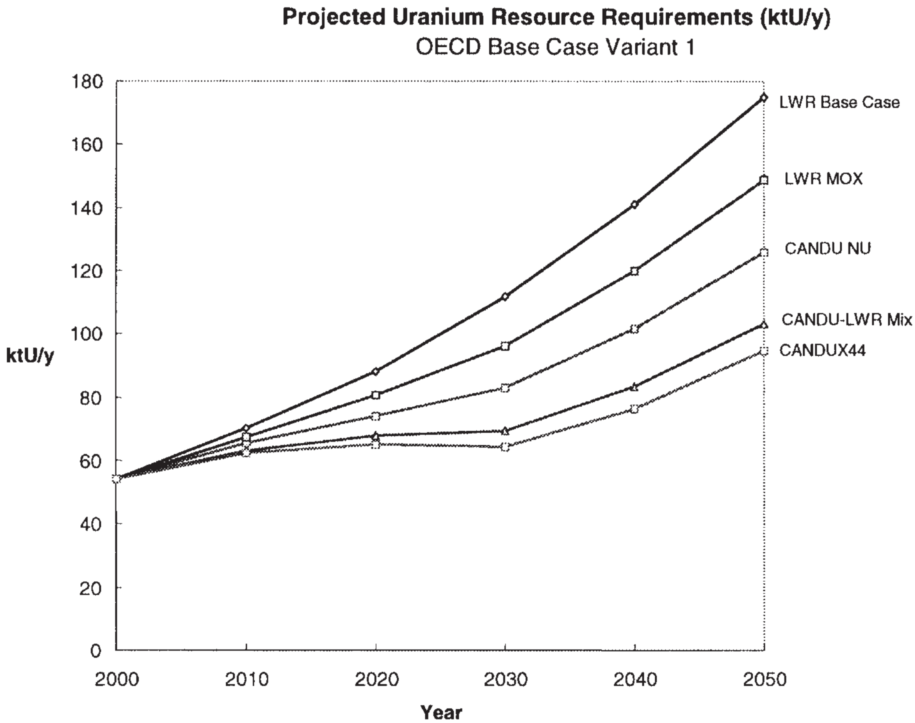


FIG. 220. The CANDU fuel cycle.

term products, such as the SCW cooled HWR. An example of nuclear cogeneration recently identified is the synergistic combination between an HWR and emerging technologies such as hydrogen. This opportunity arises because it is necessary to reduce greenhouse gas emissions, whilst keeping economic growth stable. There is a well-documented proportionality between gross world product and energy. Since energy use has been dominated by carbon based fuels, the same proportionality exists between gross world product and CO_2 emissions [295]. It is recognized that this trend must cease, and this has led to initiatives such as the Kyoto Accord which has set targets for world greenhouse gas reductions.

Thus far, nuclear is the only technology to have been proven to yield large scale reductions in greenhouse gas emissions, and yet it is largely ignored in the emissions debate. Nuclear can reduce emissions in the electricity generation sector, but it cannot directly influence the transportation sector or substantial parts of the industrial sector, which, together, are responsible for almost half the total greenhouse gas emissions. Hydrogen is considered a carbon free technology, which has a direct link especially to the transportation sector. However, hydrogen is presently produced by

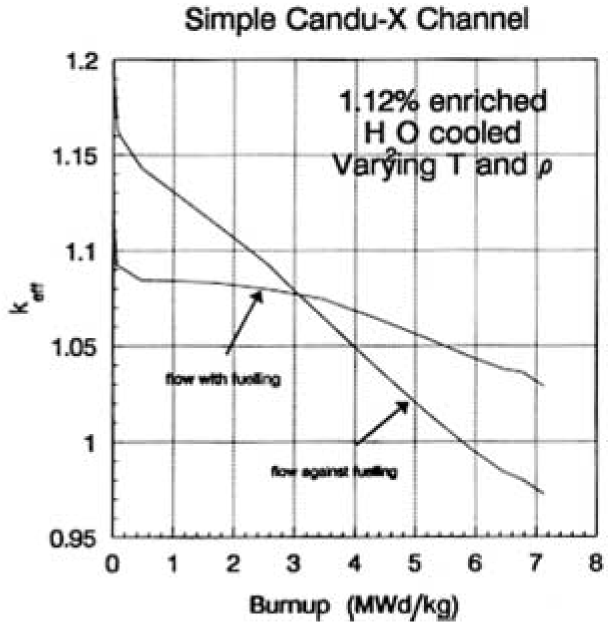


FIG. 221. Effect of fuelling direction on k_{eff} as a function of burnup.

the consumption of carbon based fuels, which leads to CO_2 emissions. The total cycle, therefore, cannot claim significant emission reductions unless the production process becomes carbon free. Indeed, the inefficiencies introduced by the production, distribution and end use of hydrogen may increase overall emissions.

It is, therefore, proposed that an HWR be used to generate electricity for the grid, as well as power an electrolytic cell to produce H_2 by the electrolysis of water. This scheme serves both the electricity *and* transportation sectors without greenhouse gas production. Moreover, heavy water, which can be used in additional HWRs, is produced very economically as a by-product of the electrolysis process.

Projections indicate that a fleet of about 20 CANDU 6 HWRs installed by 2020 would meet all of Canada's estimated needs for electricity, and could reduce CO_2 emissions by a further 50 million t/a compared with the most efficient gas turbine generation. One CANDU 6 plant could also produce enough D_2O to fill a second reactor in about four years, and could power approximately 660 000 H_2 vehicles/d, on the basis of current projections for mileage and H_2 fuel cell efficiencies.

In the long term, as the CMC temperature in HWRs increases and new hydrogen production technologies emerge, other types of cogeneration plant will become feasible.

One example is the use of waste heat from nuclear power plants for seawater desalination. The consumption of fresh water in the world increases with the growing population and rising levels of industrialization, and a shortage in freshwater supply is currently limiting economic growth in parts of the world. Many believe that early in this century, desalination based on energy supplied by nuclear power plants will be essential to the economic development of many regions with little or no freshwater reserves [296].

There are several types of desalination process suitable for coupling with nuclear power plants. Adopting the product evolution approach, only those with minimum impact on the design of the HWR electricity generating plant (EGP) have to date been considered. The first concept is based on the use of a reverse osmosis process. Discharge cooling water from the main condenser of the EGP is used to preheat feedwater for a reverse osmosis plant, which can be operated independently of the EGP. The preheating provided improves the efficiency and, hence, reduces the cost of the freshwater production.

A second desalination option is based on the use of low temperature multieffect distillation (LT-MED). Again, the desalination plant and the EGP are loosely coupled. Preheated sea water (discharge from the condenser) is used as feedwater to the first effect in the LT-MED plant. A small amount of steam from the EGP is passed through the evaporator tubes in the first effect to provide heat for the initial evaporation. The rest of the process is identical to the conventional multieffect distillation process: heat for evaporation in a given effect is provided by the condensation of vapours created in the previous effect.

As the demand for nuclear desalination becomes evident, and as the CMC temperature increases, other desalination technologies may be considered in the future.

Another advantage of the cogeneration plant is that it provides the flexibility for HWRs to operate at a constant power level, independent of grid demand. Excess energy (either thermal or electric) can be used to produce hydrogen or fresh water at off-peak hours. As a result of the constant power level, reactivity control systems currently used for load following can be simplified or eliminated.

8.6. THE INDIAN AHWR

8.6.1. Introduction to the Indian design

The AHWR is a 235 MW(e) heavy water moderated, boiling light water cooled, vertical pressure tube type reactor with its design optimized for utilization of thorium for power generation [297]. The conceptual design and the design feasibility studies for this reactor have been completed and at present the reactor is in the detailed design stage. The reactor design has a number of passive features described in subsequent sections. The overall design philosophy includes achievement of simplification to the maximum extent possible.

The detailed economics of operation of the AHWR have yet to be worked out, pending finalization of plant design. The reactor incorporates several features that simplify the design and eliminate certain systems and components, and which are likely to make the AHWR economically competitive with other available options for power generation. Some important elements in the AHWR design, which have a bearing on its improved economics, are as follows;

- Elimination of high pressure heavy water coolant, thereby leading to reductions in heavy water inventory, heavy water leaks and exposure of personnel to tritium;
- Replacement of complex and long delivery items such as replacement of the steam generator by a steam drum of simple construction;
- Minimizing dependence on active systems such as primary coolant pumps (owing to natural circulation of light water coolant), thus enabling usage of conventional equipment for performing duties that have much less safety importance attached to them;
- Shop fabrication of major components of the reactor, such as coolant channels, to reduce construction cost and time.

8.6.2. Description of the nuclear systems

8.6.2.1. PHTS

The PHTS is shown in Fig. 222. This system is designed to cool fuel assemblies by boiling light water, which flows through the coolant channels by natural circulation.

The steam–water mixture from each coolant channel is fed through 125 mm nominal bore tail pipes to four steam drums, which are located so as to have an elevation difference of 39 m with respect to the inlet feeder (coolant channel bottom). The steam, at a pressure of 70 kg/cm^2, is separated from the steam–water mixture in steam drums. The steam is fed to the turbine by two 400 mm nominal bore pipes. The steam from the turbine is condensed and after purification of the condensate and preheating, is pumped back to the steam drums at a temperature of 165°C. The feedwater is mixed with the water separated from the steam–water mixture at 285°C in the steam drums. The water level in the steam drum is a function of reactor power and is maintained at a set level during power operation.

A nearly uniform exit quality of steam in all the channels is maintained by providing orifices at the bottom of the reactor coolant channels.

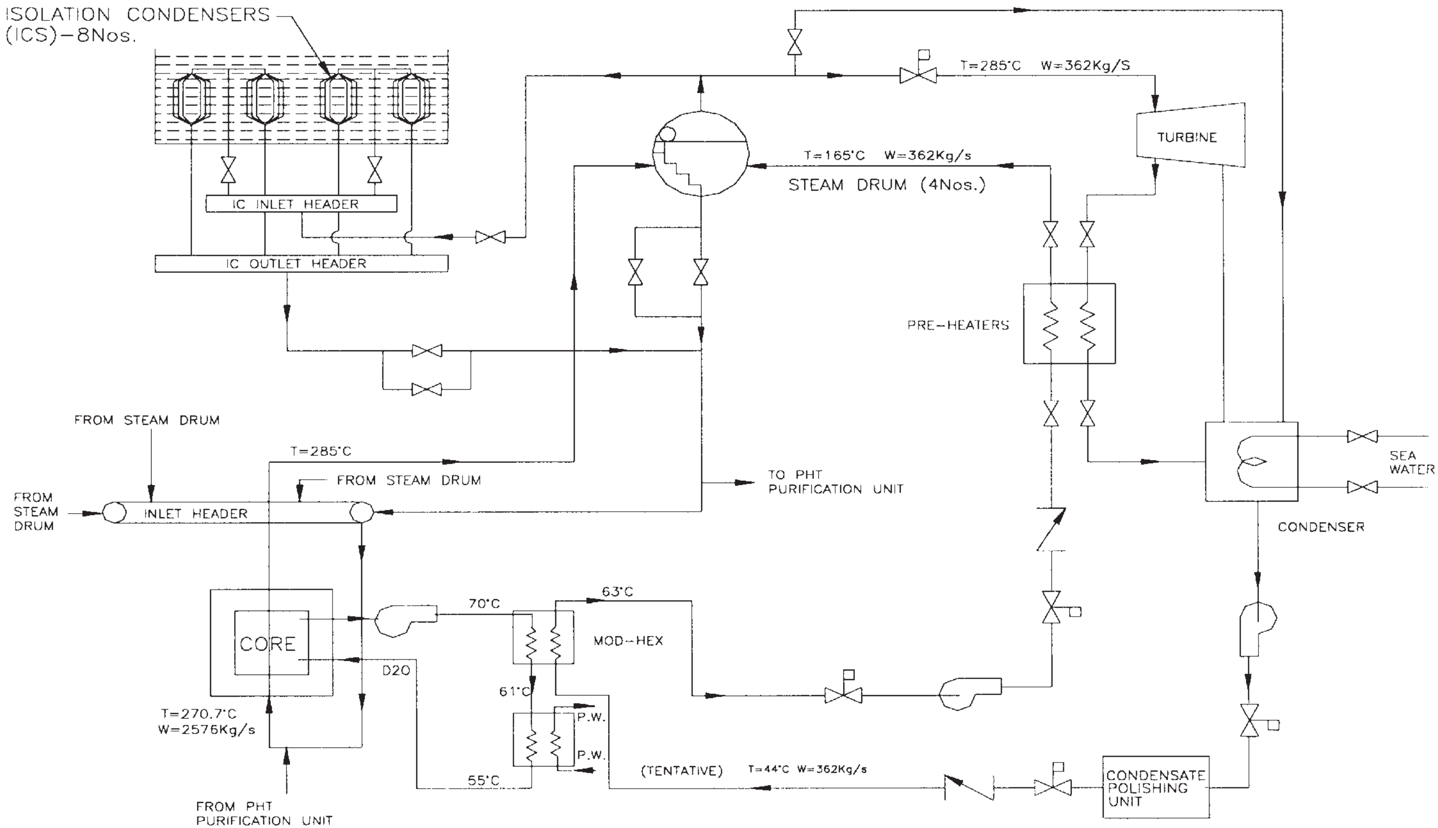


FIG. 222. Simplified PHTS flow sheet.

8.6.2.2. *Core decay heat removal system*

Reactor core decay heat removal through isolation condensers is a passive safety feature for the removal of core decay heat during normal reactor shut down. The system is designed to remove the core decay heat for a period of three days without operator intervention.

The core decay heat removal system is designed to remove heat at 3% of full reactor power, with a steam temperature of 150°C, and has the capability to remove decay heat at 6% of full power for a few seconds duration, when the steam temperature is 285°C. The isolation condensers consist of vertical tubes, joined at both ends to cylindrical headers and submerged in a GDWP. Steam from the coolant channels enters isolation condenser tubes from the top end via the steam drums and is condensed by the surrounding cool water of the GDWP. The condensate returns by gravity to the PHTS through an isolation condenser outlet header.

The system is designed for 4 × 50% capacity. Eight isolation condensers (of which four are operative at any one time) are located in eight compartments of the GDWP. The capacity of the GDWP (for cooling) is based on satisfying the requirement of having a 2 m head of water above the isolation condensers.

8.6.2.3. *Active shutdown cooling system*

An active shutdown cooling system is provided to lower the temperature of the PHTS from 150°C to 60°C during a long shutdown of the reactor for maintenance. The system consists of four loops, of which two are operative at any one time. This system is designed to take care of the non-availability of isolation condensers for removal of the reactor core decay heat.

8.6.2.4. *Moderator system*

The moderator system is designed as a full tank concept for normal operation. Helium is used as a cover gas in the AHWR.

8.6.2.5. *ECCS*

The ECCS is designed to remove the core heat by passive means in the event of a postulated LOCA occurring. In the event of rupture/breakage of the primary coolant pressure boundary, cooling is achieved initially by the action of a large flow of borated light water derived from advanced accumulators. Subsequent cooling of the core is achieved by water stored in the GDWP, the inventory of which is adequate to cool the reactor core for a period of three days without operator intervention.

The ECCS consists of four accumulators with a total capacity of 260 m^3 and a GDWP of 6000 m^3 capacity, both connected to the ECCS header. The ECCS header is connected to individual coolant channels above the tail pipe. The ECCS coolant enters the core through tubes in the centrally located burnable absorber rod in the fuel cluster so as to ensure the wetting of fuel pins by spray action. The coolant, after issuing from a ruptured pipe, accumulates in the reactor cavity along with the PHTS coolant and is recirculated through heat exchangers to ensure long term core cooling.

8.6.2.6. Reactor core and fuel design

(a) Design objectives

The reactor physics parameters are finalized to meet the following important design objectives:

- Power in thorium fuel: 75% (approximately).
- Slightly negative void coefficient of reactivity.
- Discharge burnup greater than 20 000 MW·d/t.
- Initial plutonium inventory and consumption to be as low as possible.
- Self-sustaining in ^{233}U.
- Thermal power: 750 MW.

(b) Fuel cluster design

The reactor core has 452 coolant channels. The fuel cluster consists of 30 $(Th,U^{233})O_2$ and 24 $(Th,Pu)O_2$ pins, termed thoria and MOX fuel pins, respectively. To generate a lower power fraction in MOX fuel, the plutonium content in the MOX is kept low, at 3% (typically), and these pins are located in the outermost ring of the fuel cluster. The maximum channel power is 2.3 MW.

(c) Moderator and reflector

The reactor core is contained in a calandria having a heterogeneous mixture of heavy water as moderator and pyrocarbon material as scatterer. Heavy water is provided as a reflector in the radial direction with thickness of 300 mm. Heavy water also acts as a reflector in the axial direction with thickness of 750 mm at the bottom location and 600 mm at the top location. This arrangement, evolved after detailed analysis of a number of cases, meets the requirements of a satisfactory k_{eff} value and a negative void coefficient of reactivity.

(d) Shutdown systems

The AHWR is provided with two, independent, fast acting shutdown systems (primary and secondary shutdown systems). The primary shutdown system consists of absorber rods having boron carbide as the neutron absorbing material. Boron carbide fills an annulus of thickness 1.5 mm, formed by stainless steel shells. The secondary shutdown system consists of a liquid poison injection system in which borated solution is injected into the radial reflector region.

(e) Reactor fuel design

(i) Design objectives

The fuel assembly of the AHWR is designed to provide:

- Continuous full power operation,
- Low pressure drop of the coolant,
- Stable neutronic/thermohydraulic coupling during all stages of reactor operation,
- On-power fuelling operation,
- Reconstitution of fuel clusters,
- Spray on fuel pins from the ECCS during a LOCA.

(ii) Description

The fuel assembly consists of components such as the fuel cluster and the shield plug. The 4.2 m long fuel cluster is suspended inside the pressure tube of the coolant channel from the top by a hanger assembly and has features that permit the separation of the shield plug from the spent fuel inside the fuelling machine and also the joining of the same shield plug with new fuel.

The fuel pins are arranged in three concentric rings in the cluster. The fuel pin of 11.2 mm outside diameter consists of a Zircaloy clad tube, 0.6 mm thick. In addition to fuel pins, the fuel cluster has a ZrO_2 displacer rod of 38 mm outside diameter containing about 3% dysprosium as a burnable poison and a facility for spraying emergency core cooling water directly on the fuel pins during a LOCA. The bottom and top tie plates are connected through the central absorber rod. The fuel pins rest on the bottom tie plate and are free to expand axially at the top. The interelement spacing between fuel pins is maintained with the help of six Zircaloy spacers.

