Fundamentals of Power Reactors

Module Two
Nuclear Reactor Systems

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Site Selection

Training Objectives

On completion of this lesson for a CANDU Station the participant will be able to:

- Name the site characteristics which are important for design safety
- Identify the three phases in a typical site selection process
- Identify the natural site factors affecting design safety
- Identify the human-related site factors affecting design safety
- Name the site characteristics which are important in determining the impact of the plant in the region in which it could be located.

Table of Contents

1. Overview of Site Requirements for CANDU 6 Nuclear Power Plants

The siting of Nuclear Power Plants differs from that of conventional plants in that the safety and protection of the public and the environment against the radiological impact of normal and accidental releases of radioactivity must be taken into account and are considered of primary significance in the site selection.

The site characteristics which are important to plant design, and the site characteristics which, when the plant is operational are important in terms of occupational and regional public safety, must be investigated. This means that considerably greater effort must be taken in Nuclear Station site selection than for a standard large industrial plant or a conventional fossil fired station.

A large number of Codes and Standards by Environment Canada, Canadian Standards Association (CSA), Atomic Energy Control Board (AECB) of Canada, U.S. Nuclear Regulatory Commission (NRC) and International Energy Agency (IAEA) are available which address in detail the various aspects of site selection for nuclear power plants.

A summary of the more significant aspects of a nuclear plant site selection is given here.

Space and Water Requirements

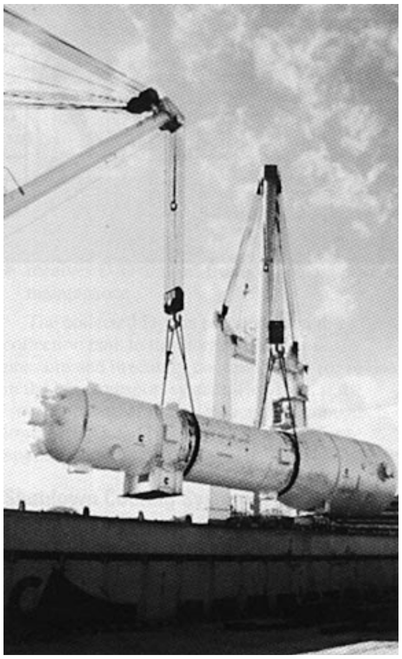
The space requirements for the plant are determined by factors such as, the number of units intended for the site, exclusion zone requirements, construction methods, waste storage technology and transportation methods.

The site, if possible, should be selected to accommodate future units. This allows the extensive cost and schedule of site selection activities to be spread over a large amount of power generation. It is also easier, from the point of view of public acceptance to site further units on an existing site.

A modular construction concept is used in the construction of new stations. On-site and off-site module fabrication are both possible. For on-site module fabrication the selected site should accommodate a fabrication facility area of 50 x 100 m. For off-site module fabrication, the space and transportation access should be adequate for the shipping of large size modules to the site. Certain major pieces of equipment such as the reactor structure (calandria), the steam generators and fuelling machines require special technical knowledge in their production and will be built off-site on the suppliers premises. The site must therefore be suitable for shipment of these major components. Figure 1.1.1 shows a steam generator being shipped overseas.

Figure 1.1

Loading a Steam Generator for Shipment Overseas



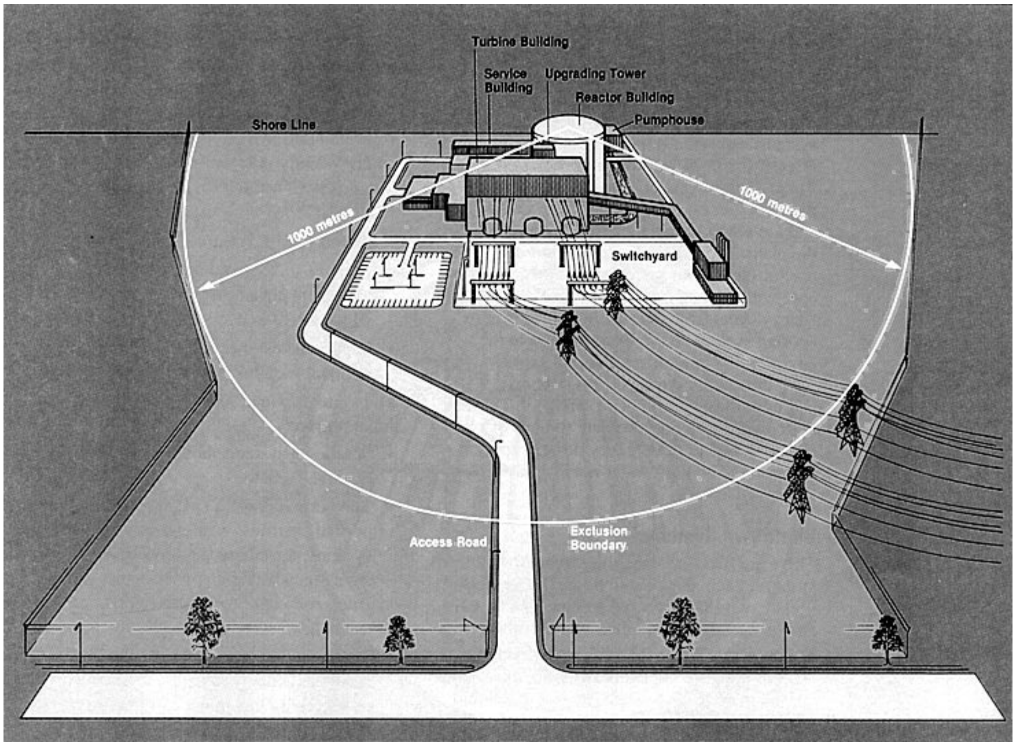
Exclusion Zone

The exclusion zone, in which no permanent habitation is allowed, is presently chosen as a circle of 1.0 km with its centre at the centre of the reactor building, (Figure 1.1.2). Radiation protection design criteria will then ensure that individual doses within the exclusion zone and individual and population doses outside the exclusion zone are within the legal limits set by AECB.

Radioactive Waste Storage Requirements

Space is required at the site for waste storage facilities for radioactive waste produced in the lifetime of the plant. The material requiring storage may range from highly radioactive irradiated fuel to material of low radioactive level such as used clothing, tools, debris, etc. CANDU stations currently provide water pool storage facilities for the irradiated fuel for a number of years of operation. For longer term storage a dry storage method in which irradiated material is placed within above-ground concrete containers is utilized by some of the existing CANDU stations.

Figure 1.2:
Typical CANDU Layout Showing Exclusion Zone



Cooling Water Requirements

Present CANDU stations use an open circuit cooling water system using sea water, lake water, or river water as the coolant for the turbine steam condenser. If adequate water from such sources is not available, a design with a closed circuit cooling water flow, employing cooling towers, can be used to reduce the cooling water flow to that required for make-up of loss from the system.

2 First Stage of Site Selection Process

The first stage of three stages of site selection is the site survey which leads to identification of one or more preferred candidate sites from consideration of both safety and non-safety factors. The phases involved in this stage are:

Phase 1 -

Regional analysis to identify potential sites. This phase consists of simple rejection criteria based upon information which is readily available such as cooling water availability, population density, surface faulting, volcanic activity, seismicity.

Phase 2 -

Screening of potential sites to select candidate site. This phase will involve visits to potential sites and application of simple comparison methodologies.

Phase 3 -
Screening, comparison and ranking of candidate sites to obtain the preferred candidate site. This phase requires limited field work and use of sophisticated scaling and comparison techniques.

During the site survey stage, disciplines such as the following are required:

- Power Engineering
- Nuclear Engineering
- Radiological Protection
- Ecology/Radioecology
- Demography
- Emergency Planning
- Seismology
- Soil Mechanics
- Geology
- Hydrology
- Meteorology
- Civil Engineering

Much of the expertise in the above disciplines is required for siting conventional plants. However a more specialized level of expertise is required in nuclear plant siting.

A well balanced site survey team should include a nuclear engineer, a geologist with seismic expertise, a soil mechanics engineer, a hydrologist, a meteorologist, and an expert on siting.

Starting with this stage, all screening data which will be necessary for safety purposes should be subjected to a Quality Assurance (Q.A.) program as specified in the applicable codes.

For each site, the availability of labour and the existence of infrastructures such as housing, airports, schools, etc. should also be considered as they affect the construction cost and schedule.

3. Site Parameters Affecting Design Safety

There are two major classifications of the external factors (that is, factors not stemming from the plant itself), which must be considered in siting:

1. **Natural Factors; included in this category are:**
    - Surface faulting
    - Seismic Conditions
    - Subsurface (Foundation) Stability
    - Flooding
    - Climactic Conditions
    - Blockage of Cooling Water Sources

2. **Human-Induced Events; included in this category are:**
   - Aircraft Crashes and Impact of Missiles
   - Chemical Explosions
   - Release of Hazardous Materials

4. Site Selection - Natural Factors

Surface Faulting

The site should not be close to known capable geological faults. Proximity to known capable faults imposes extensive analytical and licensing efforts if a decision is made to use the site.

Seismic Conditions

For nuclear stations the seismic requirements that must be met are much more rigorous than those given in building codes. Areas of high seismic activity should be avoided. Site specific seismicity studies are required for the selected site. These studies include historic earthquake records, regional tectonics and seismological characteristics and micro-earthquakes monitoring. Using these data, probability analysis will be carried out in order to determine the Design Basis Earthquake at the site. These studies will have to be carried out regardless of the earthquake zoning defined in the National Building Code for the area. Standard plants such as CANDU 6 are designed for a defined maximum ground acceleration which will meet the requirements of most Canadian and overseas sites.

The safety objective of the seismic design of the plant is to have sufficient capability to perform the essential safety functions following a Design Basis Earthquake. This means that certain systems must be seismically qualified. The safety functions that must be provided for are:

1. The ability to shut the reactor down and maintain it in a safe shutdown state.
2. The ability to remove residual heat.
3. The ability to maintain a barrier to limit the release of radioactive material.
4. The ability to perform essential safety-related control and monitoring functions.

Included in this requirement is that there be a seismically qualified lighted route from the Main Control Room to the Secondary Control area. The important control functions that must be maintained are

- the ability to initiate Shutdown System 1 (SDS1) and Shutdown System 2 (SDS2),
- the ability to initiate controlled cooldown using the steam generators, the feedwater system, followed by manual initiation of the Shutdown Cooling System,

- the ability to initiate the necessary portions of Emergency Core Cooling (ECC) system,
- the ability to initiate containment isolation,
- the ability to isolate certain systems not qualified to the Design Basis Earthquake standard, if their failure would jeopardize qualified systems.