(f) Fuel handling and transport system: Design philosophy

The AHWR is designed to have on-power fuel handling features to increase the capacity factor of the reactor by maintaining the designed reactivity in the core and by optimizing the fuel burn up. The fuelling frequency is estimated to be six assemblies in a month. The fuelling machine has the following major components:

- A fuelling machine head for handling the fuel clusters by means of ram drives and a snout drive for coupling and making pressure tight joints with the coolant channel;
- A carriage for movement of the machine on rails, laid between reactor block and storage block;
- Shielding (lead, steel and paraffin wax) to limit the surface dose rate to below 0.6 mr/h, so as to make the machine and the reactor top approachable during fuelling operations;
- A cooling system to remove the decay heat from the fuel clusters;
- A control and electrical system for remote operation of the machine from the control room in auto or manual mode.

(g) Primary components

The primary components of reactor pile block structure consist of equipment and components contained in the reactor cavity. These include calandria with vertical coolant channels, end shields at top and bottom, concrete vault filled with light water, tail pipe vault at top and feeder vault at bottom. Figure 223 shows the layout of the AHWR reactor components.

(i) Design bases

The major bases for the design of AHWR pile block components and structures must make provision for:

- Ease of replacement of coolant channels,
- Heterogeneous moderator and reflector system comprising heavy water and pyrocarbon material,
- Features to facilitate in-service inspection and maintenance,
- Ease of erection,
- Adequate shielding to enable accessibility to areas outside the pile block during reactor operation,
- Direct emergency core cooling,
- On-power refuelling.

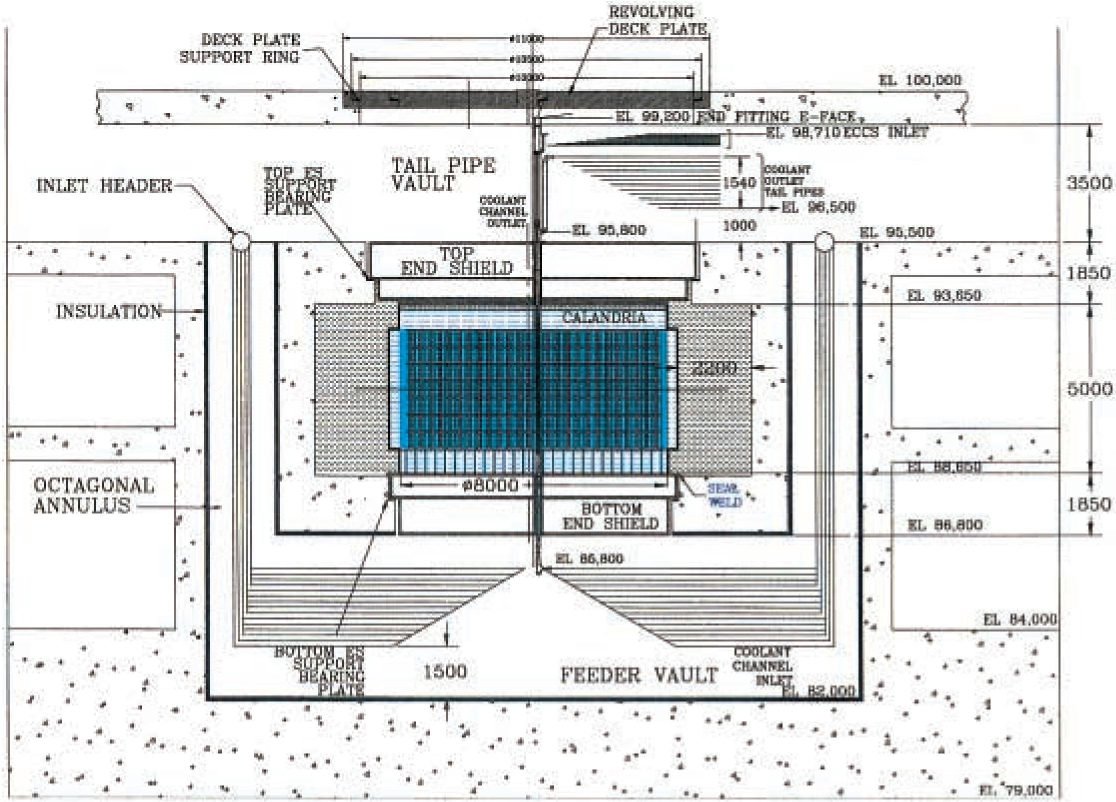


FIG. 223. *The AHWR reactor component layout.*

(ii) Calandria

The calandria is a vertical cylindrical shell structure with a subshell at each end, connected by a flexible annular plate. Both subshells are in situ welded to the shells of the end shields. Vertical calandria tubes are arranged on a square lattice pitch and rolled to the lattice tubes of the end shields. Nozzles and penetrations are provided in the shell and subshell regions of the calandria for the circulation of both heavy water moderator and helium (cover gas). Vertical penetrations are provided for primary and secondary shutdown systems, reactivity mechanisms and in-core neutron monitoring. The calandria is provided with overpressure relief devices to mitigate pressure rise in the event of an accident.

(iii) Coolant channel

The coolant channel accommodates the fuel assembly, maintains thermal insulation between the hot pressure tube and the cold calandria tube, and provides an interface for coupling to the PHTS at both ends. It also facilitates injection of light water directly into the fuel clusters from the ECCS in the event of a LOCA and provides an interface to facilitate on-power fuelling operations.

The design provisions are made to take care of:

- Thermal expansions,
- Creep/growth related dimensional changes,
- Remote replacement of coolant channels.

The coolant channel consists of a pressure tube, with end fittings at the top and bottom ends. The coolant channels are supported on the top end shield. The top end fitting has provision for connecting to an outlet tail pipe and the ECCS injection pipe. It also has suitable features to enable engagement of the fuelling machine. The bottom end fitting is connected to the inlet header, which is located above the core, through an individual feeder pipe. Calandria tubes, concentrically located outside the pressure tubes in the calandria region, are rolled to the lattice tubes of the end shields.

(iv) End shields

End shields are provided at the top and bottom ends of the calandria and are in situ welded to the calandria subshells. The end shields are designed to achieve a dose rate of less than 0.6 mr/h in the tail pipe vault and the feeder vault after one hour of reactor shutdown, to allow access of personnel into these areas. The shielding materials are arranged in different layers such as steel, water, and a mixture of water and carbon steel balls. The top end shield supports coolant channels and other vertical penetrations. The top end shield is provided with a composite tube sheet at the bottom end for the circulation of heavy water, which removes the heat generated in the composite tube sheet of the end shield and calandria tubes in the cover gas region. From thermal stress considerations, both end shields are equipped with a recirculation cooling system using light water so as to maintain a temperature of 55°C. The end shields are supported through bearing plates on the concrete structure of the reactor block.

(v) Deck plate

The deck plate provides shielding above the tail pipe vault to limit the dose rate to less than 0.6 mr/h during full power operation, so as to make the reactor top accessible for on-power fuelling and for other operations. The deck plate serves as a platform for the removal of fuel assemblies and supports the shielding skirt of the fuelling machine. The deck plate consists of inner and outer revolving floors which are supported on special bearings to facilitate alignment to any lattice position by selection of the proper combination of rotation of the revolving floors. The inner revolving floor has a central opening of 750 mm diameter which is normally closed by a shielding plug and a flapper mechanism. During fuelling operations, the shielding plug is removed and the flapper is opened after lowering the shielding skirt of the fuelling machine. The shielding skirt also makes a leaktight joint with the inner revolving floor.

(vi) Reactor vault

The calandria is surrounded by a heavy density concrete vault, filled with light water, to provide thermal and biological shielding against neutrons and gamma rays. The thickness of water shield and concrete are derived from the fact that the dose rate in adjacent rooms is less than 0.6 mr/h during reactor operation and in the annulus after one hour of reactor shut down. The vault cooling system is designed to remove the heat generated in the vault water due to the attenuation of gamma rays and that heat transferred from the calandria. The inlet and outlet piping of the calandria vault is provided with inverted U-bends to prevent draining of the vault in the event of a pipe break/rupture.

8.6.3. Turbine generator plant system

8.6.3.1. Steam and feedwater system

(a) Design requirements

The steam and feedwater system is a closed loop system designed to meet the following design requirements:

- Generation of 99.9% dry steam in steam drums for operation of the turbine;
- Condensation of steam in the condenser which is exhausted from the turbine in operation mode or in bypass mode;
- Purification of full flow of condensate and pump back to steam drums through preheaters and feedwater pumps, which are conventionally available pieces of equipment;
- Function as a heat sink for the reactor under emergency conditions.

(b) Steam drum and steam system

The system has four steam drums, constructed from carbon steel and lined with stainless steel, of overall size 3.6 m (diameter) × 10 m (length). Each steam drum is connected to 113 tail pipes which in turn are connected to the reactor coolant channels and which carry a steam–water mixture. The water level in each steam drum is controlled by a water level regulator using a comparison with the set point level and the flow of feedwater with respect to steam flow rate.

The steam from each steam drum is tapped from the top location by a 300 mm nominal bore pipe. The outlet pipes from two steam drums are connected to a 400 mm nominal bore pipe and two of these pipes (from four steam drums) are connected to the steam chest of the turbine. The pressure relief system (consisting of four safety valves and four relief valves) is installed on 400 mm nominal bore pipelines within the primary containment of the reactor to protect against overpressure in case of rupture of the pipeline.

8.6.4. Instrumentation and control systems

8.6.4.1. Design concepts

The function of the instrumentation and control system is to monitor and control various plant parameters such as neutronic, thermohydraulic and process parameters reliably, using the principles of redundancy, diversity, testability and maintainability. This is achieved by having triplicated channels, using the principle of two out of three logic and fail safe criteria for the safety systems. The system is also provided with a feature for the on-power testing of channels. The instrumentation for the control and protection system is independent and separate. An extensive operator information system is provided with features such as display, alarm, record, retrieval of plant parameters, etc. The details of this system are being worked out.

8.6.4.2. Reactor protection system

The shutdown system designed for fast transients consists of two completely independent and redundant devices. The fast acting primary shutdown system consists of mechanical shut-off rods and a secondary shutdown system to inject liquid poison into the radial reflector.

8.6.4.3. Reactivity control

The reactivity is controlled by the following methods:

- Refuelling to take care of reactivity loss due to fuel depletion,
- Addition of poison (boron) into the moderator to control long term excess positive reactivity,
- Use of adjuster rods for reactivity control and xenon override operations.

8.6.5. Electrical systems

The salient features of the electrical system are as follows:

- Minimum of two, independent off-site power sources of 220 kV for startup through one startup transformer.
- Two independent power supply sources for normal power operation:
  — From grid through startup transformer,
  — From main generator through unit transformer.
- Automatic transfer of station auxiliaries to other source in case of failure of one source.
- Three Class 1E emergency diesel generators, one feeding to each of the two independent bus sections and one on stand-by to either of the bus sections to provide on-site stand-by power for Class 1E equipment.
- Three independent Class 1E 240 V AC systems with a stand-by and automatic switching and battery backup for the reactor protection channel.
- Three independent 2 × 100% 48 V DC systems with battery backup for the reactor protection channel.
- AC voltage levels of 6.6 kV and 415 V.
- DC voltage levels of 220 V and 48 V.

8.6.6. Safety concepts

8.6.6.1. Safety requirements and design philosophy

Prevention of accidents is the basic design philosophy behind the AHWR. All proven measures of current safety concepts ensuring reliable operation are incorporated in the design in order to prevent accidents. These include use of the following:

- Systems and components designed with conservative margins,
- Redundancy concept for operating systems to increase their reliability,
- Preventive maintenance,
- In-service inspection,
- Large water reservoir in the GDWP,
- Passive safety features,
- Negative void coefficient of reactivity.

8.6.6.2. Safety systems

(a) Passive safety features

The AHWR is being designed to incorporate many passive systems/elements in order to facilitate the fulfilment of safety functions, e.g. reactor operation, reactor shutdown, residual heat removal, emergency core cooling and confinement of radioactivity. As regards removal of heat from the reactor core under operating as well as accident conditions, the heat removal paths and systems are shown in Fig. 224. These systems are described in the following sections.

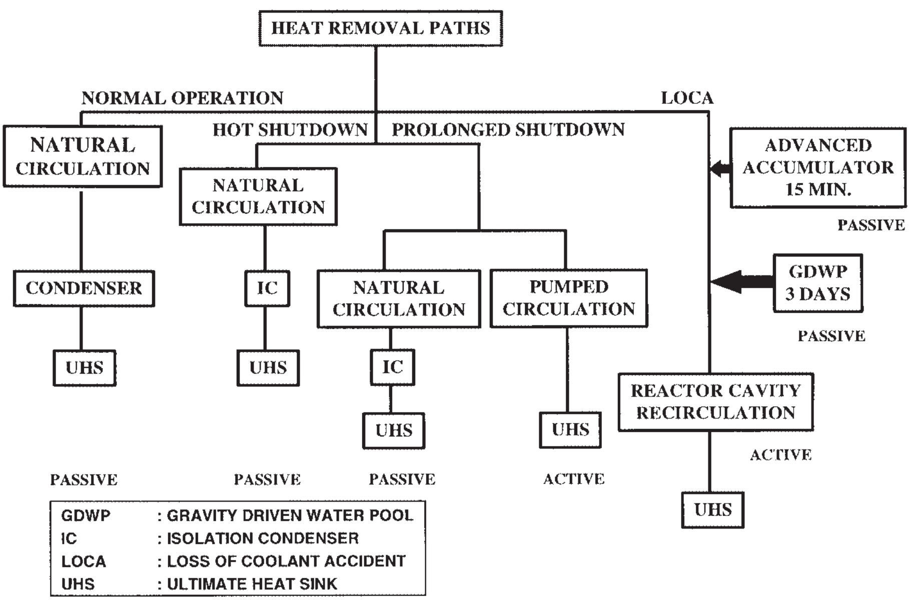


FIG. 224. Heat removal paths in the AHWR.

(b) Natural circulation of primary coolant

During normal reactor operation, full reactor power is removed by natural circulation caused by the thermosyphoning phenomenon. Primary circulation pumps are eliminated and the necessary flow rate is achieved by locating the steam drums at a suitable height above the centre of the core, taking advantage of the reactor building height. By eliminating nuclear grade primary circulation pumps, their prime movers, associated valves, instrumentation, the power supply and control system, the plant is made simpler, less expensive and easier to maintain when compared with options involving forced circulation in the primary coolant circuit. The above factors also lead to considerable enhancement of system safety and reliability, since pump related transients have been removed.

(c) Core decay heat removal

During normal reactor shutdown, core decay heat is removed by isolation condensers which are submerged in the GDWP, located above the steam drum. The steam, fed to the isolation condensers by means of natural circulation, condenses inside the isolation condenser pipes and heats up the surrounding pool water. The condensate returns by gravity to the core. The water inventory in the GDWP is adequate to cool the core for more than three days without operator intervention and without the water boiling. A GDWP cooling system is also provided, as is an active shutdown cooling system for the removal of core decay heat in case the isolation condensers are not available.

(d) Shutdown systems

Two completely independent and redundant fast acting devices (mechanical shut-off rods and liquid poison injection) are provided to shut down the reactor. These devices are actuated by active systems. In case of the failure of these devices to act, the reactor will be shut down as a result of negative void coefficient of reactivity.

(e) Emergency core cooling

During a LOCA, emergency coolant injection is provided by passive means to keep the core flooded and thereby prevent overheating of the fuel. The ECCS is designed to fulfil the following two objectives:

- To provide a large amount of cold borated water directly into the core at an early stage of the LOCA and then a relatively small amount of cold borated water for a longer time to quench the core. This objective is achieved through the ECCS accumulator.

- To provide water, through the GDWP, to cool the core for more than three days.

Long term core cooling is achieved by active means by pumping water from the reactor cavity to the core through heat exchangers.

(f) Core submergence

After a LOCA, water from the PHTS, advanced accumulators and the GDWP, after cooling the core, will be guided and collected in the space around the core known as the reactor cavity. Thus, the core will be submerged under water. If the GDWP fails during any postulated scenario, its inventory will collect in the reactor cavity and provide a heat sink to the core.

(g) Failure of ECCS during a LOCA

The reactor core of the AHWR contains a huge inventory of heavy water moderator, as well as the surrounding vault water. Although the possibility of failure of the ECCS is very small, if for any reason the ECCS is not available during a LOCA, the fuel temperature will rise and ballooning of the pressure tubes will occur. As a result of ballooning, the pressure tubes will come into contact with the calandria tubes and heat will be transferred to the moderator, and from the moderator to the vault water.

(h) Passive containment isolation

To protect the population at large from exposure to radioactivity, the containment must be isolated following an accident. To achieve this, passive containment isolation, in addition to the closure of the normal inlet and outlet ventilation dampers, has been provided for in the AHWR. The reactor building air supply and exhaust ducts are shaped in the form of U-bends of sufficient height. In the event of a LOCA, the containment becomes pressurized. This pressure acts on the GDWP inventory and pours water, by swift establishment of a siphon, into the ventilation duct U-bends. Water in the U-bends acts as a seal between the containment and the external environment, providing the necessary isolation between the two. Drain connections provided to the U-bends permit the re-establishment of containment ventilation manually, when desired.

(i) Passive containment cooling

Passive containment coolers are utilized to achieve post-accident primary containment cooling in a passive manner, and to limit the post-accident primary containment pressure. A set of passive containment coolers are located below the GDWP and are connected to the GDWP inventory. During a LOCA, the mixture of hot air and steam is directed to flow over them. Steam condenses and hot air cools down at the passive containment cooler tube surface, which provides long term containment cooling after the accident.

8.6.6.3. Severe accidents

The primary objective followed in the development of the AHWR is the enhancement of the level of safety to such an extent that the probability of a severe accident occurring becomes negligible, on account of the presence of the safety features already described. This will be confirmed by a PSA. In this context, it may be noted that the core submergence, discussed earlier, and the presence of a large pool of water below the reactor will, following an accident, serve as effective barriers to the escalation of any severe accident.

8.6.7. Plant layout

8.6.7.1. General arrangement of the reactor building

The reactor building of the AHWR is a cylindrical concrete structure consisting of two coaxial cylindrical shells closed at the top by dome structures. The inner structure, termed the primary containment, accommodates high enthalpy systems such as the reactor core, primary coolant systems, fuelling machine, etc. The primary containment has an internal diameter of 50 m and an internal height of 66 m and is constructed from prestressed concrete. The GDWP is located near the top of the primary containment and is designed to perform several passive safety functions. The outer structure, known as the secondary containment, has a diameter of 64 m and a height of 75.5 m and is constructed from reinforced concrete. Both structures are supported on a concrete raft. The AHWR reactor building elevation is shown in Fig. 225.