There are several CSA seismic standards applicable to CANDU stations.

Subsurface (Foundation) Stability

The following aspects need to be investigated in this regard:

1. geological and subsurface conditions,
2. static and dynamic elasticity,
3. liquefaction potential should be avoided
4. feasible foundation types,
5. bearing capacities and settlement ranges,
6. ground water levels,
7. possible geological hazards such as slope instability, surface collapse, subsidence or uplift.

Standard plants are able to accommodate a range of site conditions from relatively soft rock to very hard rock (soil/rock shear modules of 5000 to 100,000 kg/cm^2) and solid bearing pressure of 10 kg/cm^2 and higher. It also assumes the competent rock to be shallow (2 m deep). In sites where competent rock is very deep, foundation designs using rafts, piles and caissons can be used.

Flooding

CANDU stations must be guarded against floods by providing adequate diversion and protection facilities. The following need to be investigated in this regard:

1. Estimation of Probable Maximum Flood conditions.
2. Comparison of drainage area vs peak flow
3. Records of historic floods
4. Flood due to precipitation and due to failure of water control systems
5. Water velocity
6. Floating debris
7. Sedimentation and erosion

Climactic Conditions

CANDU station buildings are designed for ambient temperature ranging from extreme winter cold to extreme summer warm temperatures.

Sites less affected by cyclones and tornados are preferred. The design basis tornado however, should consider the rotational and translational wind speed, radius of maximum rotational wind speed and pressure differentials and rate of change of pressure. Tornado generated missiles should also be considered. This is discussed more fully in another lesson. Standard plants such as CANDU 6 satisfy the tornado requirements of Canadian sites.

Blockage of Cooling Water Source

This is an extremely important consideration as water must be available at all times for heat removal from fuel. Cooling water may become unavailable for a variety of reasons - river blockage and diversion, reservoir depletion, reservoir or cooling water blockage through freezing and ice formation, ship collision, oil spills and fires. The probability of occurrence of these events should be investigated for probability of occurrence and effect. A recent concern in siting on the Great Lakes in Ontario is the blockage of intake piping or structures by zebra mussels.

5. Site Selection Human - Related Factors

Aircraft Crashes and Missile Impact

The probability of an aircraft impacting on critical areas of the plant should be low enough to pass the licensing requirements or the design should accommodate the resulting conditions. In general, sites in proximity to airports should be avoided. The likelihood of nearby facilities being the source of missiles that can impact the plant and affect the safety must be examined.

Chemical Explosions

The probability and effect of explosions from industrial facilities and pipelines should be examined and shown to be acceptable.

Release of Hazardous Materials

Facilities that process,. store, transport or handle toxic, corrosive or radioactive materials which, if released, could adversely affect plant safety must be examined. An example of this is the presence on the Bruce site of a heavy water plant. This plant contains approximately 800 tons of H_2S and an exposure of 1000 ppm for 10 minutes to this gas is fatal. Sites should avoid as far as possible such potential hazards, but if this cannot be done then provision must be made for operator, plant and public safety in case of a major release of the toxic material.

6. Site Characteristics Influencing Plant Impact on the Region

There are site characteristics which, when the plant is operational, influence the impact on the region. These are:

- atmospheric, surface and ground water characteristics,
- population distribution,
- use of land and water in the region.

Atmospheric and Water Dilution

Gaseous effluent dispersion and liquid effluent dispersion in receiving water should be determined in order to establish valid effluent release limits (or targets). In the absence of actual atmospheric dilution values, standard dilution factors may be used. If this approach is taken, then an appropriate degree of conservatism must be used in the selection of dilution factors. This practice is fully described in module 3 lesson 5.

In order to take advantage of liquid dispersion, e.g. when discharge is into a river, dye dilution studies may have to be carried out. This was done for the first Canadian power stations NPD and Douglas Point. If there is a municipal or other drinking water plant drawing water from the discharge plume, the dilution factor from plant discharge to the intake of the drinking water should be determined, even although it may not be intended to take advantage of lake flow or river flow dilution capacity in establishing release limits.

Site hydrology should be investigated. This is necessary if it is intended to place any radioactive waste, for example low level waste, in in-ground facilities. This ensures knowledge of water drainage patterns and flows and permits appropriate monitoring sites to be selected.

Population Distribution

For accidents associated with postulated failure conditions there are prescribed population dose limits. The integrated dose received by the population around the plant must be kept below these limits. To comply with this requirement, sites with surrounding low population densities are more desirable than sites with high population densities (U.S Regulatory Guide 4.7 identifies high population density as more than 500 person per square mile at initial operation and 1,000 persons per square mile projected at the end of the plant lifetime within a radius of 50 km). The immediate area of the plant should be investigated for the presence of special institutions such as schools, hospitals, prisons, etc. Population density and special institutions affect the extent and complexity of emergency plans required by provincial authorities.

Use of Land and Water in the Region

The effect of the plant on the region requires the investigation of land and water usage for agricultural products, dairy farming, fishing, commercial and residential, wildlife and sports. The ambient radioactivity of the site and surrounding district should be determined prior to commissioning so that the impact of the eventual operation of the plant on the region may be determined .

Sources of Figures

Figure No.	Source
1.1	Technical Summary CANDU Nuclear Generating Station, 901212PA Page 24
1.2	Technical Summary CANDU Nuclear Generating Station, 901212PA Page 39

Station Layout Safety Considerations

Lesson Objectives

On completion of this lesson the participant will be able to:

- Describe the development of the CANDU power reactor and be able to identify the early power reactors in Canada
- State the main components in the layout of a CANDU 6 station - the NSP, the NSPS and BOP
- Describe the difference in layout of a CANDU 6 and an Ontario Hydro station.
- State at least three ways in which radiation doses to the public are kept low by design action
- State how a CANDU station compares with reactors in other countries in terms of collective dose per unit of electrical production.
- Describe the grouping and separation philosophy employed for systems in a CANDU station and know how this is achieved.
- Describe the precautions taken against damage from tornado and missiles in a CANDU station.
- State the basis of fire protection in a CANDU station.

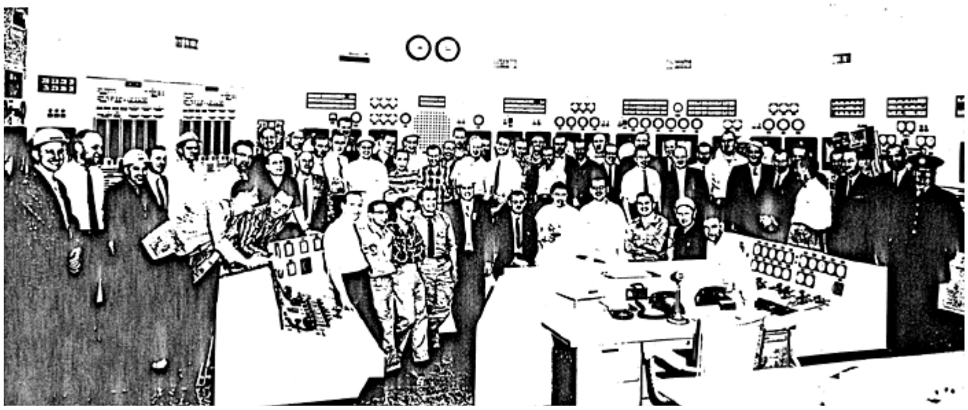
Table of Contents

1. Historic Development of the CANDU System

The development of reactor types is closely related to nuclear weapons development work carried out by various countries in the second world war. The United States which took the major role in this development in the Western world had at the end of the war the facilities to produce uranium enriched in uranium-235 and so pursued the development of reactors using enriched fuel and ordinary water as moderator. The United Kingdom and Canada cooperated in their war-time activities and established a joint laboratory - the Montreal Laboratory. One of the activities of this laboratory was to investigate and carry out preliminary design studies on a reactor for the production of plutonium using natural uranium with heavy water as a moderator. A supply of heavy water had been spirited out of France prior to its occupation by German forces, to England. This heavy water was subsequently transferred to the Montreal laboratory. In addition heavy water was being produced at Trail B.C. under contract to the U.S. The U.K. eventually developed reactors using natural uranium with a graphite moderator and Canada developed reactors using natural uranium with heavy water as moderator. A decision was made in 1944 to build a reactor of this type - the NRX reactor and a site was selected at Chalk River. A simple zero power reactor, known as ZEEP, which used heavy water as moderator was built at Chalk River to study the size, shape and composition of uranium fuel rods and their distance apart. ZEEP went critical in 1945. By 1946 the staff of the Montreal Laboratory gradually moved to Chalk River. NRX was followed at Chalk River by a second natural uranium-fuelled, heavy water moderated, research reactor - NRU, and from this beginning the conception of the CANDU nuclear power emerged. Experienced gained in the operation of NRU and NRX, especially that resulting from the accident to NRX, has provided the basis for the safety principles of the CANDU reactor.

The first power reactor was the NPD reactor. The initials standing for Nuclear Power Demonstration. This was a joint venture by Atomic Energy of Canada Ltd. and Ontario Hydro. This 20 Mw(e) reactor was built at Rolphton about twelve miles up the Ottawa river from Deep River the townsite for Chalk River. It was owned by AECL and operated by Ontario Hydro. NPD achieved criticality in April 1962 and operated successfully until it was closed down in 1987. The Douglas Point station followed on NPD. This was a 200 Mw(e) CANDU reactor built on an Ontario Hydro site near Tiverton on Lake Huron. The reactor achieved criticality in 1967 and operated until 1984 when it was taken out of service. It provided valuable design and operational experience during its service.

NPD Start Up April 1962



After Douglas Point there was a divergence in the design of the CANDU system. Ontario Hydro concentrated on the design and construction of four-unit stations starting with Pickering which started to produce power in 1971. This was followed by the Bruce "A" and "B' stations, Pickering "B' and most recently Darlington. AECL concentrated on the development of the single unit 600 Mw(e) CANDU, which has become known as the CANDU 6. Units of this type have been built in Canada at Point Lepreau for New Brunswick Power, Gentilly for Hydro Quebec and overseas in Argentina and Korea. Earlier versions were built in Pakistan by General Electric Canada and in India by AECL. AECL is currently building a multi-unit CANDU in Romania. AECL has designs for, but has not yet built, units of 300 Mw(e) and 900 Mw(e) known as CANDU 3 and 9 respectively.

Design, construction and siting of Nuclear Power Plants differs from conventional plants in that the safety and protection of the public and the environment against the radiological impact of normal and accidental releases of radioactivity are of primary significance.

The plant design and site parameters, which under normal and accident conditions impact on public and occupational safety and the environment or the site region, must be investigated at a very early stage of the design process. This means that the provisions of National and Provincial building codes for non-nuclear facilities, although applicable, are not always sufficient and must be supplemented with additional nuclear codes and standards to comply with the nuclear requirements.

A large number of Codes and Standards of Environment Canada, the Canadian Standards Association (CSA), the Atomic Energy Control Board (AECB) of Canada, the U.S. Nuclear Regulatory Commission (NRC) and the International Energy Agency (IAEA) exist. These address in detail the many aspects of plant layout and site selection for nuclear power plants. The more significant safety aspects of a nuclear plant layout are discussed in this lesson. In this module the emphasis is on the AECL CANDU 6 but information is also included on Ontario Hydro multi-unit stations, mainly with reference to the Darlington station.

2. Typical Layout of a CANDU Station

The station layout of a typical CANDU 6 is shown in Figures 1.2.1A to 1.2.1D and for a multi-unit station (Darlington NGS) in Figures 1.2.2A to 1.2.2C. CANDU stations are generally divided into three main areas that are called:

1. Nuclear Steam Plant (NSP). This includes the reactor and its auxiliaries and boilers housed in the reactor building, the service building and reactor auxiliary building.
2. Nuclear Steam Plant Services (NSPS). This includes the control room, the spent fuel management areas, D_2O management areas and vapour recovery areas, laboratories, maintenance building and the secondary pumphouse.
3. Balance of Plant (BOP) is all the other services of a power plant which do not produce radiation or contain radioactive material. This includes the Turbine Building, the main pumphouse, the water treatment plant, intake and outfall structures.

This classification is not rigid and in some units such as the Bruce and Darlington multi-unit stations the NSP and NSPS components are not totally segregated.

NSP - Reactor Building Functional Requirements

The Reactor Building design objectives are:

- To house, support and protect the nuclear steam supply system components including the reactor, heat transport and safety systems contained within the reactor building.
- To withstand the specified normal and abnormal loading conditions, including the design basis earthquake and other external hazards such as flood and missiles.
- To accommodate and manage internal flooding.
- To prevent and minimize the spread of fire.
- To provide access to the systems, components and structures for installation, inspection, maintenance and replacement.
- To assist in keeping exposure of radiation to the public and operating personnel within acceptable values, during normal plant operation.
- To keep the release of radioactive material to the environment under postulated accident conditions within licensing limits.