8.6.7.2. Criteria for design of layout

The layout of the reactor building is designed to:

- Minimize the primary containment volume;
- Provide effective utilization of space in the annulus between the primary and secondary containments;
- Provide unrestricted entry to the reactor top for on-power fuel handling and transfer operations;

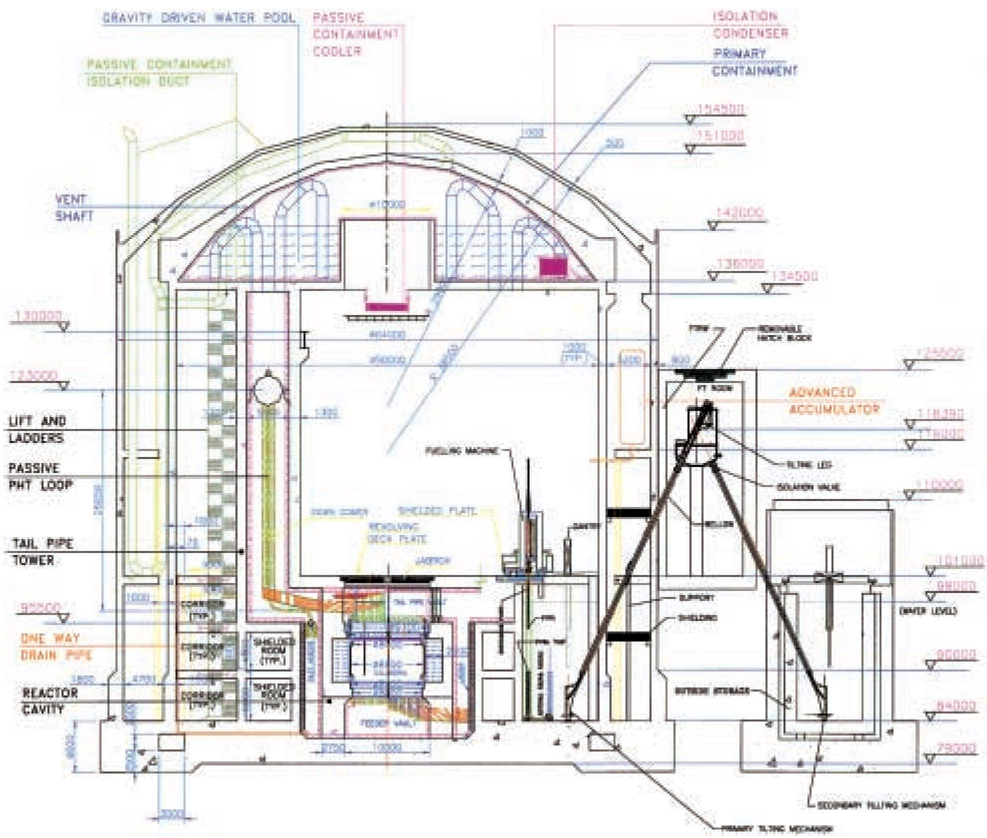


FIG. 225. AHWR reactor building elevation.

- Provide adequate shielding against radiation and prevent the spread of radioactive contamination during normal and accident conditions;
- Provide a large water inventory at a suitable height, capable of supporting a number of passive systems;
- Provide for submergence of the reactor core under water before exhaustion of the ECCS inventory;
- Provide ease of access to the maximum number of pieces of equipment for O&M during normal and accident situations;
- Provide for fire prevention and control.

8.6.8. Project status and planned schedule

The conceptual design of the AHWR was completed in December 1997. On the basis of first level analytical studies and experimental work, the feasibility of the design concept was established and a feasibility report issued.

Detailed design of the AHWR's nuclear systems is in progress. It is planned to develop design details for nuclear systems, conduct supportive analysis and experimental development, prepare detailed specifications for non-nuclear systems and issue a detailed project report in 2002.

8.7. THE HWR 1000 ULTIMATE SAFE GAS COOLED REACTOR

8.7.1. Introduction

A reactor design concept for an ultimate safe reactor has been developed in the Russian Federation under the direction of ITEP. The prototype used for this conceptual design was the KS150 reactor at Bohunice.

8.7.2. Key design concepts

The key design concepts incorporated in the reactor are as follows:

- The entire primary system, including main gas circulators, steam generators and intermediate heat exchangers, is contained within a multicavity, prestressed concrete vessel which retains the primary coolant pressure.
- Low temperature heavy water is used as the moderator.
- Gaseous coolant, either CO_2 or a mixture of CO_2 and helium, is used.
- Low fissile content fuel is used.

8.7.3. Design description

This design uses a prestressed concrete vessel to retain coolant pressure (Fig. 226). Since the concrete pressure vessel retains the pressure, the channel tubes are not significantly loaded and are thus thin walled, large diameter (10 cm) components used for reducing parasitic neutron absorption.

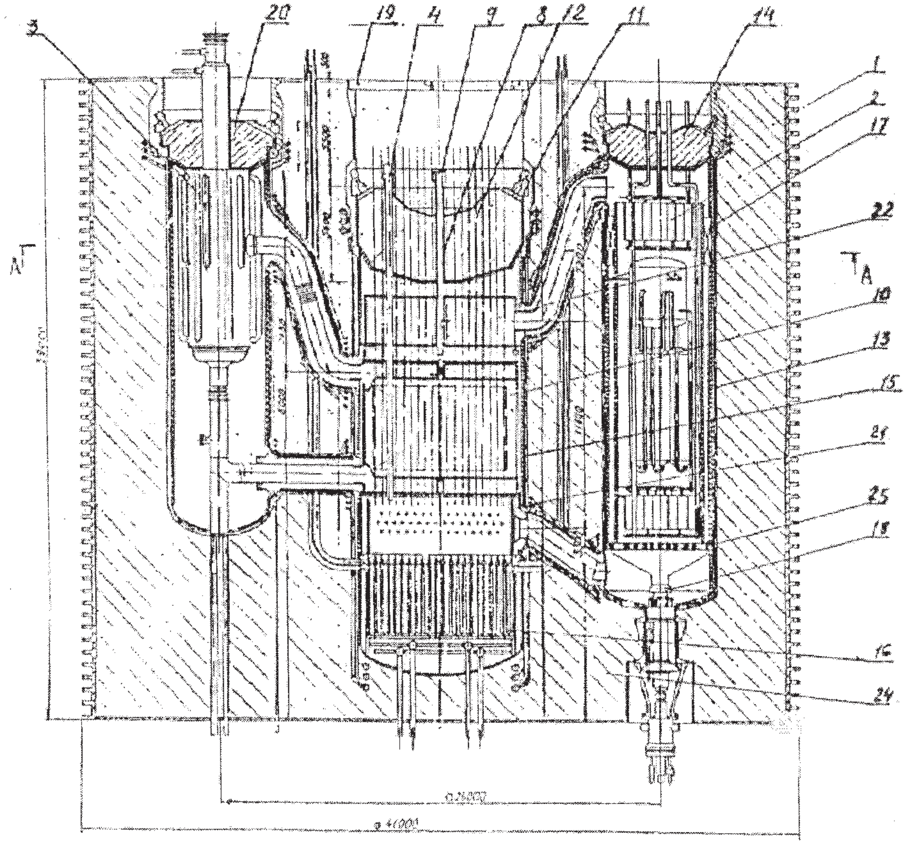
The gas cooling has some advantage in the case of a small break LOCA where leakage of gas does not involve a phase change, as happens when water turns to steam, and does not result in a large pressure rise. Also, the large fuel channel radius produces a high probability of new fission occurring and forms the basis for effective fissile isotope utilization.

The reactor and the main equipment of the gas and heavy water loops are located within a multicavity prestressed concrete vessel having a stainless steel liner, and core debris catching and cooling systems, as well as a system for collecting and returning any gaseous coolant leaks through the prestressed concrete vessel (see Fig. 227). The prestressed concrete vessel, as well as the gas and heavy water loops and auxiliary equipment, are housed within a leaktight steel shell (primary envelope) which has an excess design pressure of ~350 kPa and which is capable of retaining all primary coolant. In turn, the primary envelope, together with reactor rooms, are surrounded by another steel shell (secondary envelope) which has an excess design pressure of ~40 kPa and which is also capable of retaining all primary coolant.

The stainless steel reactor vessel, located in the central prestressed concrete vessel cavity, houses 362 calandria tubes in a vertical orientation, arranged in a triangular lattice with a pitch of 41 cm. Of this total, 341 tubes, with an internal diameter of 20 cm, surround fuel channel tubes and provide a gas filled annular insulating gap between them. The fuel channels contain 126 metallic fuel rods 0.6 cm in diameter, coated in a zirconium alloy and cooled with a gaseous coolant (either CO_2 or helium in different reactor versions). The remaining calandria tubes are reserved for control devices. The latter have a tube within a tube configuration, with the inner, low absorbing tube used for regulating and normal load following, and the outer high absorbing tube used for reactor shutdown and compensation. The main core characteristics are as follows: core diameter – 8.20 m; core height – 5.00 m; radial reflector thickness – 0.60 m; axial upper and lower reflector thickness – 0.40 m; core thermal power – 3200 MW, including power in fuel (3000 MW), and in moderator (200 MW). The other key core neutronics characteristics are presented in Table LIX.

The six steam generators, together with the gas circulators and the two heavy water moderator intermediate heat exchangers, are located in the eight vessel cavities provided on the prestressed concrete vessel's periphery.

Npp TR-1000 US in the vessel of
prestressed reinforced concrete.



Longitudinal Section of reactor

1 - vessel, 2 - steamgenerator, 3 - cooler of moderator,
4 - technological channel, 8 - channel of emergency system,
10 - tank of heavy water, 11 - main sealing, 12 - upper cover,
13 - internal heat isolation, 14 - cover of steamgenerator hole,
15 - liner of internal vessel, 16 - trap, 17 - exchanger for removing of decay heat, 18 - gas circulator, 19 - turning circle,
20 - cover of moderator cooling hole, 21 - inlet chambers,
22 - exit chamber, 23 - check valve

FIG. 226. Longitudinal section of the HWR 1000 ultimate safe reactor.

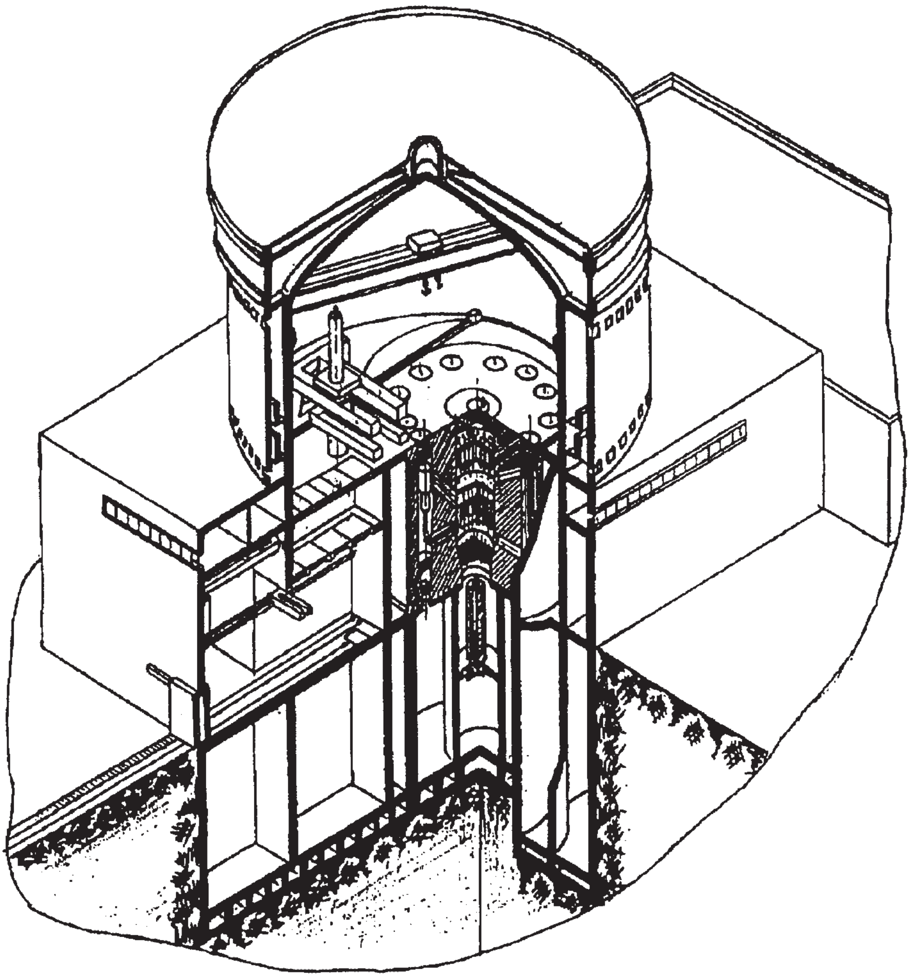


FIG. 227. Layout of the HWR 1000 ultimate safe reactor.

8.7.4. Safety features

The safety features employed in this ultimate safe reactor concept are as follows:

- The prompt neutron power excursion is eliminated.
- Accidental withdrawal of all control rods in the core during operation adds a relatively small amount of reactivity to the system and is compensated for by the negative reactor power coefficient.
- Core debris reconfiguration is eliminated.

TABLE LIX. THE HWR 1000 ULTIMATE SAFE REACTOR: KEY PHYSICS PARAMETERS IN THE ONCE THROUGH NATURAL URANIUM CYCLE

Parameter	Value
Fuel inventory (t natural U)	160
Fuel specific heating rate (MW/t)	20
Fuel irradiation time (full days)	468
Fuel reloading time (full days)	78
Fuel mean thermal neutron flux ($n \cdot cm^{-2} \cdot s^{-1}$)	0.5×10^{14}
Core radial power peaking factor:	
Start of cycle	1258
End of cycle	1232
Mean fuel burnup (MW·d/t)	9500
Breeding ratio	0.800
Equilibrium fuel feed at load factor of 0.80 (t/a)	100
Feed fuel U-235 concentration (kg/t)	7.10
Discharge fuel concentration (kg/t):	
U-235	1.75
Pu-239	2.70
Pu-240	1.32
Pu-241	0.30
Pu-242	0.17
Neutron balance, absorptions (fissions):	
Fuel cladding	59
U-235	2215 (1845)
U-238	3685 (252)
U-236	14
Fuel admixtures	18
Pu-239	2326 (1600)
Pu-240	332 (2)
Pu-241	221 (166)
Pu-242	3
Saturated fission products	347
Other fission products	374
Fuel channel tubes	76
Calandria tubes	178
Moderator	152
Total absorptions	10 000
Total fissions	3865
Temperature reactivity coefficients 1/Ê:	
Fuel	-1.3×10^{-5}
Fuel cladding	$+0.2 \times 10^{-5}$
Coolant	$+0.14 \times 10^{-5}$
Moderator:	
Start of cycle	-1.0×10^{-5}
End of cycle	-1.5×10^{-5}

Pressure vessel brittle failure is not considered possible because such a vessel under increasing pressure would crack and leak, and therefore reduce the pressure.

8.7.5. Key physics parameters

The key physics parameters in a once through natural uranium cycle are listed in Table LIX. The calculated mean fuel burnup is ~10 000 MW·d/t. Other fuel cycles, including a uranium–plutonium fuel cycle, are possible.

8.8. THE NEXT GENERATION OF CANDU

8.8.1. Introduction

AECL has established a successful line of pressurized HWRs internationally, in particular the medium sized CANDU 6 reactor design. Building on this experience and expertise, AECL is continuing to adapt this basic design to develop the next generation (NG) of medium sized reactor to meet the needs of both traditional and emerging energy markets. The NG CANDU design features major improvements in economics, inherent safety characteristics and performance, while retaining the proven benefits of the earlier family of PHWRs.

This section summarizes the plant's main features, including the major systems and key components associated with the nuclear steam plant, the enhanced safety - features and the balance of plant. It also identifies the advanced design and construction methods that are being implemented in the overall design programme.

8.8.2. Background

The development work for the NG CANDU plant has three clear thrusts:

- Enhanced economics — lower capital and O&M costs, coupled with improved performance and reliability;
- Enhanced safety — via improved engineered and passive safety characteristics;
- Enhanced sustainability — use and conservation of resources, protection of the environment, reduction of wastes and emissions.

The plant described in this section has a gross electrical output of the order of 650 MW(e), with a net output of approximately 610 MW(e). This unit size has been selected to match the requirements, in increasingly deregulated electrical power markets, for plants with lower plant capital and operating costs, plus reduced project schedules, through the use of improved design and construction methods and operational improvements. However, the basic concept described is suitable for a range of plant sizes with gross outputs in the 400–1200 MW(e) range.

The reference plant design is suitable for standard cooling with ambient sea or lake water heat sinks or with cooling towers, where a suitable body of cooling water is not available. A pictorial view of the proposed two unit plant arrangement is shown in Fig. 228.

8.8.3. High level requirements

Over the past two decades, international electric utility and atomic energy organizations have periodically reviewed and updated their requirements for advanced nuclear reactor designs to ensure that they meet customer needs, including the need for public acceptance. The requirements arising from those activities have been described in reports such as the Electric Power Research Institute's Advanced Light Water Reactor Requirements document, the European Utility Requirements document and IAEA Safety Reports. This section summarizes the high level requirements addressed by the NG CANDU design.

FIG. 228. Pictorial view of two unit NG CANDU plant arrangement.

8.8.3.1. Generation IV nuclear power system requirements

The Generation IV international initiative addresses the development of technologies for nuclear power. The broad objective of this initiative is the development of technologies that can satisfy the preconditions for expansion of nuclear energy systems throughout the world.

The fundamental issues that need to be addressed by advanced nuclear power systems are safety, economics and sustainability. The NG CANDU design addresses all of these utility requirements and represents the application of Generation IV principles to a product directed towards the near term generation market.

8.8.3.2. Safety

The NG CANDU design will have an increased margin of safety. Taking maximum advantage of both engineered and passive safety features, the likelihood of a severe accident occurring will be low, and the potential for off-site releases of radioactive material will be sufficiently low that a target of ‘no evacuation’ can be achieved.

8.8.3.3. Economics

A strong driving force in the evolution of the NG CANDU design is economics. All of the plant systems, components and equipment have been adapted from existing proven systems, with refinements made to reduce costs while retaining high performance and reliability features. The result is a power system that is competitive with all other electricity generation systems of comparable size, including oil, coal, natural gas and other nuclear power systems.

8.8.3.4. Reliable operations

The operational characteristics of the plant have been improved by taking into consideration feedback from the operating plants, and the plant life management and plant life extension programmes, and by adopting specific design changes to reduce maintenance requirements and improve the overall performance. The design will incorporate ‘smart’ CANDU operating support technologies to reduce the O&M costs and improve plant monitoring capabilities.

8.8.3.5. Environmental acceptance

In an age of increased environmental sensitivity and concern about widespread human impacts such as global warming, the public expects large scale investments to meet the test of sustainability. This means sustainability in the use and conservation of resources, in the protection of the environment, and in the reduction of wastes. All nuclear power systems rank extremely high in terms of environmental sustainability since they use very small quantities of mined uranium fuel to generate electricity. The present CANDU power systems generate more electricity per quantity of mined uranium than other reactor types. The NG CANDU offers even greater efficiency, through the use of SEU fuel and higher HTS temperature and pressure.

8.8.3.6. Emissions

Plant emissions will be reduced through an improved steel lined, prestressed concrete containment design. Hazardous wastes, both radioactive and inactive, will be collected within the plant, treated, and stored for final disposal. Production of tritium will be reduced by the use of light water as the coolant in the HTS. The use of SEU fuel will reduce the volume of high level spent fuel waste produced, and reduce the demand for temporary wet storage and long term dry storage facilities at the site.

8.8.4. Design objectives

The overall design objectives considered in the development of the NG CANDU concept are to:

- Retain the basic proven features of the CANDU reactor, while optimizing the design by utilizing SEU fuel to reduce the reactor core size, which reduces the amount of heavy water moderator required and which eliminates the need to use heavy water as the reactor coolant. As a result, both the heavy water cost and the number of heavy water management systems can be reduced substantially.
- Meet the specific capital cost, construction schedule and operating cost targets for a 600 MW(e) class plant competitive with natural gas and coal fired plants, and other types of nuclear power plant.
- Improve on the traditional CANDU advantages through enhanced safety margins, lower emissions, reduced radiation exposure, higher capacity factors, improved construction methods and maintenance practices, and lower operating costs.
- Improve the overall cycle efficiency through the use of higher pressure and temperature conditions in the reactor coolant and steam turbine systems.
- Standardize the nuclear steam plant and balance of plant designs such that they are suitable for a variety of sites with minimum changes to the reference design and documentation.

- Accommodate the division of plant structures and systems to facilitate a variety of financing arrangements, contractual arrangements, or partnerships with one or more organizations.
- Employ state of the art technologies including design, modular construction and project management technologies developed by AECL, consistent with construction of the first unit in the 2005–2010 period.
- Reduce component cost, maximize component life, minimize component installation time and provide a means of component replacement at the end of component life that is short and simple, thereby minimizing radiation exposure and replacement costs.
- Ensure that the plant can achieve the target capacity factor of 90% for the 40-year design life, including a major mid-life refurbishment.
- Consider the feedback from the existing CANDU plant life management and plant life extension programmes to ensure that the 40-year plant design life is achieved and determine whether it can be further extended to a 50-year plant design life.
- Include human factors considerations in the design of systems, facilities, equipment and procedures, and in all interfaces with plant personnel.
- Reduce the O&M personnel requirements by simplification of plant design, and standardization of equipment and maintenance procedures utilized.
- Implement enhanced 'smart' CANDU information systems for improved control and monitoring of plant performance.
- Ensure that the NG CANDU continues to make efficient use of uranium resources and maintains the CANDU advantage of fuel cycle flexibility.
- Ensure, through adjustments to the number of fuel channels within the reactor core and appropriate modifications to system designs, that the basic reactor design is suitable for smaller or larger plant designs, in the 400–1200 MW(e) range.

8.8.5. Basis of the design

AECL has adopted the evolutionary approach for this development programme, accommodating significant changes to design while retaining traditional CANDU strengths. The design incorporates the following features:

- Modular horizontal fuel channel core design;
- Available, simple, economical fuel bundle design;
- On-power refuelling;
- Separate, cool, low pressure moderator with backup heat sink capability;
- Relatively low neutron absorption for good fuel utilization.

A number of enabling technologies have been developed at the component level, many of which have been integrated into the basic design. For example, a bore seal closure and an improved fuelling machine design make the lattice pitch reduction and consequent improved reactor physics characteristics possible. At the same time, the design is firmly rooted in the principles and characteristics of the existing CANDU, and takes full benefit from the extensive knowledge base of CANDU technology built up over several decades of operation. The following key features are incorporated into the new design concept:

- SEU fuel contained in CANFLEX bundles;
- Light water replacing heavy water as the primary coolant;
- More compact core design with reduced lattice pitch, reduced heavy water inventory and highly stable core neutronic behaviour (see Fig. 229);
- Enhanced safety margins;
- Higher coolant system and steam supply pressure and temperature, resulting in improved overall thermal efficiency;
- Reduced emissions;
- Improved performance through advanced O&M information systems.

These advancements, along with improvements to project engineering, manufacturing and construction technologies, allow significant reduction in both capital cost and construction schedule while enhancing the inherent safety of the basic CANDU design. Figure 230 shows an overall flow diagram of the NG CANDU plant.

8.8.6. Unit output

The gross electrical output of the generator is 650 MW(e). The estimated unit service power is about 40 MW(e), yielding a net unit electrical output of approximately 610 MW(e). The modular design of the NG CANDU allows for the plant output to range from 400 MW(e) to 1200 MW(e) with minimal impact on the overall plant design characteristics.

8.8.7. Technical data

The key technical data for a single unit NG CANDU plant are presented in Table LX.

8.8.8. Plant design

The basis of the NG CANDU plant design concept is described in Section 8.8.4. The design approach involved developing an envelope of site parame-

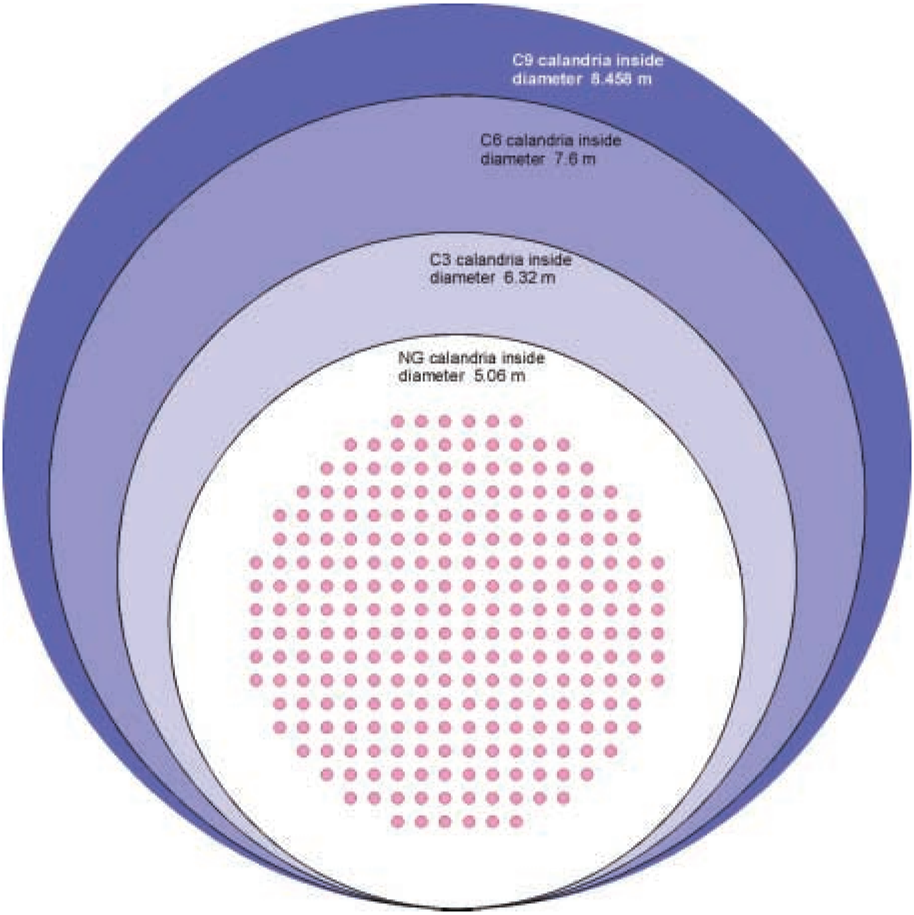


FIG. 229. NG CANDU reactor size versus other CANDU reactors.

ters that permits the plant to be located on a large number of potential sites without requiring significant design or documentation changes. For example, sufficient space is provided in the building layout to accommodate the larger pumps and heat exchangers required by a site with high water temperatures and/or a 50 Hz power grid. The design of relevant buildings and structures will meet internationally accepted seismic requirements for a nuclear facility.

8.8.8.1. Cooling water

The recirculated cooling water system is used for all nuclear steam plant cooling requirements to accommodate saltwater or freshwater sites. The system's heat

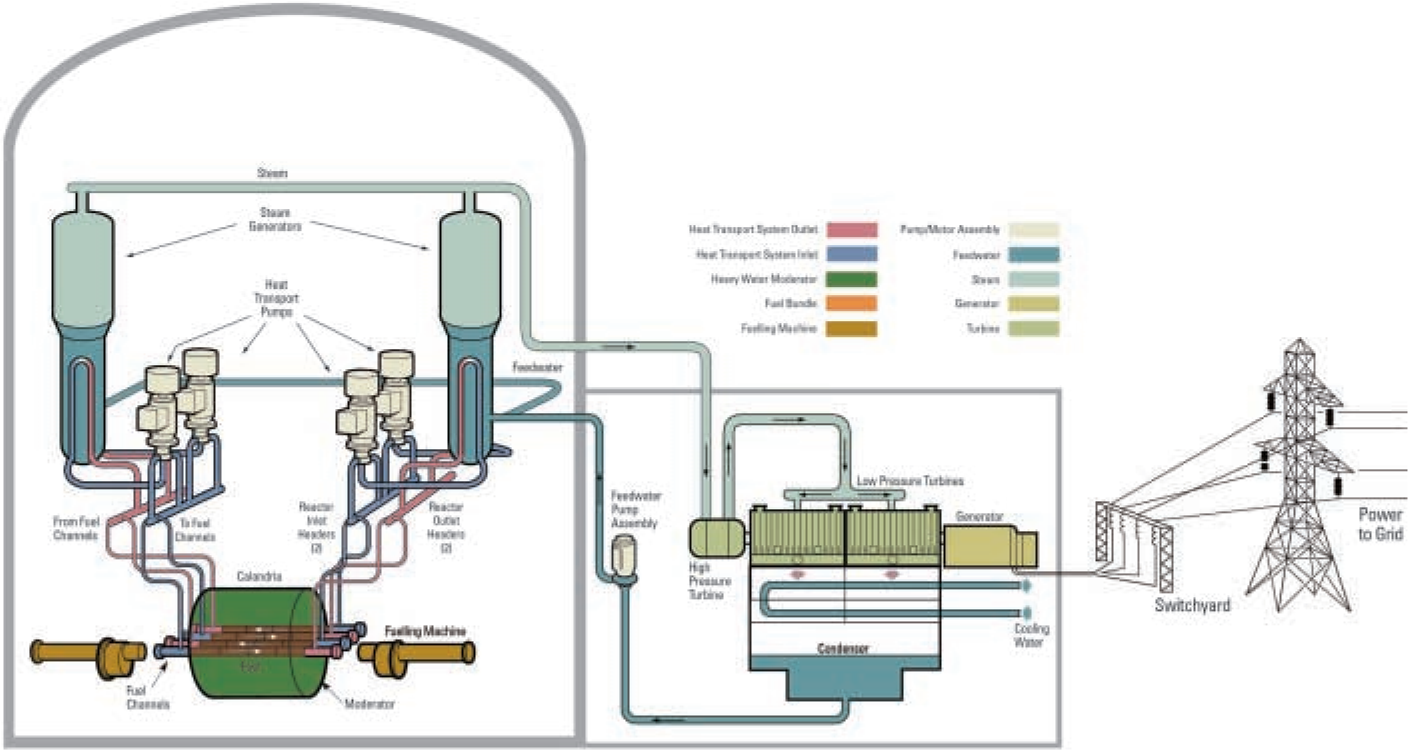


FIG. 230. Overall NG CANDU plant flow diagram.

TABLE LX. NG CANDU UNIT DATA

Parameter	Value
Reactor:	
Type	PHWR
Thermal output (MW(th))	1792
Coolant	Pressurized light water
Moderator	Heavy water
Core diameter (m)	5.1
Fuel channel	Horizontal Zr–2.5%Nb pressure tubes with type 403 stainless steel end fittings
Number of fuel channels	256
Lattice pitch (mm)	220 (square)
Reflector thickness (mm)	550
Fuel:	
Form	Compacted and sintered, slightly enriched UO_2 pellets
Enrichment level (wt% U-235)	1.65
Fuel burnup (MW·d/t U)	20 000
Fuel bundle assembly	43 element CANFLEX
Length of bundle (mm)	495.3
Outside diameter (maximum) (mm)	103
Bundle weight (kg)	23.1 (includes 17.8 kg U)
Bundles per fuel channel	12
HTS:	
Reactor outlet header pressure (MPa (abs))	13.0
Reactor outlet header temperature (ºC)	331
Reactor inlet header pressure (MPa (abs))	14.2
Reactor inlet header temperature (ºC)	286
Reactor core coolant flow (total) (Mg/s)	6.2
Single channel flow (maximum) (kg/s)	26.0
Steam generators:	
Number	2
Type	Vertical U-tube with integral steam drums
Steam temperature (nominal) (ºC)	286
Steam quality (minimum) (%)	99.9
Steam pressure (MPa (abs))	7.0
Heat transport pumps:	
Number	4
Pump type	Vertical, centrifugal, single suction, double discharge
Motor type	AC, vertical, squirrel cage induction
Rated flow (L/s)	2100
Motor rating (MW(e))	6.1

TABLE LX. NG CANDU UNIT DATA (cont.)

Parameter	Value
Containment:	
Type	Steel lined, prestressed concrete reactor structure
Inside diameter (m)	~37
Height (m)	~53
Turbine generator:	
Steam turbine type	Impulse type, tandem compound double exhaust flow, reheat condensing turbine with a last stage blade length of 1.32 m
Steam turbine composition	One single flow, high pressure cylinder and one double flow, low pressure cylinder
Net heat to turbine (MW(th))	1790
Gross*/net electrical output (nominal) (MW(e))	650/610

* Gross electrical output is dependent on cooling water temperature and on the turbine generator and condenser designs.

exchangers are cooled by the raw service water system. Once through, raw water cooling is utilized in the turbine condenser.

In the case of inland sites, where insufficient cooling water is available, the plant will be designed to operate using conventional cooling towers.

8.8.8.2. Tornado protection

Since the frequency and intensity of tornados varies widely around the world, the tornado criterion used for design will be based on site evaluation. The NG CANDU layout and structures are being designed with additional modifications available to accommodate various levels of tornado protection.

8.8.8.3. DBE

The DBE used in the design of the safety related structures and systems of the NG CANDU plant are:

- DBE: Peak horizontal acceleration is taken to be 0.3g.
- SDE: Peak horizontal acceleration is taken to be 0.15g.
- Vertical acceleration is taken as two thirds of the horizontal acceleration.

In the case of non-safety related structures and systems, a design earthquake level will be used for design, consistent with the provisions of the applicable building code for the design of all buildings, systems and equipment supports not covered under safety related structures and systems.

The plant is designed to earthquake levels suitable for sites which experience medium seismic activity, with additional modifications readily available to accommodate sites of high seismic activity.

8.8.8.4. Exclusion area

A factor in restricting the radiation exposure of members of the public to within allowable limits is the provision of an exclusion area from which all unauthorized persons are excluded and within which habitation is not permitted. As a result of the improved steel lined, prestressed concrete containment design, the NG CANDU plant will have a reduced exclusion area radius of 500 m, measured from the centre of the reactor building.

8.8.8.5. Other data

Geotechnical design parameters, such as stratigraphy, foundation medium properties and groundwater levels, along with the demographic and meteorological data for a particular site, will be considered in the design and safety assessments of the NG CANDU plant.

8.8.9. Plant layout

The principal structures of each NG CANDU unit, as shown in Fig. 231, are:

- Reactor building,
- Reactor auxiliary building,
- Turbine building,
- Services building,
- Maintenance building.

In addition, there are auxiliary structures, including:

- Group 1/Group 2 pump house and/or cooling towers,
- Main switchyard,
- Administration building,
- Water treatment facility.

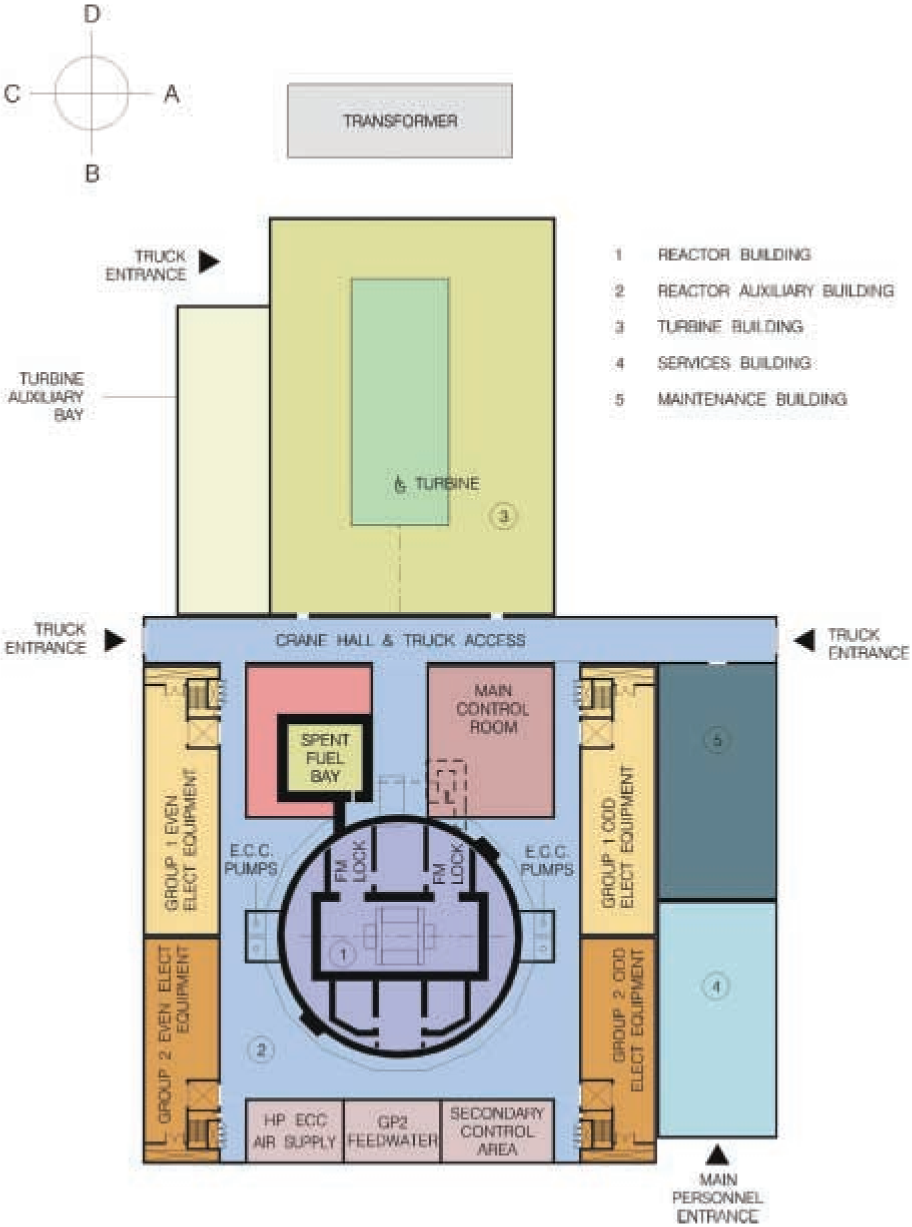


FIG. 231. NG CANDU single unit plant layout.