Figure 1.2.1A
Typical CANDU 6 Service Building Plan

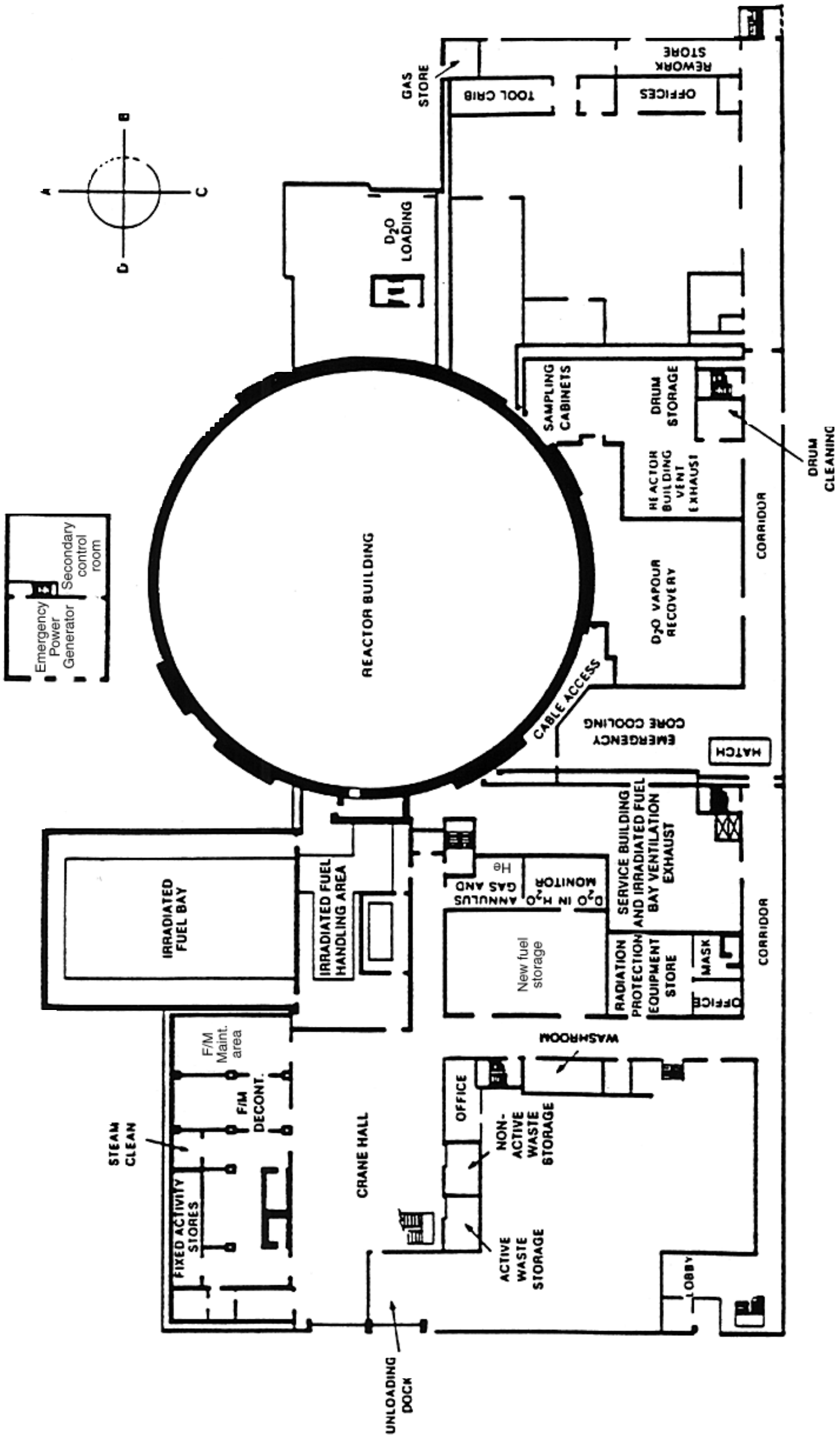


Figure 1.2.1B
Typical CANDU Reactor Building

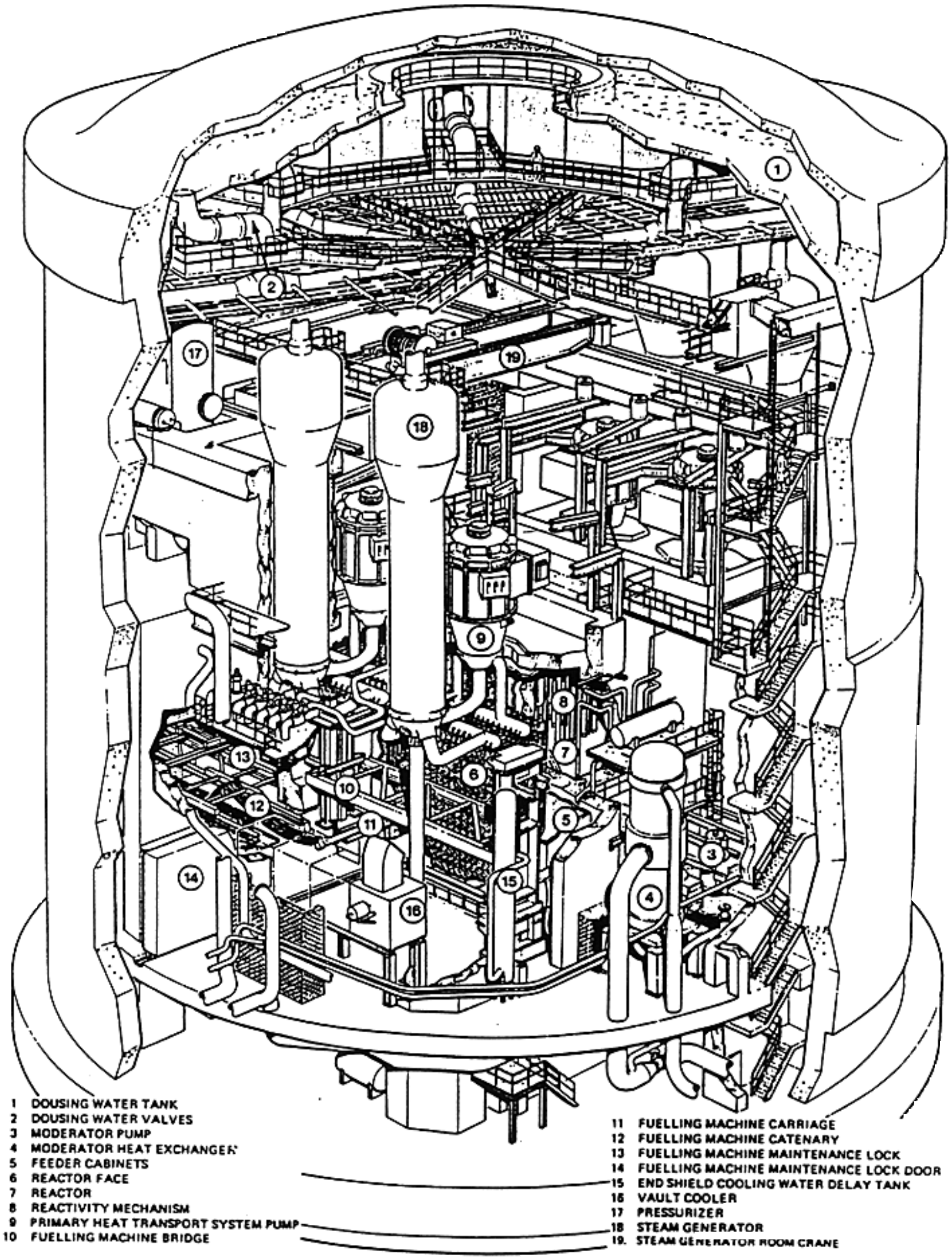


Figure 1.2.1C
Typical CANDU 6 Reactor Building Plan Grade Floor

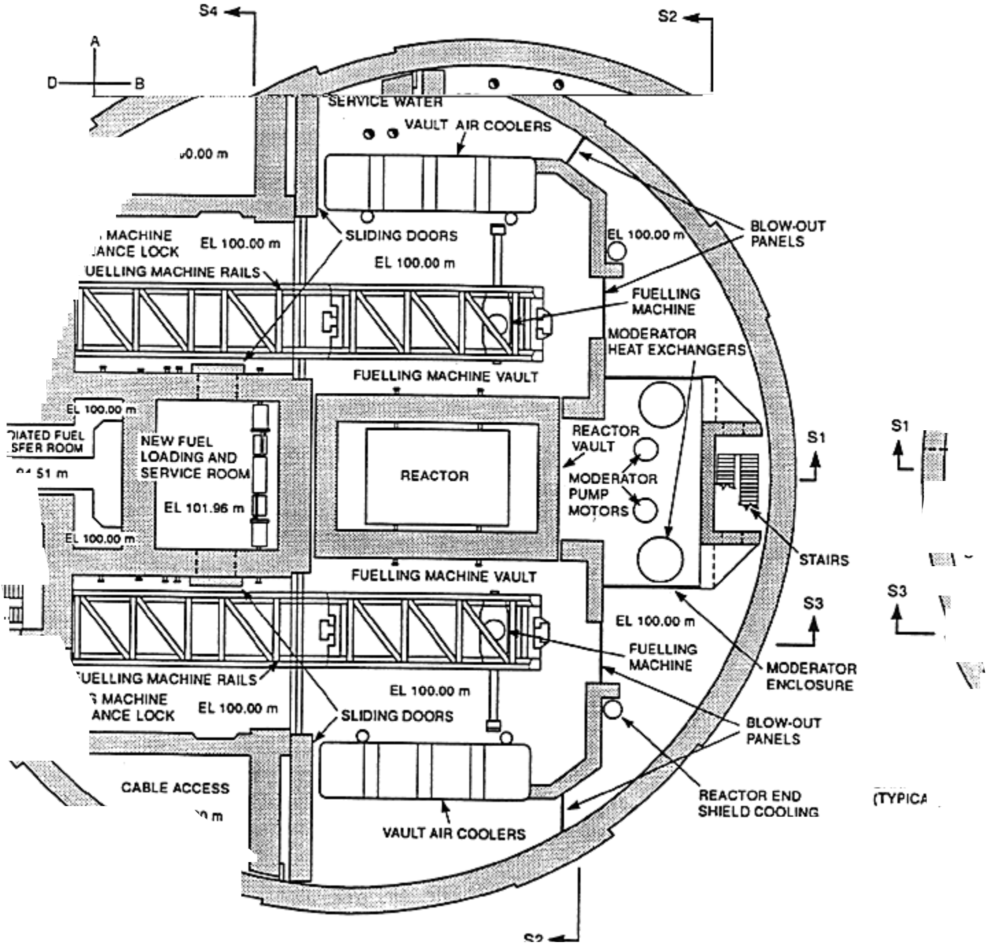
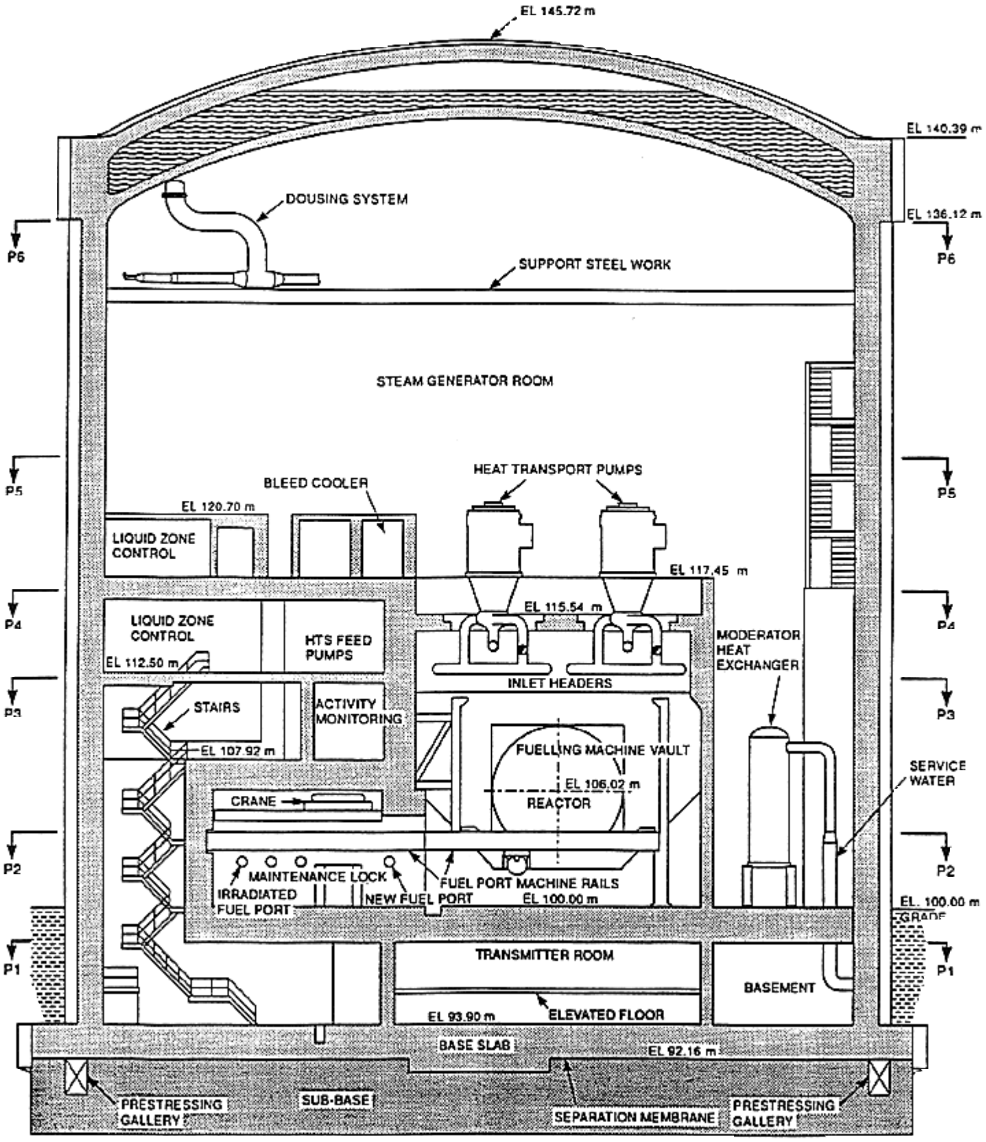


Figure 1.2.1D
Typical CANDU Reactor Building Section



NSP - Reactor Auxiliary Building (RAB)/Service Building Functional Requirements

Definition of this area varies depending on the station. On some stations such as the CANDU 6 this building includes the maintenance areas and is called Service Building (SB). However, regardless of the name used the design objectives of these structures are similar and can be described as follows:

- to house, support and protect the NSPS related systems, equipment and services (e.g. dryers, fuel transfer, etc.) as well as the distinct safety related systems and equipment such as emergency core cooling , secondary control area, emergency power),
- to withstand the specified normal and abnormal loading conditions, including the design basis earthquake and other external hazards such as flood and missiles,
- to accommodate and manage internal flooding,
- to prevent and minimize the spread of fire,
- to provide access to the systems, components and structures for installation, inspection, maintenance and replacement,
- to provide for the survival of operators in the main control room and enable them to reach the secondary control area during or following a design basis earthquake.

In future AECL nuclear station designs (CANDU 3 and CANDU 9) it is intended to separate physically and geographically power production and safety systems. For example a building provided for this purpose would contain a secondary control room, diesel generators, power supplies, and control equipment rooms to monitor control and bring the reactor too a safe shutdown. remove the decay heat of the fuel and maintain containment of any radionuclides released from fuel and reactor systems. These facilities are currently provided for in Reactor Auxiliary Buildings or Service Buildings although some may be in separate buildings. For example in the CANDU 6 design the secondary control room and the emergency power supply buildings are in separate buildings. At Darlington NGS the secondary control room is in the Reactor Auxiliary Bay and the emergency power supplies are in separate buildings.

Nuclear Steam Plant Services

Many of these facilities in current units are within the Service Building or the Reactor Auxiliary building as is the case at Darlington. An important safety related system/building that comes under the NSPS group of structures is the emergency water supply system and pumphouse. All CANDU stations have a separate cooling water supply available for essential cooling in the event that normal cooling water is not available. There is a variety of methods by which this is achieved - from having ponds with a pumphouse on the pond to pumphouses drawing water from the main cooling water source but located away from the normal pumping installations.

BOP Structures

Balance Of Plant structures consist of all other structures that are not part of NSP or NSPS, such as:

- the Turbine Building containing the steam-driven turbine generator set and its auxiliaries plus equipment such as the deaerator, the deaerator storage tank, the feedwater tanks and pumps, etc.,
- the Switchyard which accommodates the equipment and facilities to transmit the station electrical output to the utilities grid,
- the Water Treatment Plant which provides the station with water supplies of appropriate purity and chemical characteristics for use as domestic water and make-up water to various systems,
- the Main Pumphouse which supplies all normal station water requirements such as condenser cooling water and raw service water,
- the Administration Building which provides office and records space, cafeteria and first aid facilities,
- Guard House that provides a controlled site access point,