The principal structures of the NG CANDU are, to the maximum extent possible, self-contained units having the minimum number of connections to the other structures.

The NG CANDU layout is designed to minimize the ‘footprint’ and achieve the shortest practical construction schedule by: simplifying, minimizing and localizing interfaces; allowing parallel fabrication of modules/assemblies and civil construction; reducing construction congestion; providing access to all areas; providing flexible equipment installation sequences and reducing material handling requirements.

A typical two unit plant layout is shown in Fig. 232. The services building in a two unit plant arrangement is located between the units in order to optimize and integrate the common services and thus helps further reduce capital and operating costs. The services and maintenance facilities provided in the services and maintenance buildings include heavy water management, central stores, maintenance facilities and change rooms for the two units. Multiple units can be located on the same site, using the footprint of the two unit arrangement as the basic building block with which to optimize the number of units on a particular site and maximize the additional electrical output from it.

The plant layout of the NG CANDU nuclear steam plant is also suitable for installation at existing nuclear sites in order to allow the existing auxiliary facilities and infrastructure to be utilized.

8.8.9.1. Reactor building

The reactor building contains the NSSS, including the reactor assembly, the HTS, the moderator system, the steam supply systems and the safety systems, together with the auxiliary systems and equipment for control and safety of the plant. The reactor building also provides the overall containment boundary of the NSSS.

8.8.9.2. Reactor auxiliary building

The reactor auxiliary building contains the main control room, secondary control area and associated equipment, those portions of safety systems located outside the reactor building, the Group 2 feedwater system and the safety related systems associated with safe operation of the plant.

8.8.9.3. Turbine building

The turbine building contains the turbine generator, condenser, pumps, feedwater heaters and the de-aerator equipment associated with the Group 1 feedwater system.

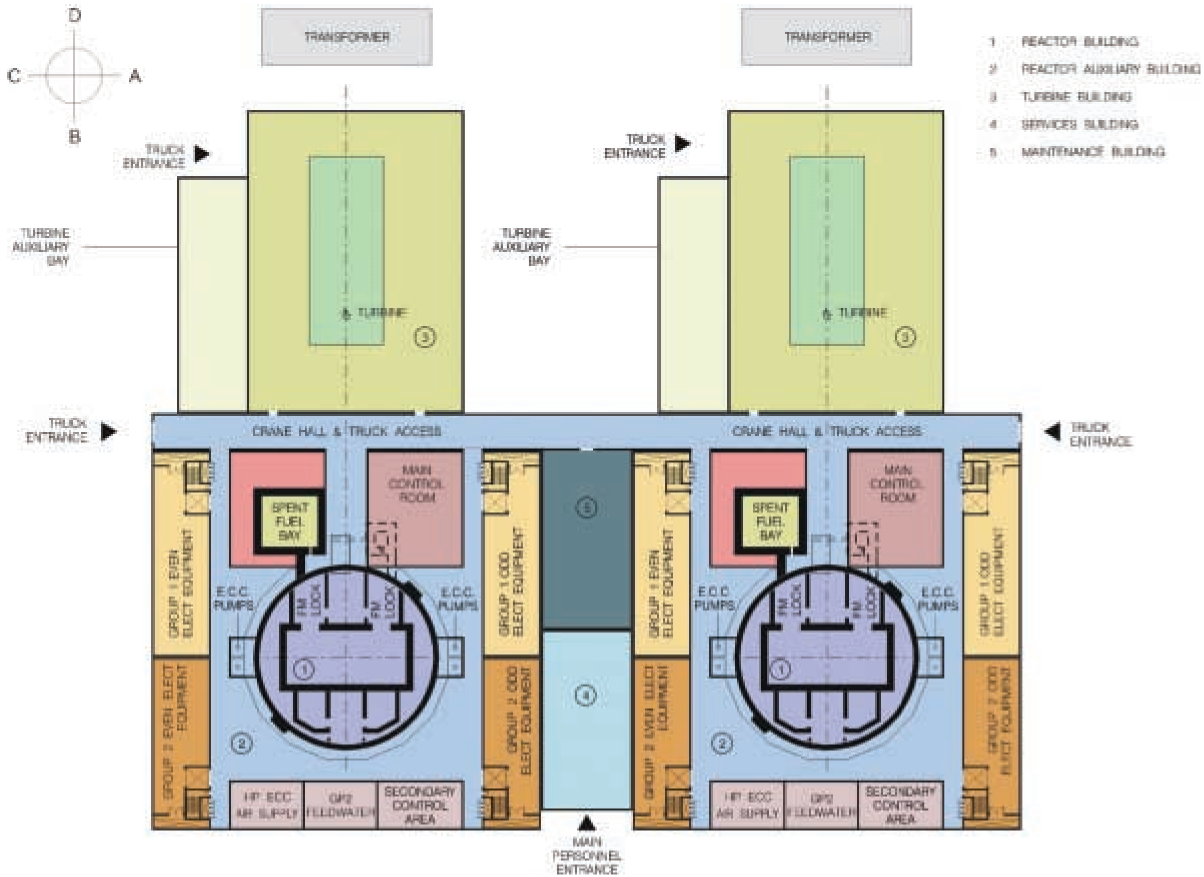


FIG. 232. NG CANDU two unit plant layout.

8.8.9.4. *Maintenance and services buildings*

The maintenance and services buildings contain the heavy water management, central stores, maintenance facilities and change room facilities. These facilities service two units in order to reduce the overall capital and operational costs of each unit.

8.8.9.5. *Auxiliary structures/buildings*

The Group 1 pump house contains the condenser cooling water pumps and the Group 1 raw service water pumps. The Group 2 raw service water pumps are located in the Group 2 pump house. Where sufficient cooling water is not available, cooling towers can be used to perform the functions of the condenser cooling water and raw service water systems.

The other auxiliary structures/buildings include the intake/outfall structures, administration buildings, water treatment facility and main switchyard. These structures/buildings will be located to suit the needs of any specific site.

8.8.9.6. *Layout approach*

Consistent with the grouping and separation approach, the Group 1 and Group 2 systems are located in physically separate areas of the plant (Fig. 233). Group 1 services are housed in the Group 1 areas of the reactor auxiliary building and the main pump house, and in a portion of the turbine building auxiliary bay. The reactor auxiliary building is seismically and environmentally qualified. The main steam safety valves and their enclosure are seismically qualified and protected from severe external events.

The majority of Group 2 services are accommodated within the Group 2 portion of the reactor auxiliary building and, to the extent practicable, are physically separated from the Group 1 areas. Group 2 areas and all essential equipment within them are seismically qualified and protected against design basis external events.

Both the main control room and the secondary control area are located in the reactor auxiliary building. They are seismically and environmentally qualified to protect the operator from all design basis events. A secure route is provided to allow the movement of personnel from the main control room to the seismically qualified secondary control area, located in the Group 2 portion of the reactor auxiliary building, following an event that causes a loss of operability or habitability of the main control room.

8.8.10. Nuclear steam plant

The nuclear steam plant comprises the reactor building and reactor auxiliary building, including all systems and equipment associated with the NSSS. Also

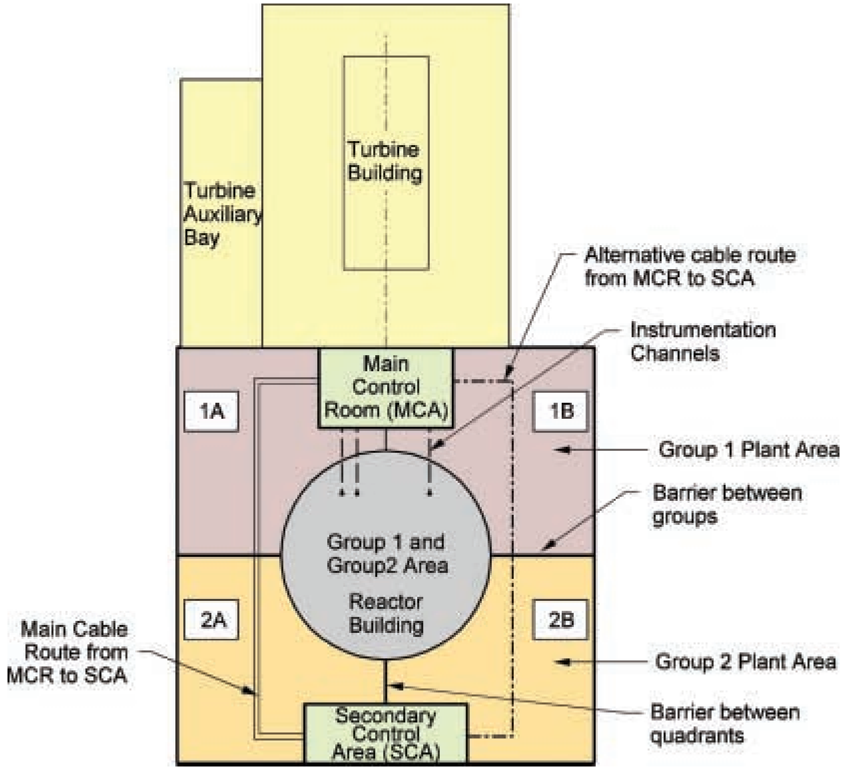


FIG. 233. Layout and grouping.

included are the safety and safety support systems, auxiliary systems and equipment for control and operation of the plant, the main control room, and the secondary control area.

The NSSS is similar in concept to the current CANDU reactor designs, as illustrated in Fig. 234. It includes the following features:

- Reactor assembly, consisting of an integral calandria/shield tank with 256 channels on a reduced square lattice pitch with larger diameter calandria tubes (Fig. 235);
- SEU fuel contained in the NG CANDU CANFLEX bundle with discharge burnup of 20 000 M·d/t U;
- Moderator system circulating heavy water for heat removal via two pumps and two heat exchangers;

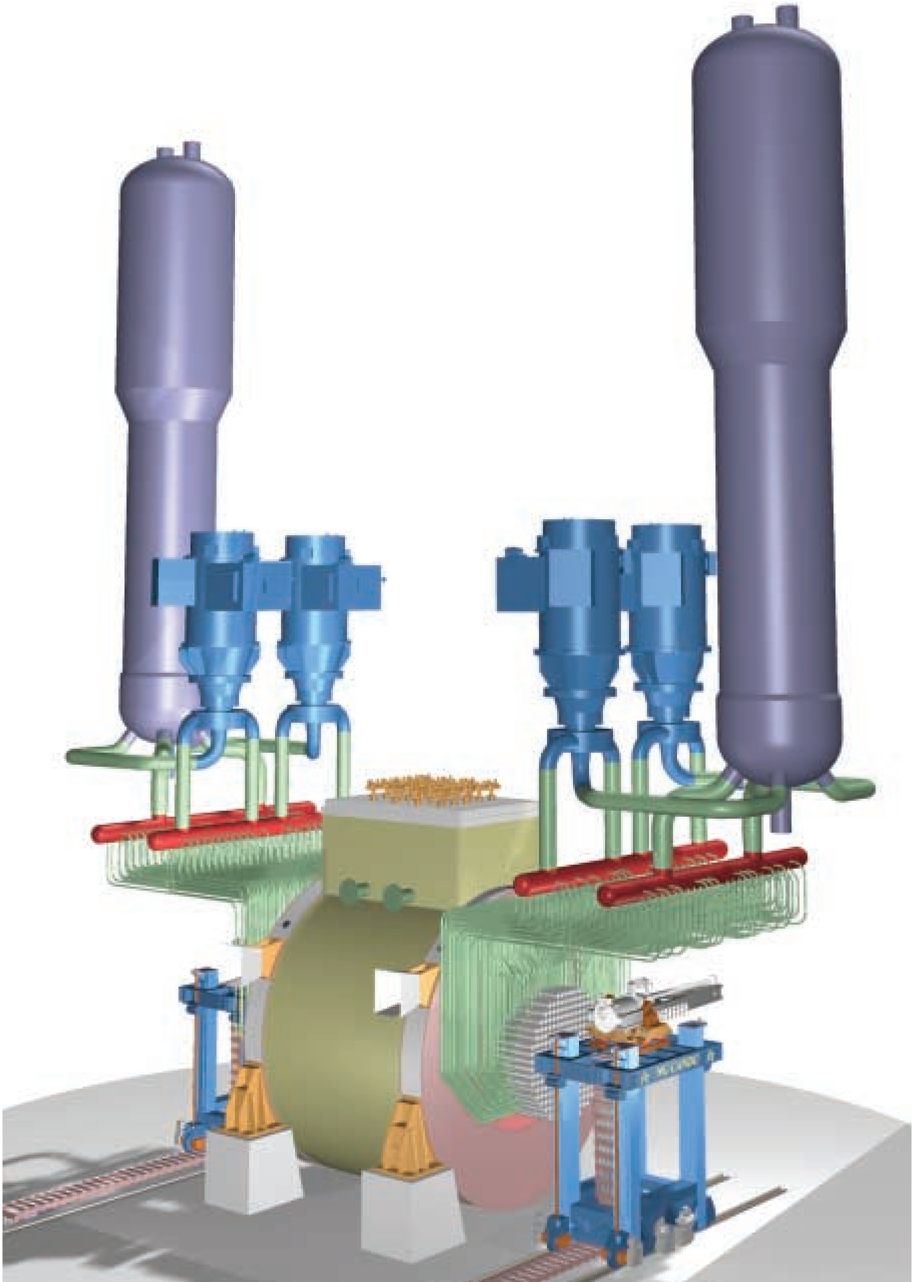


FIG. 234. Illustration of the NG NSSS.

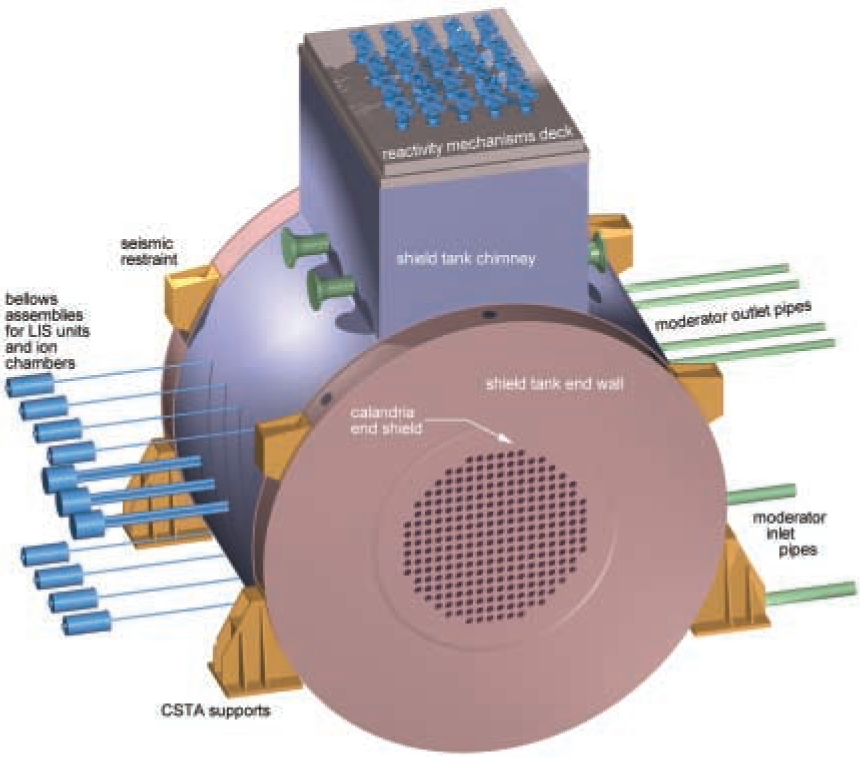


FIG. 235. The NG CANDU reactor assembly.

- HTS circulating light water coolant in a single loop, figure-of-eight configuration with two steam generators, four heat transport pumps, two reactor outlet headers and two reactor inlet headers;
- Fuel handling system consisting of two fuelling machines of improved design, mounted on fuelling machine carriages located at each end of the reactor;
- Main steam supply system with higher pressure and temperature conditions than the current CANDU 6 design, and a turbine generator that achieves an improved turbine cycle efficiency of approximately 37%;
- Compact reactor building, made possible by the simplified and more compact NSSS design.

A comparison of the major NG CANDU design parameters with those of other CANDU power systems is provided in Table LXI.

TABLE LXI. COMPARISON OF NG CANDU WITH OTHER CANDU POWER SYSTEMS

Plant/reactor	Fuel channels		HTS conditions				
	Gross/net* electrical power output (MW(e))	Number of fuel channels	Number of elements in fuel bundle	Number of loops	Reactor outlet header pressure (MPa(abs))	Maximum channel flow (kg/s)	Reactor outlet header quality (%)
Power systems:							
Pickering A	542/515	390	28	2	8.8	23.0	0.0
Pickering B	540/516	380	28	2	8.8	23.0	0.0
CANDU 6	715/668	380	37	2	10.0	24.0	4.0
Bruce A	904/840	480	37	1	9.2	24.0	0.7
Bruce B	915/860	480	37	1	9.2	24.0	0.7
Darlington	936/881	480	37	2	10.0	25.2	2.0
CANDU 9	940/875	480	37	1	10.0	25.2	2.0
NG CANDU	650/610	256	43	1	13.0	26.0	2.0

	Heat transport pumps			Steam generators			
	Total	Operating	Motor rating per pump (kW)	Number	Area per steam generator (m^2)	Integral preheater	Steam pressure (MPa(abs))
Power systems:							
Pickering A	16	12	1420	12	1850	Yes	4.1
Pickering B	16	12	1420	12	1850	Yes	4.1
CANDU 6	4	4	6700	4	3200	Yes	4.7
Bruce A	4	4	8200	8	2400	No	4.4
Bruce B	4	4	8200	8	2400	No	4.7
Darlington	4	4	9600	4	4830	Yes	5.1
CANDU 9	4	4	11 000	4	4970	Yes	5.1
NG CANDU	4	4	6100	2	6770	Yes	7.0

* Net output is dependent on condenser cooling water temperature and turbine design.

8.8.11. Safety and licensing

Various safety related systems are provided to mitigate the consequences of all accident scenarios, including four special safety systems (Section 8.8.11.1). These systems are located in the reactor building and in the reactor auxiliary building structures, both of which are seismically qualified to the DBE and designed to withstand any external events applicable to the particular site.

8.8.11.1. Special safety systems

The NG CANDU design retains the four special safety systems of the existing CANDU design:

- SDS1, consisting of mechanical shut-off rods similar to the CANDU 6 design, modified to accommodate the reduced lattice pitch;
- SDS2, consisting of gadolinium injection of a design similar to the CANDU 6 but with relocated and modified injection nozzles;
- ECCS, based on the CANDU 9 design, with further improvements made possible owing to the use of light water coolant, plus improved separation and redundancy;
- Containment system, including a steel lined, prestressed concrete reactor building structure that provides double isolation on all penetrations, plus a passive means of long term cooling following a postulated LOCA.

8.8.11.2. Safety support systems

The safety support systems provide all essential services necessary to satisfy the performance and reliability requirements of the four special safety systems and heat sinks. They include backup feedwater to the steam generators, electrical power, process cooling water, air supply and other auxiliary systems or equipment required to ensure safe shutdown of the plant. These systems meet the same availability, separation and redundancy requirements as the respective safety systems they support.

The grouping and separation approach used on existing CANDU plants has been extended in the NG CANDU design to provide improved protection against postulated common mode events including fires, tornadoes, earthquakes and hurricanes. This improvement is achieved by retaining two separate groups within the reactor building, then by providing further separation of these two groups within the reactor auxiliary building by installing redundant subsystems in each group. The resultant separate quadrants provide maximum separation and protection against common mode events.

8.8.11.3. Electrical systems

The major components of the electrical systems are grouped in areas to reduce installation effort, as services and general access are combined and centralized.

The Group 1 electrical systems and the Group 2 electrical systems are each located in separate areas that allow the requirements of each group to be efficiently satisfied. For example, the seismically and environmentally qualified Group 2 electrical distribution area is accommodated in qualified areas within the nuclear steam plant. The Group 1 system is also centralized to allow for ease of installation.

8.8.11.4. Instrumentation and control

A major advance in instrumentation and control, one which significantly enhances a plant's 'constructability', is the plant control and monitoring system. This is an integrated, plant wide digital control system consisting of channelized, redundant programmable control stations connected to channelized, redundant computer control stations. The distributed control system, utilized in NG CANDU, replaces the relay logic, analogue controllers and control computer input/output subsystems used in previous CANDU plants. Cabling, wiring and space requirements in the control equipment room are significantly reduced.

Another improvement to 'constructability' is that instrument tubing is routed to an instrument area immediately adjacent to the location of the measurement. By eliminating many kilometres of instrument tubing utilized on previous designs, both civil construction and installation of the tubing are made easier.

8.8.11.5. Balance of plant

The balance of plant consists of the turbine building, steam turbine, generator and condenser, the feedwater heating system and auxiliary equipment associated with the Group 1 feedwater supply and electricity generation. The balance of plant also includes the services building, pump houses and/or cooling towers, main switchyard and the associated equipment needed to provide all conventional services.

8.8.12. Advanced design methods and construction technology

The NG CANDU plant will take advantage of advanced design and construction technologies currently being implemented by AECL on the Qinshan CANDU 6 project. Further advancement of the design methods and construction technologies will be integrated into the NG CANDU plant throughout the design phases in order to achieve further reductions in project cost and schedule. These are discussed in the following sections.

8.8.12.1. Design methods

AECL has developed an integrated set of electronic aids to document the CANDU design, including flowsheets, equipment specifications and complete three dimensional computer aided drafting design models of the CANDU 6 plant design. These aids can also be used to improve engineering and procurement activities, materials management control, wiring/cabling information during construction, and project management control at a remote site. These aids have been implemented and used for these functions on the Qinshan CANDU 6 project.

Further advancements to these integrated aids and design methods will be implemented during the NG CANDU design phase, facilitated by using leading edge computer systems and information management systems, to provide an improved range of capabilities that will reduce design, capital and O&M costs, and enhance safety and configuration management. These capabilities include:

- Electronic 3-D design, often referred to as computer aided design or computer aided engineering;
- The use of common, controlled electronic databases for all project information and activities;
- Automated electronic data transfer (for example, from the 3-D model to analysis codes or to bills of material);
- Simulation of construction sequences and maintenance activities during design;
- Checking of structural, equipment and component spatial interferences.

These capabilities result in a substantial reduction in capital costs and improve the construction and project schedules by:

- Optimizing construction sequences and equipment and installation procedures and sequences.
- Reducing engineering and construction problems by eliminating interferences and space allocation control problems.
- Ensuring consistency of data throughout the project (design, analysis, licensing, commissioning and operation).
- Reducing information management and transfer costs, by transferring up-to-date and consistent information electronically to all participating parties in the project, including manufacturers, the regulatory authority, contract partners, construction companies, and the utility owner and operator of the plant.
- Reducing commissioning costs and schedule through the use of 3-D graphics and advanced database management.
- Reducing humanpower and schedule requirements in all aspects of the project, including conceptual design, detailed design, licensing, construction and commissioning.
- Improving materials management during procurement and site supply by automated production of isometric drawings and specifications for piping spools and piping hangers, etc.
- Facilitating automation of many operational activities (for example, plant surveillance and operations work control).

8.8.12.2. Construction methods

(a) Basis of the construction strategy

In the NG CANDU design, particular attention is paid to 'constructability' and to minimizing construction cost and schedule. A reduced construction schedule saves on interest incurred during construction, lowers capital cost and gives greater flexibility to the owner in adjusting the timing for installation of new generation capacity to meet market requirements.

The construction method/strategy for NG CANDU was defined early in the concept phase, as it is *the* major item impacting construction schedule. This strategy is addressed in the project design requirements and is considered from the earliest stages of layout to completion of the detailed design.

In developing the construction strategy for NG CANDU, experience gained with the Qinshan CANDU 6 project and the advanced techniques developed for other CANDU products were incorporated. The NG CANDU is therefore being developed with the construction schedule as a key requirement.

(b) Prefabrication/modularization

Prefabrication and modular construction techniques are integral parts of the costing and scheduling of advanced reactor designs. Some modules will comprise equipment and systems others will be structural, but most will be a combination of these. Modularization allows a significant proportion of work to be moved off-site, or at least away from the plant location. Advantages associated with modularization include reduced worker site population, improved access to work areas, improved safety, improved quality, elimination of shoring, shortening the duration of activities and paralleling activities. In order to achieve the shortest possible construction schedule, an overall target of 40% of construction person-hours to be shifted from site work to off-site modularization has been established. In certain critical areas, such as the reactor building's internal structure, even higher modularization targets will be achieved.

(c) Open top construction

Recent developments in very large mobile crane technology have made the installation of large pieces of equipment and prefabricated modules very practicable. In the case of the reactor building, it is desirable to eliminate temporary openings in the containment wall and very heavy lift (VHL) cranes have made it possible to leave the top off the containment structure, allowing installation of the internals through the open top of the reactor building. The use of VHL cranes has been

FIG. 236. Qinshan site: VHL crane lifting dousing system module through open top of reactor building (June 2001).

proven in the construction of the CANDU 6 units for Qinshan Phase III, as shown in Fig. 236.

Open top construction is optimized through installation of all major equipment, modules and materials through the top of the reactor building using external cranes. This not only applies to heavy lifts using the VHL crane, but also to all material lifts using conventional tower cranes.

(d) Parallel construction

Parallel construction is the paralleling of activities that were traditionally completed in series, e.g. both mechanical and electrical installation can proceed in parallel with the civil work. The schedule logic for NG CANDU makes maximum use of those parallel activities that greatly reduce the pressure on the critical path. Modularization and prefabrication are ideal techniques to use to support this strategy, as the modules can be fabricated in a shop while the civil work is progressing, ready to receive them.

(e) Use of up-to-date construction technologies

Up-to-date construction technologies are contributing to a shortened construction schedule through the use of:

- Prefabricated rebar,
- Large volume concrete pours,
- Pipe bending to replace elbow fittings,
- Automatic welding,
- Composite structures (concrete filled steel sections),
- Bridging systems and prefabricated permanent formwork,
- Prefabricated concrete elements.

8.8.13. Fuel cycle flexibility

The fundamental principle of the CANDU design is neutron economy. In the case of natural uranium fuelled CANDU plants, this principle is met through the use of heavy water, serving as both the moderator and the HTS coolant. In addition, on-power refuelling enables the CANDU reactor to operate with minimum excess reactivity in the core. The CANDU reactor is an efficient user of fissile material and the traces of fissile material found in natural uranium are sufficient to run the reactor.

In the NG CANDU reactor core, neutron economy continues to be a design advantage that is preserved through the use of heavy water in the moderator. However, the heavy water in the HTS of the conventional CANDU has been replaced with light water to reduce capital cost. The use of light water slightly reduces neutron economy and, as a result, the NG CANDU reactor requires the use of more fissile material, which can be obtained in uranium fuel with a slight enrichment in the fissile content. The NG CANDU cannot be operated with natural uranium fuel alone. However, the use of slightly enriched fuel enables the NG CANDU design to operate with the coolant at higher pressures and temperatures, enabling more electrical energy to be extracted from a given amount of fuel. Despite this trade-off, the NG CANDU still makes extremely efficient use of uranium resources and continues to share the CANDU heritage of fuel cycle versatility and flexibility.

The NG CANDU design has a unique synergy with the LWR fuel cycles, through the TANDEM cycle. There is sufficient fissile content in spent LWR fuel to burn in the NG CANDU as MOX fuel. In the TANDEM cycle, only the rare earth, neutron absorbing fission products are removed from the fuel; plutonium is not separated. This is a simpler and cheaper technology than conventional reprocessing. The option of recycling spent LWR fuels in NG CANDU leads to significantly higher energy extraction from the recycled material compared with self-recycle in an LWR.

The NG CANDU is very attractive for the TANDEM cycle and for other fuel cycles using MOX fuel because the neutronic characteristics of the NG CANDU core do not require the addition of neutronic poisons to the fuel.

The NG CANDU is also capable of operating on the DUPIC fuel cycle, where there is no selective removal of fission products.

The use of thorium based fuel cycles offers an opportunity to greatly extend the sustainability and flexibility of nuclear power production. The NG CANDU will be capable of operating using a variety of Th–^{233}U fuel cycles.

The neutron economy of the NG CANDU impacts on radioactive waste disposal, as higher burnup results in a smaller quantity of spent fuel. Also, the NG CANDU reactor could be used to reduce the radiotoxicity of spent fuel through a novel fuel cycle using inert matrix fuel. There is sufficient fissile content in mixtures of plutonium and the higher actinides (a by-product of LWR spent fuel reprocessing) to be used as fuel in NG CANDU when diluted in an inert matrix material such as SiC, with no addition of uranium or other fertile fuel material. The absence of uranium prevents the formation of more plutonium during fuel burnup. The fissile content of the transuranic mix is depleted rapidly owing to the lack of plutonium formation. As a result, the level of neutron flux required to maintain reactor power at the rated level is high, and this high neutron flux is instrumental in transmuting and annihilating the toxic material. The NG CANDU reactor can therefore produce energy through the destruction of toxic waste without producing additional such waste in the process.

With the ability to switch to alternative nuclear fuels available on the global market, the NG CANDU offers an owner the capability to react to market changes in fuel cost and availability, and also provides the capability to address fuel cycle opportunities arising from synergies with other power plants or waste management initiatives.

8.9. CONCLUSIONS

HWRs have significant development potential and their design and performance are continually being improved. Improvements are based on what is needed to optimize current designs, and on what is desirable in order to exploit the development potential inherent in the design. There is a continuous competition between using existing tried, tested and proven components and introducing key design innovations.

Section 8 has focused primarily on the visionary concepts of a distant evolutionary design that represents a natural extension of HWR technology into the medium and long terms. Many of the ideas discussed would yield a reduction in LUEC and enhanced passive safety for an SCW cooled (H_2O or D_2O) reactor. The applicability of various ideas for introduction into conventional HWR designs in the short to medium term will have to be evaluated on economic grounds and on the basis of market needs. The medium term developments are those of the evolutionary HWR pressure tube concept and the AHWR. Should the needs of ultrasafe operation become dominant, the HWR 1000 design may become feasible. It is possible that the market for HWRs will expand if the intrinsic environmental advantage of meeting base load electricity requirements with zero greenhouse gas emissions is exploited. As demonstrated in Section 8.1, the greenhouse gas emission advantage of HWRs can be spread into different sectors of the economy through innovative approaches that capitalize on the synergy that exists between HWRs and alternative energy sources. These innovative approaches and conduct of the required development in the short term will ensure that HWRs remain an integral component of long term energy policy.

Appendix

PARAMETERS OF THE PRINCIPAL TYPES OF HWR

TABLE LXII. SMALL PRESSURE TUBE PRESSURIZED POWER REACTORS

Parameter	Nuclear power plant	
	Douglas Point	KANUPP
Reactor:		
Thermal output (MW(th))	693	432
Gross electrical output (MW(e))	220	137
Net electrical output (MW(e))	200	125
Operating temperature (RIH) (°C)	249	246
Operating pressure (RIH) (MPa)	9.9	11.4
Operating temperature (ROH) (°C)	293	293
Operating pressure (ROH) (MPa)	9.2	10.9
Calandria vessel:		
Form	Stepped horizontal cylinder	Stepped horizontal cylinder with integral dump space
Material	Austenitic stainless steel type 304L	Austenitic stainless steel type 304L
Inside diameter/length (m)	5.08/6.045	4.904/4.95
Moderator	D_2O	D_2O
Moderator weight (t)	144	100
Calandria end shields:		
Material	3.5% Ni steel separate from calandria shell	Austenitic stainless steel type 304L
Fill	Steel plates and concrete	Light water and steel lined heavy concrete vault
Calandria tubes:		
Material/number	Zircaloy 2/306	Zircaloy 2/208
Inside diameter/wall thickness (mm)	107.7/1.24	104.1/1.44

TABLE LXII. (cont.)

Parameter	Nuclear power plant	
	Douglas Point	KANUPP
Fuel channels:		
PT inside diameter/wall thickness (mm)	82.6/4.06	82.84/4.32
PT material	cw. Zircaloy 2	HT Zr–2.5%Nb
Number of PTs	306	208
Coolant flow rate (kg/s)	3040	1909
Lattice pitch (mm)	228.6	235
Fuel:		
Bundle length/outside diameter (mm)	495/81.7	495/82.6
Weight of UO_2/bundle (kg)	13.4	12.16
Sheath outside diameter/wall thickness (mm)	15.2/0.38	15.2/0.38
Sheath material	Zircaloy 2	Zircaloy 4
Elements/bundle	19 (wire wrap)	19 (bearing pads)
Fuel material	UO_2 (natural)	UO_2 (natural)
Fuel bundles in core	3672	2288
Fuel bundles in channel	12	11
Reactivity control unit:		
Shutdown devices	Moderator dump	Moderator dump
Control units	4, fine absorber rods Booster rods (enriched U-235) Injection of cadmium sulphate to moderator	4, fine absorber rods 8, booster rods Adjustment of moderator level Chemical shim (boron) to moderator
Primary heat transport:		
Number of loops	2	2
Primary coolant	D_2O	D_2O
Reactor inlet temperature (°C)	249	246
Reactor outlet temperature (°C)	293	293
Number of heat transport pumps	2	8 (2 on stand-by)

TABLE LXII. (cont.)

Parameter	Nuclear power plant	
	Douglas Point	KANUPP
Steam generators:		
Type/number	10 units of U-bends in hairpin type shell with common steam drum/8 units	6 U-shaped shell and tube steam generators (2 banks, each with 3 boilers in parallel)
Number of tubes/material	1950/Monel	Monel
Steam flow per reactor (kg/s)	456.9	207.5
Steam pressure at full power (MPa)	4.14	4.04
Steam temperature (°C)	250.6	250
Containment:		
Type	Cylindrical concrete with hemispherical steel dome	Concrete base slab, prestressed concrete cylinder with hemi-spherical dome, elastomeric lining on inside surface
Diameter/thickness/height (m)	39.6/1.2/42.7	35/1.346/37.5
Turbine	1, tandem compound	1, single shaft tandem compound horizontal impulse turbine
Generator	1 244 444 kVA 3-phase	137 MW, 3000 rpm, 3-phase, one synchronous generator
Main condenser:		
Coolant water	Fresh water	Sea water, one two-pass divided water box condenser, surface type, exhaust pressure 0.07 kg/cm^2
Condenser tube material	Admiralty brass (stainless steel where erosion resistance is required), 25.4 mm o.d. tubes	

TABLE LXIII. CANADIAN PRESSURIZED PRESSURE TUBE HEAVY WATER COOLED, HEAVY WATER MODERATED REACTORS

Parameter	Nuclear power plant	
	Pickering A, PHWR integrated 4-units	Pickering B, PHWR integrated 4-units
Reactor:		
Thermal output (MW(th))	1742	1744
Gross electrical output (MW(e))	540	540
Net electrical output (MW(e))	508	508
Coolant flow rate (kg/s)	7724	8308
Operating temperature (RIH) (°C)	249	249
Operating pressure (RIH) (MPa)	9.5	9.5
Operating temperature (ROH) (°C)	293	293
Operating pressure (ROH) (MPa)	8.8	8.8
Calandria vessel:		
Form	Horizontal stepped shell comprising main shell, two subshells, and two annulus plates	Horizontal stepped shell comprising main shell, two subshells, and two annulus plates
Material	Austenitic stainless steel type 304L	Austenitic stainless steel type 304L
Inside diameter/length (m)	8.04/5.94	8.04/9.94
Moderator	D_2O	D_2O
Moderator volume (m^3)	242	218
Heat load (MW(th))	90.3	86.9
Calandria end shields:		
Material	Austenitic stainless steel type 304L	Austenitic stainless steel type 304L
Fill	Steel plates/light water	Steel plates/light water
Calandria tubes:		
Material/number	Zircaloy 2/390	Zircaloy 2/380
Inside diameter/wall thickness (mm)	142/1.57	129/1.37
Fuel channels:		
PT inside diameter/wall thickness (mm)	103/4.01	103/4.01
PT material	cw. Zircaloy 2/repl. cw. Zr–2.5%Nb	cw. Zr–2.5%Nb
Number of PTs	390	380
Coolant flow rate (kg/s)	7724	7728

TABLE LXIII. (cont.)

Parameter	Nuclear power plant	
	Pickering A, PHWR integrated 4-units	Pickering B, PHWR integrated 4-units
Lattice pitch (mm)	285.8	285.8
Fuel:		
Bundle length/outside diameter (mm)	495/102.4	495/102.4
Weight of UO_2/bundle (kg)	22	22
Sheath outside diameter/wall thickness (mm)	15.2/0.38	15.2/0.38
Sheath material	Zircaloy 4	Zircaloy 4
Elements/bundle	28	28
Fuel material	UO_2 (natural)	UO_2 (natural)
Fuel bundles in core	4680	4560
Fuel bundles in channel	12	12
Reactivity control units:		
Shutdown devices	Stainless steel clad cadmium tubes: vertical, 11 off Moderator dump	Stainless steel clad cadmium shut-off tubes: 28 vertical Gadolinium nitrate injection into moderator — 6 nozzles: horizontal
Control units	Cobalt adjuster rods (18), vertical Zone control units (H_2O), 14 off, vertical Boron in moderator	Stainless steel (unit 5) cobalt adjuster rods (units 6,7,8), 21 off Zone control units (14) light water, vertical Cadmium–stainless steel control absorbers, 4, vertical Boron in moderator
Primary heat transport:		
Number of loops	2	2
Primary coolant	D_2O	D_2O
Reactor inlet temperature (°C)	249	249
Reactor outlet temperature (°C)	293	293
Number of heat transport pumps	16	16

TABLE LXIII. (cont.)