Figure 1.2.2A
Darlington Building/Site Arrangement

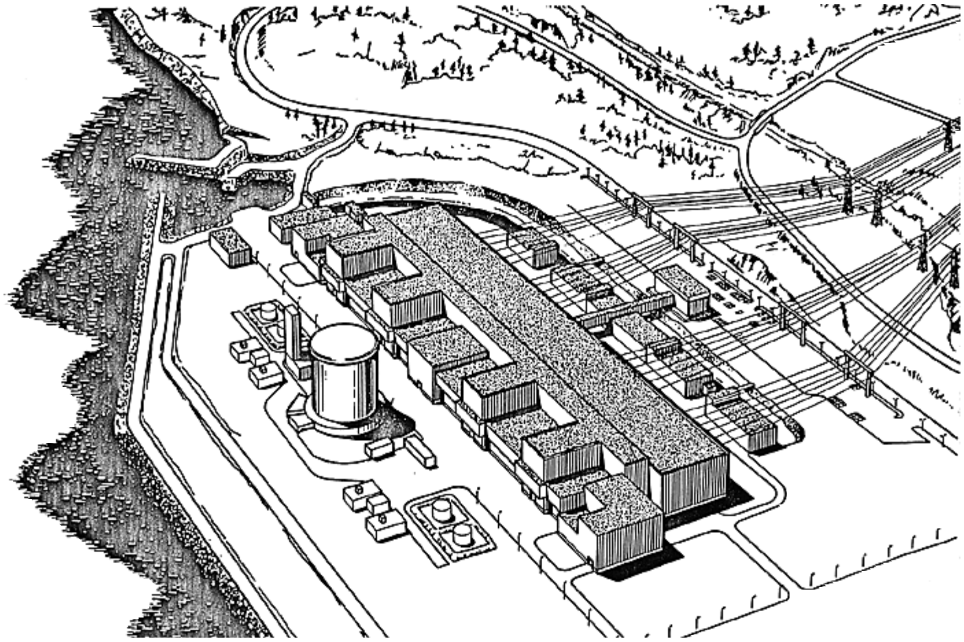


Figure 1.2.2B
Darlington Station General Arrangement

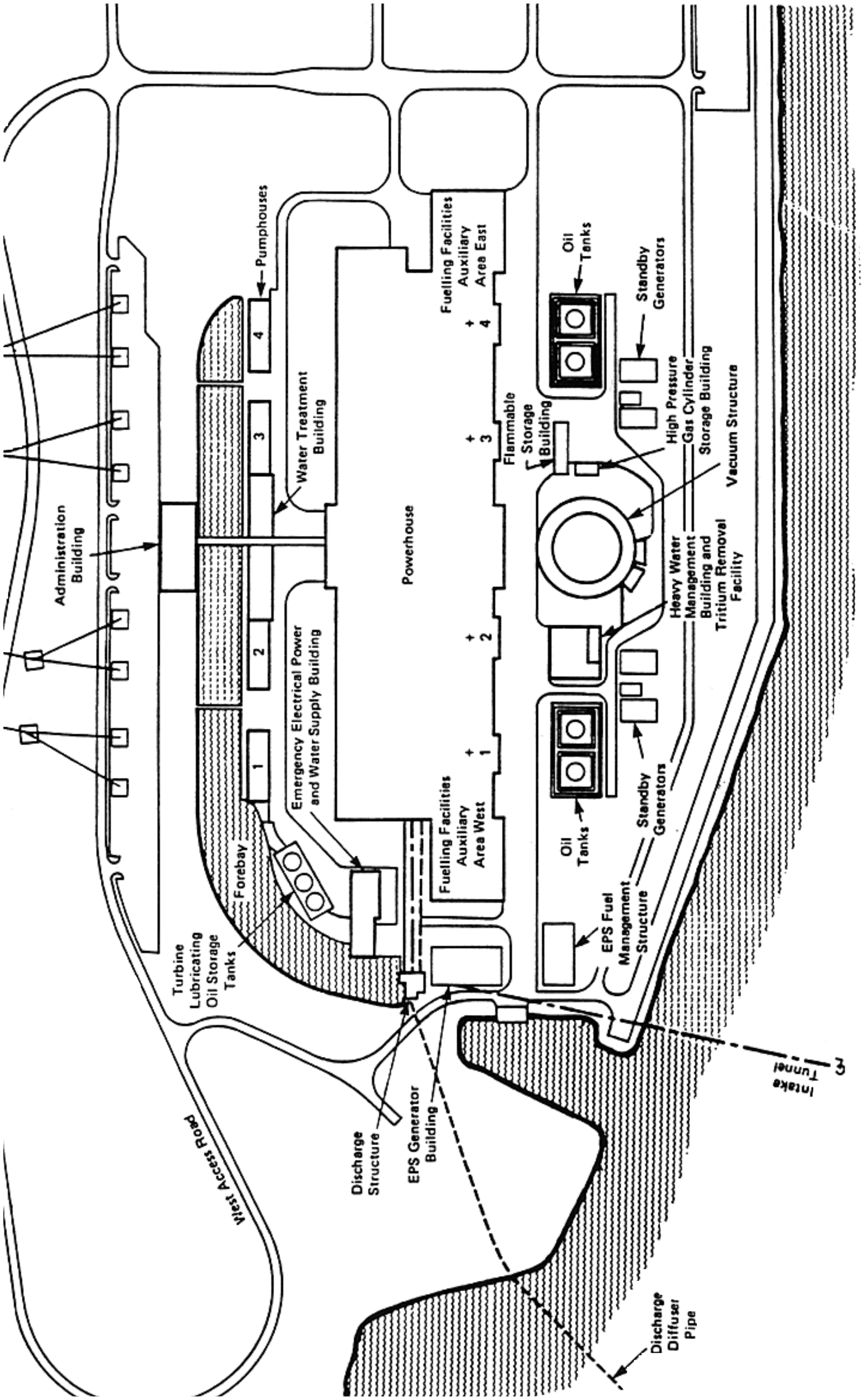
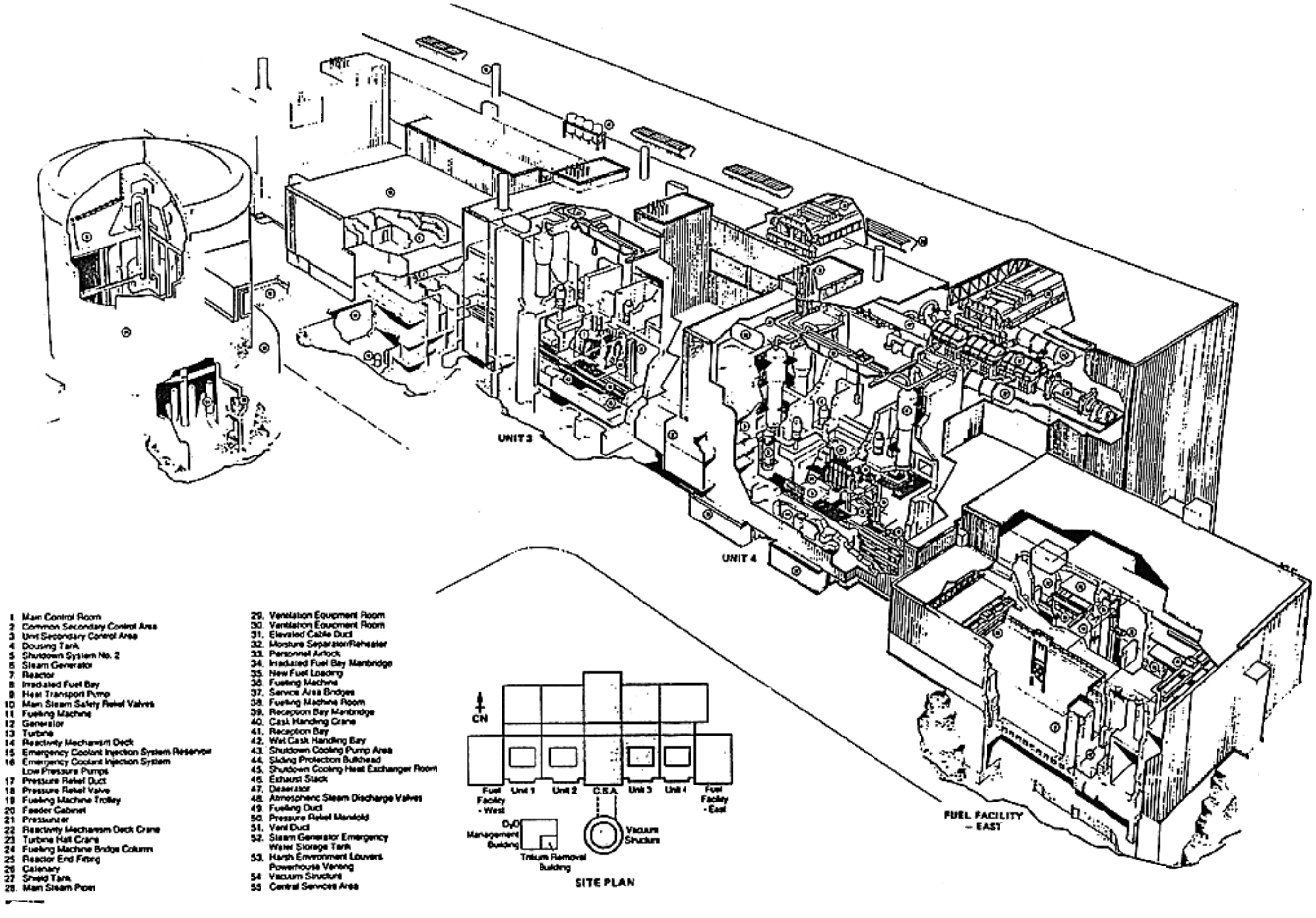


Figure 1.2.2C
Darlington Station Cutaway



3. Radiation Protection Considerations

Introduction

CANDU stations have established a noteworthy performance standard in both public and occupational radiation protection. Figure 1.2.3 shows the occupational doses to operate the Point Lepreau CANDU 6 and Figure 1.2.4 shows the Point Lepreau and Ontario Hydro stations dose in mSv/MW-y in comparison to other types of reactors in other countries. This performance is typical of CANDU stations.