Parameter	Nuclear power plant	
	Pickering A, PHWR integrated 4-units	Pickering B, PHWR integrated 4-units
Steam generators:		
Type/number	Inverted U-tube/12	Inverted U-tube/12
Number of tubes/material	2600/Monel	2573/Monel
Steam flow rate per reactor (kg/s)	815	815
Steam pressure at full power (MPa)	4.2	4.1
Steam temperature (°C)	251	251
Containment:		
Type	Reinforced concrete cylinder	Reinforced concrete cylinder with elliptical concrete dome
Diameter/thickness/height (m)	50.3/0.93/50.6	42.7/1.2/46.6
Turbine	1 tandem compound unit per reactor, 1 HP double flow cylinder, 3 LP double flow cylinders	1 tandem compound unit per reactor, 1 HP double flow cylinder, 3 LP double flow cylinders
Generator	540 MW(e), 1800 rpm	540 MW(e), 1800 rpm
Main condenser:		
Coolant water flow rate (m^3/s)	23.7	23.7
Condenser tube material	Admiralty brass/repl. 316 stainless steel	Admiralty brass/repl. 316 stainless steel

TABLE LXIV. CANADIAN INTEGRATED 4-UNIT PRESSURIZED PRESSURE TUBE REACTORS

Parameter	Nuclear power plant		
	Bruce A	Bruce B	Darlington
Reactor:			
Type	PHWR	PHWR	PHWR
Thermal output (MW(th))	2551		2949
Gross electrical output (MW(e))	791	807	936
Net electrical output (MW(e))	740	750	881
Operating temperature (RIH) (°C)	265 (outer region) 251 (inner region)	265 (outer region) 250 (inner region)	267
Operating pressure (RIH) (MPa)	10.40 (outer region) 10.25 (inner region)	10.42 (outer region) 10.3 (inner region)	11.38
Operating temperature (ROH) (°C)	305	305	310
Operating pressure (ROH) (MPa)	9.18	9.31	10.0
Calandria vessel:			
Form	Horizontal stepped shell comprising main shell, two subshells, and two annulus plates	Horizontal stepped shell comprising main shell, two subshells, and two annulus plates	Horizontal stepped shell comprising main shell, two subshells, and two annulus plates
Material	Austenitic stainless steel type 304L	Austenitic stainless steel type 304L	Austenitic stainless steel type 304L
Inside diameter (m)	8.46	8.46	8.458
Main shell thickness (mm)	31.7	31.7	31.75
Length (m)	5.95	5.95	5.981
Moderator	D_2O	D_2O	D_2O (99.95%)

TABLE LXIV. (cont.)

Parameter	Nuclear power plant		
	Bruce A	Bruce B	Darlington
Moderator volume (m^3)	306.3	306.3	312.0
Heat load to moderator (MW(th))	147	122.49	138.0
Calandria end shields:			
Material	Austenitic stainless steel type 304L	Austenitic stainless steel type 304L	Austenitic stainless steel
Fill	Light water and carbon steel balls	Light water and carbon steel balls	Light water and carbon steel balls
Weight (filled) (Mg)	229	229	237
Length (m)	1.06	1.06	1.016
Calandria tubes:			
Number	480	480	480
Inside diameter/wall thickness (mm)	129/1.37	129/1.37	129/1.37
Fuel channels:			
PT inside diameter (mm)	103.4	103.4	103.4
PT wall thickness (mm)	4.22	4.22	4.22
PT material	Zr–2.5%Nb	Zr–2.5%Nb	Zr–2.5%Nb
Number of PTs	480	480	480
Coolant flow rate/channel (kg/s)	253 (nominal)	23.8 (nominal)	25.2 (maximum)
Lattice pitch (mm)	285.8	285.8	285.8 (square)

TABLE LXIV. (cont.)

Parameter	Nuclear power plant		
	Bruce A	Bruce B	Darlington
Fuel:			
Bundle length (mm)	495	495	495
Outside diameter of bundle (mm)	102.49	102.49	102.49
Weight of bundle (nominal) (kg)	23.65	23.5	23.5
Weight of upper bundle (kg)	21.36 (UO_2)	21.23 (UO_2)	21.23 (UO_2)
Sheath outside diameter (mm)	13.1	13.1	13.1
Sheath thickness (mm)	0.4	0.4	0.4
Sheath material	Zircaloy 4	Zircaloy 4	Zircaloy 4
Elements per bundle	37	37	37
Fuel material	UO_2 (natural)	UO_2 (natural)	UO_2 (natural)
Fuel bundles in core	6240 (108 t U)	6240 (117 t U)	6240
Fuel bundles in channel	13	13	13
Maximum bundle power (kW)	787	787	787
Reactivity control units:			
Shutdown devices	Shut-off rods, 30 stainless steel/cadmium/stainless steel tubes, vertical, –40 mk in 2 s	Shut-off rods, 32 stainless steel/cadmium/stainless steel sandwich in tubular form, vertical, –49 mk in 2 s	Shut-off rods, 32 stainless steel/cadmium/stainless steel tubes, vertical, –49 mk in 2 s
	Liquid poison injection, gadolinium nitrate into moderator, horizontal, –65 mk	Liquid poison injection, gadolinium nitrate into moderator, horizontal, 8 nozzles, –55 mk	Liquid poison injection, gadolinium nitrate into moderator, horizontal, –55 mk

TABLE LXIV. (cont.)

Parameter	Nuclear power plant		
	Bruce A	Bruce B	Darlington
Control units	Zone control units, light water in compartments in 14 separate zones, vertical, 6.3 mk (±0.115 mk/s) Control absorbers, 4 stainless steel tubes Addition of boron to moderator	Zone control units, light water in compartments in 14 separate zones, vertical, 6.3 mk (±0.1 mk/s) Control absorbers, 8 stainless steel adjuster tubes, –17.5 mk Adjuster rods, 3 vertical banks, each with 8 adjuster rods, –17.5 mk	Zone control units, light water in compartments in 14 separate zones, vertical, 16.3 mk (±0.1 mk/s) Control absorbers, 4 stainless steel/cadmium/ ss rods, –9 mk (±0.1 mk/s) Adjuster rods, 16 stainless steel rods, +12.6 mk
Primary heat transport:			
Number of loops	4	4	4
Primary coolant	D_2O	D_2O	D_2O
Reactor inlet temperature (°C)	249	250	267
Reactor outlet temperature (°C)	300	306	310
Steam generators:			
Type/number	Inverted U-tube, 8 per unit common steam drum	Inverted U-tube, light bulb shell, individual units (8 per reactor)	Inverted U-tube, light bulb shell, 4 per reactor
Number of tubes/material	4200/I600	4200/I600	4663/I800
Steam flow rate for 8 steam generators (kg/s)	680.6	680.6	1311.1
Steam pressure at full power (MPa)	4.27	4.27	5.068
Steam temperature at full power (°C)	255.7	256	264.7

TABLE LXIV. (cont.)

Parameter	Nuclear power plant		
	Bruce A	Bruce B	Darlington
Feedwater temperature (°C)	168	247	176.7
Heat transport pumps:			
Number	4	4	4
Rated capacity (m^3/s)	3.3	3.3	3.1
Rated head (m)	210	213	224.2
Containment:			
Type	Rectangular building, reinforced concrete	Rectangular building, reinforced concrete	Rectangular building, reinforced concrete
Length/width/height (m)	31.7/28.04/49.58	31.7/28.04/49.58	49.8/28.6/51.1
Reactor vault:			
Design pressure (kPa)	–48.3 to +68.95	–48.3 to +82.8	–53.1 to +96.5
Turbine	One tandem compound unit per reactor, 1 HP stage, 3 LP stages	One tandem compound unit per reactor, 1 HP stage, 3 LP stages	One tandem compound unit per reactor, 1 HP double flow cylinder, 3 LP double flow cylinders
Generator	One per turbine, 1800 rpm	One per turbine, 1800 rpm	One per turbine, 1800 rpm
Main condenser:			
			3 per unit, 2 pass
Coolant water flow rate (m^3/s)	38.0	41.7	31.6

TABLE LXV. INDIAN PRESSURE TUBE PRESSURIZED POWER REACTORS

Parameter	Nuclear power plant		
	Rajasthan 1/2 Kalpakkam 1/2	Narora 1/2 Kakrapar 1/2	500 MW(e) Tarapur 1/2
Reactor:			
Thermal output (MW(th))	693.5[a]	862	1835
Gross electrical output (MW(e))	100–200[b]	220	500 (approx.)
Net electrical output (MW(e))	90–187[c]	202	450
Operating temperature (RIH) (°C)	249	249	260
Operating pressure (RIH) (MPa)	9.9	9.9	11.4
Operating temperature (ROH) (°C)	293	293	304
Operating pressure (ROH) (MPa)	9.2	8.7	9.9
Calandria vessel:			
Form	Horizontal cylinder	Horizontal cylinder	Horizontal stepped cylinder
Material	Austenitic stainless steel type 304L	Austenitic stainless steel type 304L	Austenitic stainless steel type 304L
Inside diameter/length (m)	5.08/6.045	5.08/6.045	7.86/4.664
Moderator	D_2O	D_2O	D_2O
Moderator weight (t)	146	144	260
Calandria end shields:			
Material	3.5% NT steel separate from calandria	Austenitic stainless steel type 304L integrated with calandria shell	Austenitic stainless steel type 304L integrated with calandria shell

TABLE LXV. (cont.)

Parameter	Nuclear power plant: Rajasthan 1/2, Kalpakkam 1/2	Nuclear power plant: Narora 1/2, Kakrapar 1/2	Nuclear power plant: 500 MW(e), Tarapur 1/2
Fill	Steel slabs, water cooled concrete air vault	Steel ball filled, cooled with light water, water filled vault	Steel ball filled, cooled with light water, water filled vault
Calandria tubes:			
Material/number	Zircaloy 2/306	Zircaloy 2/306	Zircaloy 4/392
Inside diameter/wall thickness (mm)	107.7/1.24	107.7/1.24	129.2/1.4
Fuel channels :			
PT inside diameter/wall thickness (mm)	82.6/4.06	82.6/4.06	103.4/4.5
PT material	Zircaloy 2 (Rajasthan 2 retubed with Zr–2.5%Nb)	Zircaloy 2 (Narora 1/2, Kakrapar 1) Zr–2.5%Nb (Kakrapar 2)	cw. Zr–2.5%Nb
Number of PTs	306	306	392
Coolant flow rate (kg/s)	3260	3260	7814
Lattice pitch (mm)	228.6	228.6	285.75
Fuel:			
Bundle length/outside diameter (mm)	495	495	495.3/102.36
Weight of UO_2/bundle (kg)	15.2	15.2	21.78
Sheath outside diameter/wall thickness (mm)	15.2/0.41	15.2/0.41	13.08/0.4

For footnotes see end of table.

TABLE LXV. (cont.)

Parameter	Nuclear power plant		
	Rajasthan 1/2 Kalpakkam 1/2	Narora 1/2 Kakrapar 1/2	500 MW(e) Tarapur 1/2
Sheath material	Zircaloy 2	Zircaloy 2	Zircaloy 4
Elements/bundle	19 (wire wrap)	19 (spacers)	37
Fuel bundles in core/channel	3060/10	3060/10	4704/13
Reactivity control units:			
Shutdown devices	Mechanical shut-off rods Liquid poison injection (12 nozzles) Moderator dump	14 mechanical shut-off rods (shim rods) Liquid poison injection Moderator dump	Mechanical shut-off rods (28 rods, –79 mk) Liquid poison tube system Liquid poison injection into moderator (–300 mk, 6 units)
Control units	Absorber rods Booster rods Boron in moderator	Absorber rods Booster rods Boron in moderator	14 liquid zone (–7 mk) 17 adjuster rods (–12 mk) 4 rods (–10 mk)
Primary heat transport:			
Number of loops	1	1	2
Primary coolant	D_2O	D_2O	D_2O (4 pumps, 215 m head)
Reactor inlet temperature (°C)	249	249	
Reactor outlet temperature (°C)	293	293	
Number of heat transport pumps	8	4	4

TABLE LXV. (cont.)

Parameter	Nuclear power plant		
	Rajasthan 1/2 Kalpakkam 1/2	Narora 1/2 Kakrapar 1/2	500 MW(e) Tarapur 1/2
Steam generators:			
Type/number	8	4	Vertical U-tube/4
Number of tubes/material	Monel	Monel	2500/Incoloy 800
Steam flow rate per reactor (kg/s)	329.9	329.9	855
Steam pressure at full power (MPa)	3.97	3.97	3.92 (g)
Steam temperature (°C)	250	250	253
Containment:			
Type	Reinforced concrete cylindrical structure with prestressed, hemispherical dome Single wall (Rajasthan) Partial double wall shell, single dome at Kalpakkam	Full double wall shell, single dome	Cylindrical, full double containment
Diameter/thickness/height (m)			51/0.75 55.94/0.61/50.55 Gap 1.86
Turbine	One	One	One, tandem compound, double flow 1 HP, 2 LP cylinders

For footnotes see end of table.

TABLE LXV. (cont.)

Parameter	Nuclear power plant: Rajasthan 1/2, Kalpakkam 1/2	Nuclear power plant: Narora 1/2, Kakrapar 1/2	Nuclear power plant: 500 MW(e), Tarapur 1/2
Generator	3000 rpm, 220 MW(e)	One per turbine, 1800 rpm	One per turbine, 3000 rpm Direct coupled, hydrogen cooled rotor, water cooled stator, 0.85 PF, 50 HZ
Main condenser:			
Coolant water flow rate (m^3/s)	Lake water	Lake water	15.97 sea water
Condenser tube material			Titanium
Back pressure (kPa)			8.43 (abs)
Heat load per condenser(kJ/s)			6.21×10^5
Minimum surface area required (m^2)			27 292
Tube sheet material			Titanium clad carbon steel

[a] Original design figure.

[b] Rajasthan 1 (100 MW(e)), Rajasthan 2 (200 MW(e)), Kalpakkam 1/2 (170 MW(e)).

[c] Rajasthan 1 (90 MW(e), Rajasthan 2 (187 MW(e)), Kalpakkam 1/2 (155 MW(e)).

TABLE LXVI. PRESSURIZED PRESSURE TUBE HEAVY WATER COOLED, HEAVY WATER MODERATED REACTORS

Parameter	Nuclear power plant	
	CANDU 6	CANDU 9
Reactor:		
Type	PHWR	PHWR
Thermal output (MW(th))	2064 (PHTS)	2720 (HTS)
Coolant flow rate (Mg/s)	7.7 (PHTS)	11 (HTS)
Design temperature (RIH) (°C)	279	279
Design pressure (RIH) (MPa)	12.7 (abs)	12.7 (g)
Design temperature (ROH) (°C)	318	318
Design pressure (ROH) (MPa)	11.0 (abs)	11.0 (g)
Fuel channels:		
PT inside diameter (mm)	103.4	103.4
Core length (m)	5.94	5.94
Core diameter (calandria) (m)	7.6	8.458
Number of PTs	380	480
Coolant flow rate (nominal)/channel (kg/s)	24	25.2
Estimated pressure drop across 12 bundles (kPa)	758	830
Fuel:		
Length of bundle (mm)	495.3	495.3
Outside diameter of bundle (over bearing pads) (mm)	102.4	102.4
Weight of bundle (nominal) (kg)	24	24
Weight of uranium per bundle (nominal) (kg)	19.2	19.2
Sheath outside diameter (cold) (mm)	13.1	13.1
Sheath thickness (average) (mm)	0.4	0.4
Sheath material	Zircaloy 4	Zircaloy 4
Elements per bundle	37	37
Fuel material	UO_2 (natural)	UO_2 (natural)
Fuel bundles in core	4560	5760
Fuel bundles per channel	12	12
Primary heat transport:		
Number of loops	2	1
Primary coolant	D_2O	D_2O
Temperature (RIH) (°C)	266	266
Pressure (RIH) (MPa)	11.7 (abs)	11.3 (abs)

TABLE LXVI. (cont.)

Parameter		Nuclear power plant	
		CANDU 6	CANDU 9
Temperature (ROH) (°C)		310	310
Pressure (ROH) (MPa)		10.0 (abs)	10.0 (abs)
Reactivity control units:			
Number of assemblies		85 vertical 19 horizontal	81 vertical 28 horizontal
Materials	(out of core)	Stainless steel	Stainless steel
	(in-core)	Zircaloy/stainless steel/cadmium	Zircaloy/stainless steel/cadmium
Steam generators:			
Type/number		Vertical U-tube/4	Vertical U-tube/4
Steam flow for 4 steam generators (kg/s)		1033	1330
Steam pressure at full power (MPa)		4.7 (abs)	5 (g)
Steam temperature at full power (°C)		260	260
Maximum moisture (%)		0.25	0.1
Feedwater temperature (°C)		187	177
Heat transport pumps:			
Number		4	4
Motor/type		AC vertical/TEWAC induction	AC vertical/squirrel cage induction
Rated capacity (L/s)		2228	3200
Rated head (m)		215	263.5
Containment:			
Type		Prestressed cylindrical concrete	Prestressed concrete with steel liner
Inside diameter (m)		41.46	57
Height above grade (m)		46.02	67.5
Total internal volume (m^3)		65 500	124 000

TABLE LXVI. (cont.)

Parameter	Nuclear power plant	
	CANDU 6	CANDU 9
Turbine	Single shaft tandem compound steam turbine directly coupled to 728 MW(e) generator. Steam turbine consists of one double flow HP cylinder, two external moisture separators/ reheaters and two double flow LP cylinders.	Single shaft tandem compound steam turbine directly coupled to 940 MW(e) (dependent on the supplier and site conditions) generator. Steam turbine consists of one double flow HP cylinder, two external moisture separators/reheaters and three double flow LP cylinders.
Generator	Rated 815 MVA at 0.9 power factor and 414 kPa (g) hydrogen pressure	
Main condenser	Designed with two separate tube sheet shells. Each shell is connected to one of the two LP turbine exhausts.	Design dependent on manufacturer

TABLE LXVII. BOILING LIGHT WATER COOLED, HEAVY WATER MODERATED REACTORS

Parameter	Nuclear power plant			
	SGHWR	Gentilly 1	Fugen	Cirene
Reactor:				
Thermal output (MW(th))	308.2	832.9	557	130
Generator output (MW(e))	102.4	260	165	40
Net electrical output (MW(e))	94.3	250	148	36
Calandria vessel:				
Form	Vertical cylinder	Vertical cylinder, conical section at top	Vertical cylinder, main shell and two subshells	Vertical cylinder, conical step at base, co-axial lower shell as dump annulus 5.2 m diameter
Material	Aluminium–magnesium alloy	Austenitic stainless steel type 304L	Austenitic stainless steel type 304L	Austenitic stainless steel type 304L
Inside diameter/height (m)	3.71/3.96	5.54/5.00	4.90 (main shell)/3.70	3.69/4.70
Moderator	D_2O	D_2O	D_2O	D_2O
Moderator weight (t)	39	212	86	55
End shields:				
Fill	H_2O, carbon shot, steel, concrete	Steel plates and H_2O	Steel plates and H_2O	Steel plates and H_2O
Upper auxiliary	Austenitic stainless steel/light water			

TABLE LXVII. (cont.)