Figure 1.2.3

Point Lepreau Annual Collective Dose

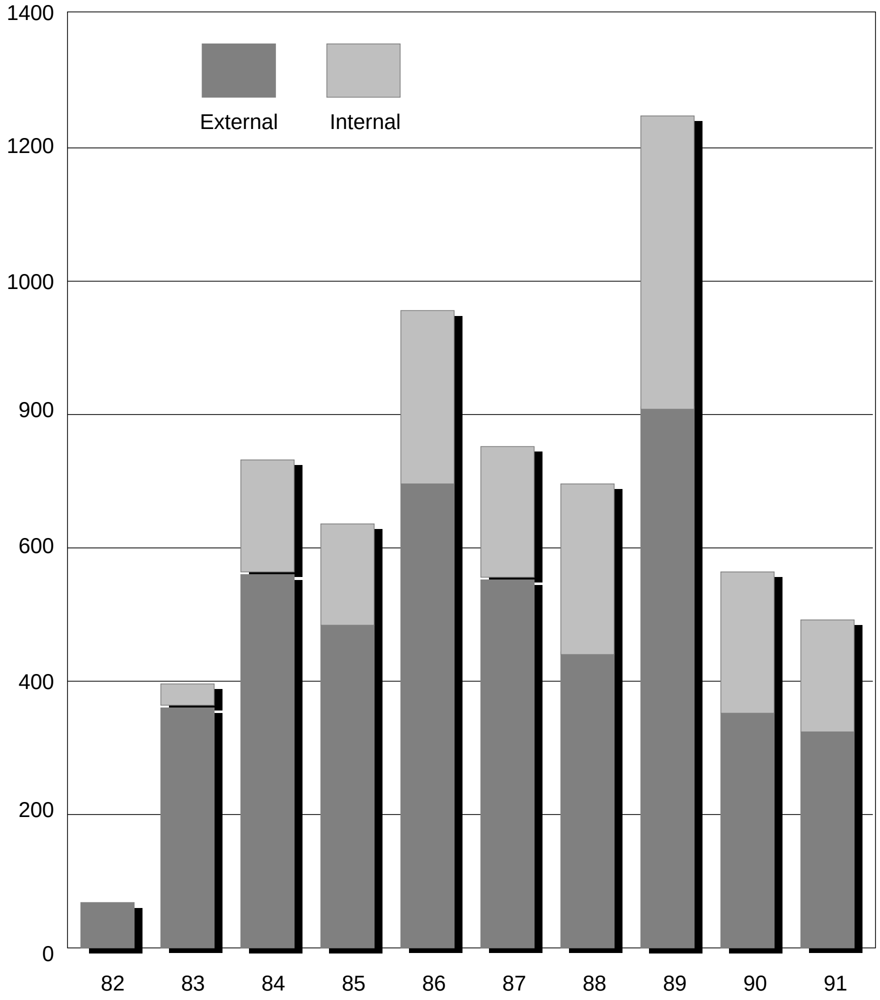
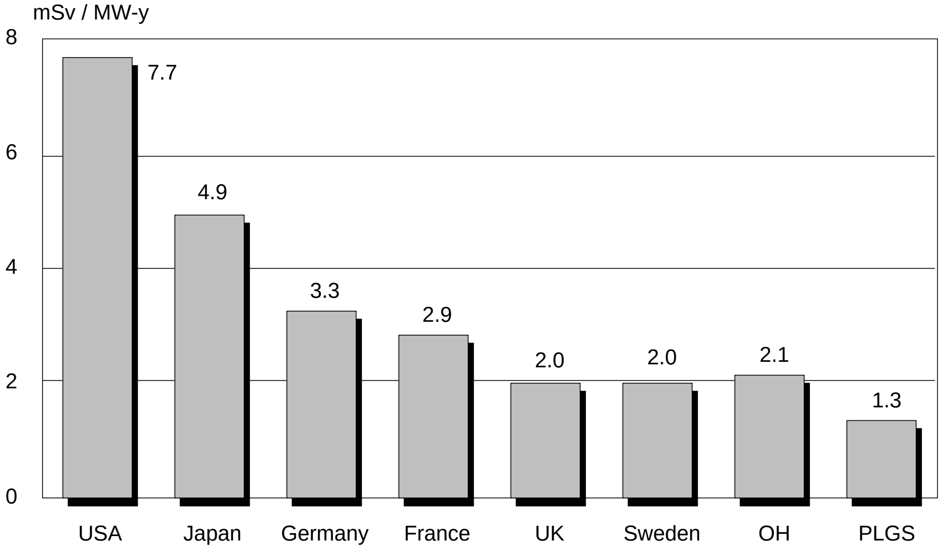


Figure 1.2.4

Station Dose per Mw(e)-y Canada and other Countries



Performance in terms of the exposure of the public as a result of station operation is also good. Table 1.2.1 gives the release of radionuclides from Point Lepreau expressed as percentage of the allowable limit, (constant operation at the allowable limit numbers could mean that members of the public close to the boundary of the station will receive a dose of 5 mSv per year).

Table 1.2.1

Emission Data for Point Lepreau 1983 to 1990

	Gaseous Emissions		Liquid Emissions		Total Annual Dose, uSv
Year	% Del	uSv	% Del	uSv	
1983	0.016	0.16	0.008	0.08	0.24
1984	0.022	0.22	0.003	0.03	0.25
1985	0.036	0.36	0.005	0.05	0.41
1986	0.075	0.75	0.003	0.03	0.78
1987	0.070	0.70	0.001	0.01	0.71
1988	0.080	0.80	0.005	0.05	0.85
1989	0.080	0.80	0.008	0.08	0.88
1990	0.095	0.95	0.017	0.17	1.12

Radiation protection will be discussed fully in Module 3 of the course but there are some practices in radiation protection that are common to all CANDU stations. Public and occupational radiation exposure may occur by two routes:

- External Radiation Exposure
- Internal Radiation Exposure

External radiation exposure is received from radiation sources external to the body and internal radiation exposure is received from radiation sources that have entered the body. The two most common entry routes are **ingestion** and **inhalation**. For CANDU stations, a third route is of importance and that is

skin absorption. This is because tritium is produced in large quantities in CANDU stations from a neutron-gamma (n,γ) reaction with deuterium in both heat transport heavy water and moderator heavy water.

$^{2}_{1}H + ^{1}_{0}n = ^{3}_{1}H + \text{gamma}$

Presence of this particular radionuclide on CANDU stations has required some special design considerations in ventilation systems and in provisions for protective clothing

Radiation Protection in Emergency Conditions

Radiation protection of the public has to take into account both normal and emergency conditions. Protection of the public from direct external exposure to radioactive material on-site in emergency conditions is achieved primarily by shielding. The radiation levels inside the reactor building subsequent to a worst case accident define the thickness of the reactor perimeter wall and dome required to keep public exposure within required limits. This shielding is also sufficient to protect station staff during an emergency and is the principal mechanism in achieving “Post Accident Habitability”.

CANDU stations also have special ventilation systems for emergency conditions to limit the internal dose that a member of the public may receive in event of a worst case accident. Systems are provided to contain any radioactive material released from the fuel. The released radioactivity is either contained in the reactor building itself in the case of CANDU 6 station designs or in the reactor building and vacuum building structures in the case of Ontario Hydro multi-unit station designs. Release of the contaminated air within these buildings can be controlled by discharging the air through an Emergency Filtered Air Discharge System (EFADS). The timing of the release may be planned within certain limits. The filtration system included both absolute and activated charcoal filters designed to remove fission products, of which the radioiodines are of most concern

Public Radiation Protection in Normal Operation

During normal operation minimization of external exposure and internal exposure for both the public and station personnel is again the primary consideration but attention has to be given also to contamination control.

For the public in regard to external exposure the design to provide safety during an emergency, that is the shielding capability of the reactor building walls and dome, is sufficient to provide a satisfactory level of public protection. Protection of the public in normal operation in regard to internal exposure as a result of atmospheric discharges in a CANDU station is achieved by several design and operating practices:

1. providing systems which identify failed fuel which may then be removed as CANDU stations have on-power fuelling capability.

2. designing and maintaining a leak-tight primary heat transport and moderator system. This requirement is primarily intended to prevent losses associated with heavy water downgrading but it also ensures that only very small quantities of contamination escape from these systems,
3. maintenance of a corrosion product activity in reactor systems at a very low level by careful selection of materials e,g. low cobalt-59 content and by providing clean-up systems with a high clean-up flow,
4. provision of closed circuit ventilation systems with filters and/or dryers to clean the circulating air. A minimum of air is extracted to maintain adequate negative pressures in the system,
5. provision of monitoring systems to measure, and if necessary interrupt, the discharge of airborne contaminants from the station.

Protection of the public from internal radiation exposure resulting from liquid discharges is dependent also on the first three practices on the above list, but in addition contaminated liquid discharges are carried out on a batch basis after sampling. Heavily contaminated liquids may if required be cleaned up before being discharged.

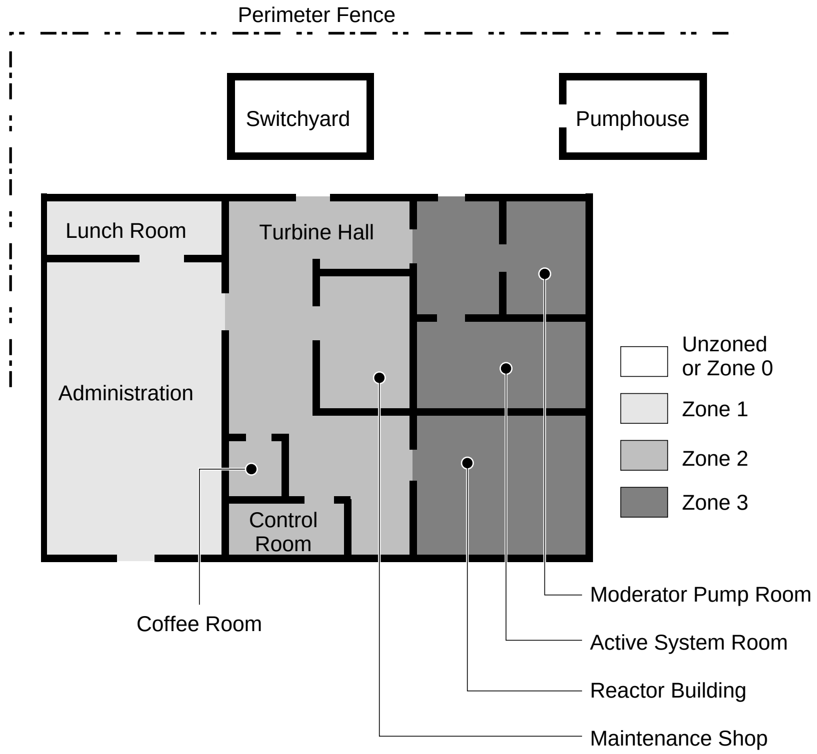
Occupational Radiation Protection in Normal Operation

External exposure is minimized in CANDU stations by provision in reactor systems of the clean-up systems and material selection (low cobalt) and by the ability to remove failed fuel.

In addition, shielding has been provided based on feedback from operating plants of how dose has been received. Systems are compartmentalized and shielded so that operating staff may maintain, inspect, etc. equipment with minimum exposure from adjacent systems.

Ventilation plays a major part in minimizing internal exposure of staff. The ventilation design principle is that air flows from clean to less clean areas of the plant. All CANDU stations have a zoning plan. The buildings in the station are laid out to segregate as far as is practicable radioactive material from personnel and zones are established based on the degree of radioactive contamination or the potential for contamination in the work area. Stations operated by the various utilities differ slightly in their specific definition of contamination control zones but the following definitions represent the practices reasonably well and are shown schematically in Figure 1.2.5.

Figure 1.2.5
Zoning Schematic



Zone 1 - This zone contains no radioactive equipment and is free from contamination. No form of radioactivity is allowed to enter this zone. Typically, this includes the administration and engineering offices. Eating is permitted.

Zone 2 - This zone is normally free of contamination and radioactive equipment. However, maintenance or the movement of radioactive material from Zone 3 can temporarily create contamination, which will be cleaned up following the completion of the activity. Eating is only permitted in the designated areas of the control centres.

Access is controlled between this zone and other plant areas. This zone includes the control room, the access corridor of the Reactor Auxiliary Building, control equipment rooms, non-active shops, stores, shower and change rooms.

Zone 3 - This zone contains the principal sources of contamination, radioactive material and equipment. The presence of contamination is normal. Sources of contamination are localized and kept under control.

It includes the reactor building, parts of the reactor auxiliary building (fuel handling and irradiated fuel storage bay areas), part of the maintenance building, including the waste management areas.