Parameter	Nuclear power plant			
	SGHWR	Gentilly 1	Fugen	Cirene
Calandria tubes:				
Material/number	Aluminium alloy/103	Zircaloy 2/308	Zircaloy 2/224	Zircaloy 2/60
Diameter/wall thickness (mm)	177.8/3.3	118.1/1.02	156.4/1.9	124/1.0
Fuel channels:				
PT inside diameter/wall thickness (mm)	130.6/5.08	104/2.4	117.8/4.3	106.1/3.15
PT material	Zircaloy 2	HT Zr–2.5%Nb	HT Zr–2.5%Nb	Zircaloy 2
Number of PTs	103	308	224	60
Coolant/flow rate	H_2O/3.66 m/s	H_2O/100 $kg \cdot h^{-1} \cdot cm^{-2}$	H_2O/3 m/s	H_2O/
Lattice pitch (mm)	260.3	279.4	240	270
Fuel:				
Bundle length/outside diameter (mm)	3660/133	495/102.4	4388/110	
Weight of UO_2/bundle (kg)	19.8	21.1	MOX fuel	
Sheath outside diameter/wall thickness (mm)	15.2/0.71	19.8/0.59	16.46/0.88	0.51
Sheath material	Zircaloy 2	Zircaloy 4	Zircaloy 2	Zircaloy 2
Elements/bundle	36	18	28	18
Fuel material	2.28% $^{235}UO_2$	UO_2 (natural)	2.0% $^{235}UO_2/PuO_2$	0.71–1.15% $^{235}UO_2$
Fuel bundles in core	103	3080	224	480
Fuel bundles in channel	1	10	1	8
Maximum channel power (MW(th))	3.8	3.1		

TABLE LXVII. (cont.)

Parameter	Nuclear power plant			
	SGHWR	Gentilly 1	Fugen	Cirene
Reactivity control units:				
Shutdown devices:				
Device #1	Boron injection into tubes in moderator	Gadolinium nitrate injection into moderator	B_4C rods/gravity drop	Moderator dump
Device #2	Moderator dump	Moderator dump	Moderator dump	Liquid poison SOR's
Control units:				
Device #1	Liquid absorber in tubes	Vertical absorber rods	B_4C regulating rods	U-shaped tubes (2-phase rods)
Device #2	Varying moderator height	Concentration of boron in moderator	Chemical shim	Moderator height
Device #3	Solid rods	Coolant flow		Concentration of boron in moderator
	Concentration of boron in moderator			
Primary heat transport:				
Number of loops	2	6	2	5
Primary coolant	H_2O	H_2O	H_2O	H_2O
Reactor inlet temperature (°C)	275	267	277	247
Reactor inlet pressure (MPa)	6.75		7.1	4.3
Reactor outlet temperature (°C)	281	270	283.5	263

TABLE LXVII. (cont.)

Parameter	Nuclear power plant			
	SGHWR	Gentilly 1	Fugen	Cirene
Reactor outlet pressure (MPa)	6.5	5.7	6.95	4.3
Number of recirculation pumps	4	6	4	5
Motor/type	Glandless, wet stator		870 kW vertical motor, 3-phase	
Steam drums:				
Steam flow rate (t/h)	541	1546	910	270
Steam pressure at full power (MPa)	6.13	5.27	6.8	4.5
Containment:				
Type	Concrete of biological shield plus turbine hall	Cylindrical, prestressed concrete, shell elliptical dome	Elliptically domed steel cylinder (dome top and bottom)	Cylindrical steel shell with hemispherical upper portion
Diameter/height (m)			36/64	
Turbine:				
Type	Saturated steam	Saturated steam	Tandem compound four flow	Saturated steam
Steam flow rate (t/h)	541	1546	910	270
Steam pressure (MPa)	6.13	5.3	6.4	4.5
Steam temperature (°C)	279	266	279	

TABLE LXVII. (cont.)

Parameter	Nuclear power plant			
	SGHWR	Gentilly 1	Fugen	Cirene
Generator:				
Type	Synchronous, 3-phase, 13.8 kW	One single line generator	One synchronous generator, 3-phase	40 MW(e)
Power	100 MW(e)	3600 rpm, 268 MW(e)	3600 rpm	3000 rpm
Main condenser:				
Coolant water flow rate (m^3/s)			9.72	

TABLE LXVIII. ARGENTINIAN PRESSURE VESSEL HWRs AND SIEMENS MZFR DESIGN: MAIN PARAMETERS

Parameter	Nuclear power plant		
	Atucha 1	Atucha 2	MZFR
General:			
Thermal reactor power (MW(th))	1179	2160	200
Gross generator power (MW(e))	367	744.7	57
Net plant power (MW(e))	345	693	50
Reactor:			
Coolant and moderator fluid	Pressurized D_2O	Pressurized D_2O	Pressurized D_2O
Number of channels in core	252	451	121
Number of fuel assemblies in core	252	451	242
Core length (m)	5.25	5.30	3.67
Uranium load (m)	39	85.1	14.2
Total number of shutdown and control rods	29	18	17
Number of loops of primary cooling system	2	2 primary, 4 moderator	2
D_2O coolant and moderator pressure (MPa)	11.3	11.5	8.45
Primary coolant flow rate through core (kg/s)	6139	10 300	78
Primary coolant's core inlet and outlet temperatures (°C)	265/299	279.9/312.3	251/280
Channel lattice	Triangular	Triangular	Hexagonal
Lattice pitch (mm)	272	272	272
Fuel assemblies:			
Fuel	UO_2 pellets	UO_2 pellets	UO_2 pellets

TABLE LXVIII. (cont.)

Parameter	Nuclear power plant		
	Atucha 1	Atucha 2	MZFR
Uranium	Initially, all natural uranium; nowadays loading 0.85% enriched uranium		All natural uranium
Present SEU load (1998)	40% of core load		
Full SEU load	Planned for being attained in 2002		
Fuel rods/structural rods, per fuel assembly	36/1	37	37
Fuel cladding	Zircaloy 4	Zircaloy 4	Zircaloy 2
Outer fuel rod diameter (mm)	11.9	12.9	
Active fuel rod length (m)	5.3	5.3	1.8
Average linear heat rating (W/cm)	232	232	116
Fuel burnup at equilibrium (MW·d/Mg)	5700–6000	7500	5000
Steam generators:			
Number	2	2	2
Thermal power per unit (MW(th))	525	1080	100
Number of tubes per unit	3945		189
Tube material	I800	I800	Stainless steel (10CrNiNb189)
Main primary pumps:			
Number	2	2	2
Nominal flow rate (kg/s)	3264	5150	1292
Nominal impulsion height (m)	123	135	

TABLE LXVIII. (cont.)

Parameter	Nuclear power plant		
	Atucha 1	Atucha 2	MZFR
Nominal power (MW(e))	4.2	9.1	
Turbine:			
Turbogenerator power (MW(e))	367	745	58
Speed (c/s)	50	50	50
Live steam flow rate (kg/s)	515.8	957	
Live steam pressure (MPa)	4.15	5.59	3.11
Live steam condition	Saturated steam	Saturated steam	Saturated steam
Number of inlets	3	3	2
Condenser pressure (kPa)	4.4	4.8	4.8
Average river water temperature at condenser (°C)	17	20	
River water flow rate at condenser (m^3/s)	17.36	38.40	4.78
Reactor pressure vessel:			
Inside diameter (m)	5.360	7.368	4.382
Shell thickness plus cladding (mm)	220	280 + 6	86/134/290
Overall height (m)	12.2	14.24	7.845
Containment	Steel sphere 50 m diameter/ concrete cylinder building	Steel sphere/concrete cylinder building	Concrete cylinder/steel shell/concrete cylinder building
Diameter/wall thickness (m)	56/0.30	56/0.30	

TABLE LXIX. PRESSURE VESSEL HWRs: SWEDISH DESIGNS' MAIN PARAMETERS

Parameter	Nuclear power plant	
	Ågesta PHWR	Marviken BHWR
Reactor vessel:		
Shell diameter/weight (mm)	4555/70	5220/76 + 5
Height (m)	5.0	23.2
Containment	Steel plate lined rock	Prestressed concrete cylinder
General:		
Thermal reactor power (MW(th))	65	463/593 (superheated)
Gross generator power (MW(e))	10	138/200
Net plant power (MW(e))	10	132/193
Reactor:		
Coolant and moderator	D_2O	D_2O (180 t)
Number of channels in core	140	147 + 32 (superheat)
Lattice pitch (mm)	270 (square)	250 (square)
Number of fuel assemblies	4 × 140	
Active core length (m)	0.362	4.42
Active core diameter (m)	0.3	4.3
Uranium load (UO_2) (t)	18.5	26.3 + 7.3 (superheat)
Number of shutdown and control rods	32/18 working rods	24 + 16
Number of loops in primary cooling system	4	2
Coolant pressure (MPa)	3.24	4.85
Primary coolant flow rate through core (kg/s)	1180	1840 (water), 215.1 (steam)
Primary coolant's inlet and outlet temperatures (°C)	205/220	120/259
Fuel assemblies:		
Fuel	Natural uranium	Enriched $^{235}UO_2$
Enrichment	Natural	~1.35% (boiler elements), 1.75% (superheater elements)
Fuel rods/structural rods, per assembly	19	36 (45 superheated channels)
Fuel cladding	Zircaloy 2	Zircaloy 2 (Incoloy superheat fuel)
Outer fuel rod diameter (mm)		12.5 (11.5 superheat fuel)

TABLE LXIX. (cont.)

Parameter	Nuclear power plant	
	Ågesta PHWR	Marviken BHWR
Active fuel rod length (mm)	300	600 (400 superheat)
Average linear heat rating (W/cm)	33	
Burnup (MW·d/t)	2800	13 000
Reactivity control rods:		
Safety rods		24
Regulating rods		16
Steam generators:		
Number	4	None—reactor operating in open cycle
Thermal power per unit (MW(th))	17	
Number of tubes per unit	2000 (U-tube)	
Tube material	Stainless steel	
Main primary pumps:		
Number	4	Natural circulation
Nominal flow rate (kg/s)	255	
Nominal impulsion pressure (kPa)	365.4	
Turbine:		
Turbogenerator power (MW(e))	12	200
Speed (c/s)	50	50
Live steam flow rate (kg/s)	30 (saturated)	215
Live steam pressure (MPa)	1.37	4.7 (4.1 with superheat)
Live steam temperatures (°C)	196/215	259 (472 with superheat)
Number of inlets	1	1

TABLE LXX(a). GAS COOLED HWRs: EL 4 AND NIEDERAICHBACH

Parameter	Nuclear power plant: El 4	Nuclear power plant: Niederaichbach
Reactor:		
Thermal output (MW(th))	250	316
Generator output (MW(e))	73	106.4
Net electrical power (MW(e))	70	100.4
Containment	Concrete cylinder prestressed	Steel plates plus concrete
Reactor vessel:		
Vessel	Horizontal cylinder	Steel cylinder 24 m diameter
Calandria diameter/length (m)	0.48/0.55	5.2/6.14
Number of fuel channels	216	351
Fuel channel diameter/length (m)	0.107/13	0.119/12.7
Fuel channel material	Zircaloy 2	Zircaloy 2
Lattice pitch (mm)	23.5 square	24.5 square
Fuel channel inlet temperature (°C)	260	252
Fuel channel outlet temperature (°C)	500–515	550
Active core length/diameter (m)	4.24	0.43
Coolant	CO_2	CO_2
Inlet pressure (MPa)	5.88	5.88
Outlet pressure (MPa)	5.10	5.30
Number of shutdown/control rods	9/16	N/A (moderator level)
Uranium load (UO_2) (t)	14.5	45.9
Number of loops in primary coolant system	2	2
Coolant flow rate through core (kg/s)	880	422
Fuel assemblies:		
Fuel	UO_2	UO_2
Enrichment	1.37% and 1.65%	1.15%
Fuel rods/structural rods/assembly	19	19
Fuel cladding	Stainless steel/ zirconium–copper	Stainless steel
Fuel rod outside diameter (mm)	11	15
Active fuel rod length (m)	1.075	1.075
Linear heat rating (W/cm)	25.5	
Burnup (MW·d/t)	12 000	11 600

TABLE LXX(a). (cont.)

Parameter	Nuclear power plant	
	El 4	Niederaichbach
Reactivity control:		
Shutdown	B_4C rods	Moderator dump
Regulating	Stainless steel shim rods, 4 stainless steel regulating rods	Cadmium sulphate in moderator level
Steam generators:		
Number	2	2
Thermal power per unit (MW(th))	125	6
Tube material		Stainless steel
Main primary pumps:		
Number	3	2
Nominal flow rate (kg/s)	880	844
Nominal power (MW(e))	7	6
Turbine:		
Turbogenerator power (MW(e))	74.3	106.4
Speed (c/s)	50	50
Live steam flow rate (kg/s)	91	102.8
Live steam pressure (MPa)	6.70	10.49
Live steam temperature (°C)	490	530
Condenser pressure (kPa)	3.5	26.8
Containment	Prestressed concrete cylinder	Steel cylinder, 24 m diameter with spherical head: total height 43.5 m, cylindrical height 31.5 m, thickness 13 mm

TABLE LXX(b). GAS COOLED HWRs: KS150 AND LUCENS

Parameter	Nuclear power plant KS150	Lucens
Reactor:		
Thermal output (MW(th))	590	30
Generator output (MW(e))	150	10.4
Net electrical power (MW(e))		7.6
Containment	Steel vessel plus concrete	Rock cavern plus reinforced concrete
Reactor vessel:		
Vessel (m)	19 × 5 diameter carbon steel	Zircaloy 2 tube shape
Calandria diameter/length (m)	4.5 × 4.3 aluminium alloy	3.13/3.65
Number of fuel channels	196 (156 fuelled)	73 + 1
Fuel channel material	Aluminium	Zircaloy 2
Lattice pitch (mm)	Square	24 (inner zone), 29 (outer zone) square
Fuel channel inlet temperature (°C)	105	223
Fuel channel outlet temperature (°C)	425	378
Active core length/diameter (m)	4/4.16	2.905
Coolant	CO_2	CO_2
Inlet pressure (MPa)	6.47	6.08
Outlet pressure (MPa)	5.59	5.60
Number of shutdown/control rods	32/8 (in channels)	10 + 4
Uranium load (t)	25.4	4.6
Number of loops in primary coolant system	1	2 (in series)
Coolant flow rate through core (kg/s)	1600	2.10 per fuel element
Fuel assemblies:		
Fuel	Natural uranium metal	U–0.1%Cr(^{235}U, 0.96%)
Enrichment	Natural uranium (0.7%)	0.96% metal
Fuel rods/structural rods/assembly	150–200 rods/ assembly	28 rods (4 × 7, 4 assemblies per channel)
Fuel cladding	Beryllium–magnesium	Mg–0.6%Zr (finned)
Fuel rod	4 mm diameter/4 m long	20.5/1.75 (fins 31.5)
Active fuel rod length (m)	4.0	
Burnup (MW·d/t)	3000	3000

TABLE LXX(b). (cont.)

Parameter	Nuclear power plant	
	KS150	Lucens
Steam generators:		
Number	3	2
Thermal power per unit (MW(th))	50	15
Main primary pumps:		
Number	6	2
Nominal flow rate (kg/s)	422	
Nominal power (MW(e))	4.65	0.885
Turbine:		
Turbogenerator power (MW(e))	150	12
Speed (c/s)	50	50
Live steam flow rate (kg/s)	1600	10.83
Live steam pressure (MPa)	2.84	2.23
Live steam temperature (°C)	400	367
Condenser pressure (kPa)	4.1	5.0
Water flow rate into condenser (m^3/s)		0.756

TABLE LXXI. INDIAN AHWR

Parameter	Value
Reactor:	
Thermal output (MW(th))	750
Gross electrical output (MW(e))	245
Net electrical output (MW(e))	235
Calandria vessel:	
Form	Cylinder welded to end shields
Material	Austenitic stainless steel type 304L
Inside diameter/height (m)	8.6/5.0
Wall thickness (mm)	50
Moderator	D_2O plus pyrocarbon
Calandria end shields:	
Material	Austenitic stainless steel type 304L
Fill	Water and steel balls
Calandria tubes:	
Material/number	Zircaloy 2/424
Fuel channels:	
PT inside diameter/wall thickness (mm)	120/4
PT material	Zr–2.5%Nb
Number of PTs	424
Coolant flow rate (kg/s)	2576
Maximum channel power (MW(th))	2.3
Fuel:	
Bundle length/outside diameter (m)	4.027
Fuel material	$(Th,Pu)O_2$ plus $(Th,^{233}U)O_2$
Sheath outside diameter/wall thickness (mm)	11.2/0.6
Sheath material	Zircaloy
Elements per bundle	52
Fuel bundles in channel	1
Fuel bundles in core	424 (320 channels with 44 $(Th,^{233}U)O_2$ and 8 $(Th,Pu)O_2$ pins) (84 channels with ThO_2 pins)
Reactivity control units:	
Shutdown devices	Absorber rods in B_4C Liquid poison injection of lithium pentaborate into moderator

TABLE LXXI. (cont.)

Parameter	Value
Reactivity control devices	Control rods Grey control rods Water displacer rods Boron concentration in moderator Moderator level control
Heat transport:	
Number of loops	4
Primary coolant	H_2O
Reactor inlet temperature (°C)	270.7
Reactor outlet temperature (°C)	285
Reactor pressure (MPa)	7.0
Steam drum number/material	4/stainless steel lined carbon steel
Primary containment:	
Form	Cylindrical double shell
Diameter/height (m)	44/72
Turbine:	
Number/type	1 horizontal/impulse reaction
Turbine sections	1 HP, 1 LP
Speed	3000 rpm
HP inlet pressure (MPa)	68
HP inlet temperature (°C)	284
Generator:	
Type/number	Static, exited, stator and rotor core hydrogen cooled and stator windings water cooled
Power	275 MV·A
Frequency (Hz)	50
Condenser:	
Type	Surface condenser
Number of tubes	21 193 (18 BWG)/413 (BWG)
Cooling water flow (m^3/s)	17
Condenser pressure (kPa)	8.4

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