In addition there is usually an unzoned area - this zone contains no radioactive equipment and is normally free from contamination. However, radioactive materials may move through the area provided they are suitably contained. Typically, this zone includes the yard area and buildings which are free of contamination such as the pumphouses and water treatment plant. Zone 0 is similar to Zone 1.

Physical barriers, such as railings, are provided, as well as procedural controls, to direct the movement of traffic between zones. To prevent the spread of radioactive contamination, a number of contamination monitors are provided at the access point between Zones. Persons leaving a zone having a higher number designation to enter a zone having a lower number designation are required to use the monitors. Except from Zone 0, doorway monitors automatically monitor all persons before they enter the administration building. If radioactive contamination is above a prescribed level then the doorway monitor will alarm.

Control of contamination spread is important to minimize internal exposure but it is also vitally important in terms of public relations. If small amounts of radioactive materials which would not deliver even a minor internal radiation dose if ingested, are discovered outside the plant, e.g. in employees homes or in non-radioactive shops where equipment has been sent for repair, then public confidence in station operation will be lost.

4. Grouping and Separation

Principles

A fundamental principle in CANDU reactor design and a regulatory requirement, is the separation of process systems, the systems for power production, from safety systems. The objective is to reduce the likelihood of common cause events which could initiate an operating upset and impair the performance of safety systems in accident conditions.

In the CANDU design during normal and accident conditions the following systems provide operational safety:

1. process systems used for power production
2. safety related systems which mitigate the consequences of accidents; these are:

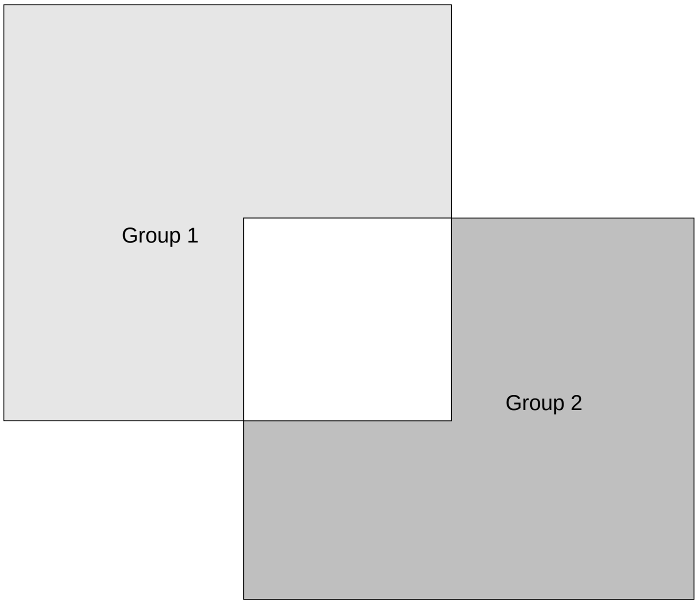
- Special Safety Systems,
- Safety Support Systems, which provide electrical power, instrument air and cooling water in the event of process failure,
- Safety related process systems which perform a mitigating function during an accident, in addition to their normal power production functions.

For accident conditions postulated for the plant, including the case where anyone of the Special Safety Systems is assumed to be unavailable, the radiation dose to the public must remain within defined limits. This leads to a requirement for independence between each Special Safety System to ensure, in the event of the failure of one, that all others can function and provide the required level of protection. The Special Safety Systems must also be independent of the process systems to the greatest practical extent, to ensure that the required safety function is not lost as a consequence of a serious process failure. As an example the location and orientation of the main turbine generator unit must take into consideration the possibility of missiles being thrown from this huge rotating unit. Important Special Safety and Safety Support Systems must be located so that in this event these systems would not be impaired.

In addition to these requirements within the plant, the systems which perform safety functions must also be protected from the effects of severe external events, such as tornadoes and earthquakes.

CANDU plant design provides for the necessary separation of power production systems and safety related systems. This is illustrated schematically in Figure 1.2.6.

Figure 1.2.6
Grouping and Separation Philosophy



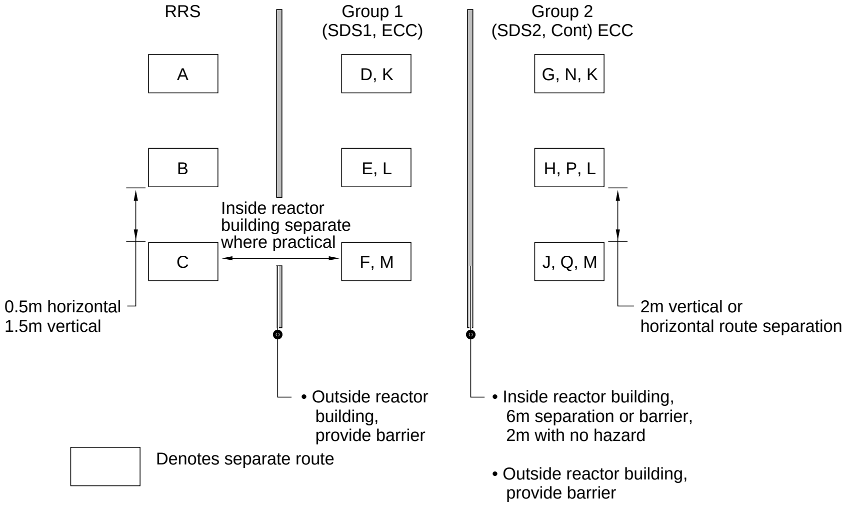
In the CANDU 6 design these are called GROUP 1 and 2 respectively. It is also necessary to ensure that failures affecting an area of the plant, e.g. fires and flooding cannot damage all Special Safety Systems. These are therefore further separated to minimize the likelihood of this and to minimize the likelihood that failure of one system will cause failure of another. Special Safety Systems are qualified to maintain necessary safety functions during and after severe external events such as fire, missiles, extreme structural loads and adverse environmental conditions.

Normal practice in CANDU systems is to triplicate important process and Special Safety Systems. For example the reactor regulating system or shutdown systems have triplicated instrumentation and physical components. These triplicated systems are called ‘channels’. The instrumentation wiring of triplicated shutdown systems is physically separated. An example of the principles being applied to the routing of important process channels and Special Safety System channels in CANDU 6 design is shown in Figure 1.2.8. Similar principles which may be different in detail are followed in CANDU 6 and Ontario Hydro stations. Separation or distance requirements exist to keep each piece of redundant equipment away from its counterpart,(Figure 1.2.9). In this way electromagnetic interference, fire, flood, or missile should not affect both pieces of redundant equipment simultaneously.

Figure 1.2.8
Triplicated Instrument Channel Designation and Separation

	Channel Routing and Identification		
	A	B	C
Reactor Regulating System	A	B	C
Group 1 a) Shutdown system # 1 Emergency core cooling system	 D K	 E L	 F M
Group 2 b) Shutdown system #2 Containment system E.C.C.	 G N K	 H P L	 J Q M

Figure 1.2.9
Separation of Process System (Group 1) and Special Safety System (Group 2) Triplicated Channels



Detailed requirements for separation distances for electrical supply and cable routings are part of the design specifications.

The importance of separation of systems was highlighted during the Browns Ferry reactor accident where a fire disabled a number of process and safety systems together and difficulty was experienced by the operators in putting the reactor in a safe shutdown state although this was eventually managed.

Plant Control in Normal and Abnormal Operation

Two control rooms are provided in a CANDU plant. The main control room (MCR) is located in the power production area of the plant. The MCR provides the operator with the capability to control and monitor all production and special safety systems (Group 1 and Group 2 systems). This effectively provides centralized operational control for the CANDU station. The design objective for personnel traffic routing in normal operation is to help achieve efficient operation and control of the station taking into account the work functions and the inter-relationship of work groups.

A secondary control room (SCR) is located in the area of the plant housing Safety Systems. The SCR is used only if the MCR becomes uninhabitable or is out of service because of some catastrophic event. The MCR can be isolated and control of Special Safety Systems (Group 2) by the MCR disabled. Failures in process system (Group 1) areas of the plant can be isolated from the SCR so that no propagation of these failures to Special Safety Systems occurs.

Plant Control for Abnormal Operation

All equipment in the SCR is qualified for operation after a seismic event, whereas the equipment in the MCR is qualified only to the degree that it will not fail in a manner which would injure the operations personnel in the MCR. This means that the MCR would be abandoned after a major seismic event, and operational control of the plant would shift to the SCR. The SCR is a smaller facility; however, it does allow the tracking and control of all the key plant parameters in the shutdown state. A seismically qualified and protected route, called the Seismic Route, is provided between the MCR and the SCR to assure that operations personnel can move freely and quickly to the SCR.

The MCR, the route to the SCR and the SCR are covered by the Post-Accident Habitability studies to assure that they are free of radiation risk to the operators following an accident.

Post-Accident Habitability

The layout of CANDU stations takes into consideration the need for access following an accident. It is necessary to demonstrate that the radiological consequences of an accident can be managed within the plant and that the safety systems function

1. to keep the reactor shutdown
2. to monitor the condition of the reactor and
3. to provide core cooling. AECB requires that post accident habitability studies:
    a) demonstrate that the accumulated dose to plant personnel over a ninety day period following an accident be less than 100 mSv in an area of continuous occupancy. Continuous occupancy is defined as the 24 hour period following an accident and 8 hours per day for the remaining 89 days and,
    b) identify areas to which access is required to satisfy the safety functions and in which the radiation field exceeds 2 mSv/h.

The post accident habitability study is carried out after the general plant layout is established, and sufficient detail design is completed. This detail, along with the radiological conditions resulting from the most serious accident postulated, enables the shielding requirements for occupied areas to be established. Special shielding for access in the post accident situation may compromise normal access requirements, so the approach taken is to provide the shielding as far as possible utilizing normal building walls and provide for shielding to be placed after the accident using appropriately placed sand bags or water tanks.

5. Tornado and Turbine Missile Protection

Tornado Missiles

Tornadoes can cause widespread damage and do so regularly in many parts of the world. Nuclear station design must take this into account. The damage comes from two sources:

1. sudden atmospheric pressure changes which cause buildings and structures to collapse and
2. missiles, that is objects hurled into the air by the tornado which impact on buildings and structures.

The design objective for protection against tornado damage is to protect the Special Safety Systems and the Safety Support Systems (the Group 2 Systems) so that the capability of these systems is not impaired: the Reactor Shutdown Systems, the Heat Removal Systems including the main steam safety valve and the steam generators for initial cooling followed by the shutdown cooling system and the Containment System. The secondary control area must be protected and the buildings which contain systems that require protection must maintain their structural integrity. Essential support services e.g. electrical power, instrument air and cooling water must remain available.

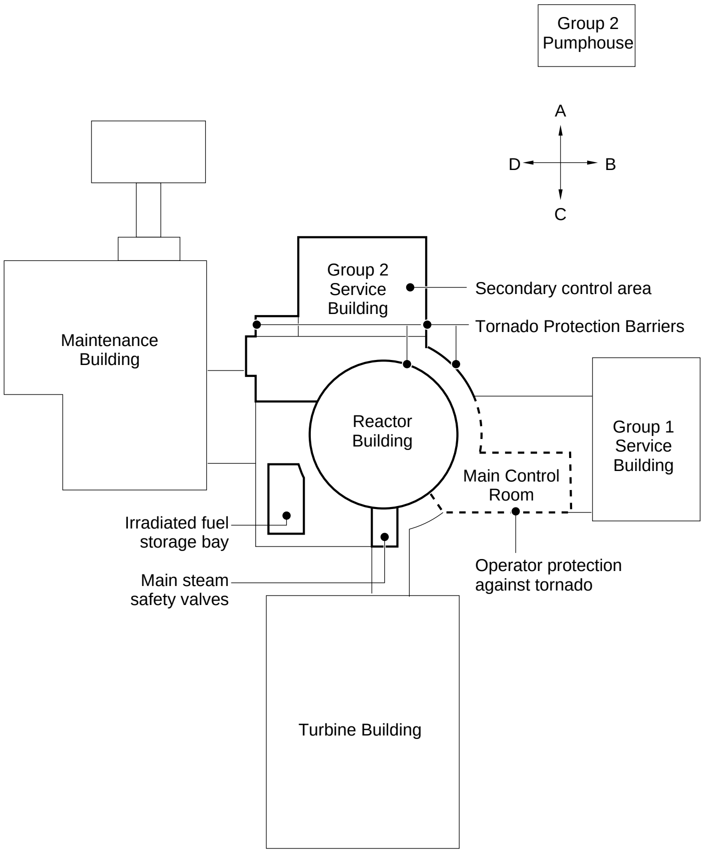
In providing for tornado protection a DESIGN BASIS TORNADO is defined. If data are available for a specific site then this may be used but in the absence of reliable data then the ANSI Standard ANSI/ANS-2.3-1983 provides a basis for design parameters. For example, the Design Basis Tornado is given below:

Maximum Windspeed	420 km/h
Translational Windspeed	92 km/h
Rotational Wind Radius	138 m
Maximum Pressure Drop	10 kPa

These values are based on U.S data and are considered conservative for sites outside the U.S.A. The recurrence frequency is in the range 10^{-5} to 10^{-6} events per year.

The types of missiles which have to be considered are large, high kinetic energy missiles which deform on impact, large rigid missiles which have the ability to penetrate structures and small rigid missiles which may possibly penetrate openings in missile barriers. Table 1.2.2 gives some examples based on the maximum horizontal windspeed given above. Figure 1.2.11 illustrates the provisions for tornado protection in the CANDU 6 design.

Figure 1.2.11
Tornado Protection



Typically for tornado missile protection a reinforced concrete wall of thickness of 30 to 35 cm is sufficient to withstand severe tornado missile impact. For protection against the negative pressure conditions experienced in a tornado blow out areas may be required

Table 1.2.2
Tornado Missile Examples

Missile	**Mass (kg)**	**Dimensions (m)**	**Velocity(km/h)**
Automobile*	1810	5 X 2 X 1.3	162
Utility Pole*	510	length 10.7 diameter 0.343	137
Steel Pipe (30 cm)	340	length 4.58 diameter 0.32	50
Steel Rod	4	length 0.915 diameter 0.035	86

*Maximum altitude: 9 m above site grade level.

Turbine Generator Missiles

In the past, in conventional stations disastrous turbine generator failures have occurred. This happened in Ontario Hydro in the mid nineteen fifties to two generator units at the R.L. Hearn station. In one of the events, part of the turbine turning gear went through the roof and landed in the parking lot. An even more disastrous event occurred in the U.K. in 1956 at the Uskmouth generating station of the Central Electricity Authority. The exciter of the generator was tripped in error when the unit was at full power. The turbine governor valves and the emergency stop valves failed to close immediately and overspeeding of the turbine caused the turbine rotor to disintegrate. The unit was totally wrecked and some heavy parts were hurled through the station roof and walls up to distances of several hundred yards. Two people were killed. Fortunately the vibrations due to the accident caused the emergency stop valves to close after a short time, or the death toll would have been much higher due to release of steam into the station.

When a turbine or generator disintegrate catastrophically for any reason missiles are thrown radially, so that the safest position for nuclear safety systems is on the axis of the turbine generator. This layout has been followed for CANDU 6 stations. However for multi-unit stations this approach is not necessarily followed. There are other factors to be considered in plant layout, such as the number and span of cranes to lift the huge turbine and generator parts. This is an important cost consideration in multi-unit stations. If the layout of safety systems is not along the axis of the turbine-generator then protection must be provided for missiles from the turbine. Heavy reinforced walls are provided at the Ontario Hydro Darlington station to protect safety systems and the control room from possible turbine generator missiles.

6. Fire Protection

Introduction

Fires have occurred in nuclear plants with varying degree of severity, causing economic loss, damage to safety systems and release of radioactive material. Fire could be an initiating event for an accident that could cause release of radioactive material. The design of CANDU reactors takes this and the potential for impairment of safety systems into account.

Safety Criteria for Fire Protection

The safety criteria for fire protection in CANDU nuclear plants are specified in CSA Standard CAN/CSA-N293-M87. In this document the following criteria are given for fire protection:

1. Shutdown:
   At least one of the two Special Safety Systems provided for shutdown shall remain available.

2. Heat Removal
   At least one group of systems or equipment to remove decay heat from the core and maintain adequate coolant inventory shall remain available.
3. Containment
   The containment boundary shall be maintained for fires that could cause a release of fission products within containment.
4. Monitoring & Control
   A control area and control equipment shall remain available and accessible, to the extent that the safety functions can be performed and the status of the plant can be monitored.
5. Support Services
   One group of systems needed to provide electrical power, cooling water, instrument air, or other services to maintain the required safety functions shall remain available.

The design objective is to ensure that the above criteria are met. This is achieved by design measures involving:

- Fire Protection.
- Fire Prevention.
- Mitigation of the Effects of Fire.

Fire prevention measures include limiting the use of combustible materials and preventing the ignition of combustibles.

Fire detection and suppression required under the standard are:

- selection and location of fire detectors
- design of fire signalling systems
- requirements for fixed fire extinguishing systems
- a requirement of all accessible areas of the plant to be protected by fire hoses and portable fire extinguishers
- provision for manual fire-fighting.

The requirements of the standard for mitigation of fires include plant layout, separation of Special Safety Systems, fire exits and design of ventilation systems to minimize the spread of fire. Detailed requirements are given in the standard and additional measures are also taken. As an example, one requirement for cable routing is that the routing of cables shall avoid, as far as is practical, areas of high temperatures or high fire load.

Fire barriers are also used to separate fire areas or fire zones. The former are, for example, walls separating buildings which will have a rating of three hours.

7. Pipe Whip and Jet Impingement

In the layout of CANDU plants the issues of pipe whip and jet impingement has been given increasing attention with each generation of station. There are two major concerns with pipe whip.

Inside the reactor building all high energy systems such as steam mains, feedwater and heat transport are studied and postulated breaks and loads identified. The first rule of layout is to separate physically high energy systems from safety related systems to the extent possible. Pipe whip restraints are provided where it is not possible to obtain adequate separation or where a whipping pipe could cause unacceptable damage.

Jet impingement target areas are also identified and safety related equipment is not located in such areas. If this is not possible then protection or redundancy is provided.

Outside the reactor building the high energy systems, primarily the steam mains and feedwater lines, are routed to avoid the control areas and on newer designs to avoid the possibility of harsh environment in the Reactor Auxiliary Building resulting from steam from a broken pipe or an upset condition in the turbine building.

Control Room Design

Training Objectives

On completion of this lesson the participant will have knowledge of:

- The design process for the main control room and the secondary control area;
- How the information which will be made available to the operators is selected; with an emphasis on the design of the human-machine interface;
- The working environment in the main control room, the control room inhabitability, and access to the secondary control area under accident conditions; and
- The role of human factors engineering in control room design.

Table of Contents

1. Scope

This module describes the design process as it pertains to the following topics:
- Control room layout;
- Human-machine interfaces; and
- Control room environment.

The objective is to provide the recipient with an understanding of the control room design process and to provide a knowledge base for the assessment of a control room throughout its life.

2. Introduction

Design and operation of control room facilities for a CANDU nuclear power station are key activities in the station life cycle. The main control room (MCR) must provide for all functions required for the normal and emergency operation of the plant systems. The secondary control area (SCA) is a backup facility to be used in the event that the main control room becomes inoperative or uninhabitable. In most instances, the two facilities contain the only sources of plant status information available to the operating staff. Because of the relative importance of the control room and the human-machine interface (HMI)* it contains, its design is a complex exercise that is of concern in the safety and licensing process. A successful design will result in a fully integrated control room system encompassing the operating staff, the human-machine interfaces, the operating procedures, and the training programme.

* The terms HMI and interface have the same meaning throughout the document.

3. Control Room Design

3.1. Historical Perspective

The history of the control of plant processes is not well documented but the use of control rooms is relatively recent. In the mid 1950's electrical generating plants were being built without control rooms. The HMI, a control panel of some sort, was usually located quite close to the process or equipment being controlled. However, in earlier plants there was no accumulation of controls and information on a panel. Operators were required to operate the plant by walking-around; checking local gauges and operating valves etc. Temperatures were often checked by "feel" by actual touch or by proximity. These operating methods had one key feature—the operators developed an intimate relationship with the plant. They knew the details of the processes and equipment they were working with, and used the knowledge to detect and diagnose problems at an early stage.

The purpose of describing the early methods of plant control is to highlight the difficulty in duplicating the "feel" effect into a control room environment. In most modern control room facilities the operators are completely isolated from the plant equipment and must rely on the information provided by the interface. CANDU control rooms are no exception to this and their designs must provide a replacement for the lost intimacy of the older methods.

3.2. The Role of Human Factors Engineering

Human Factors Engineering (HFE) is concerned with the capability of human to interact with complex technological systems e.g. a nuclear power plant. The goal is to improve human performance and minimise human error by ensuring that human needs are considered throughout a facility's life cycle. To achieve this state it is necessary to have a broad understanding of human capabilities, both cognitive and physical, and to understand, in detail, the functions to be performed by the human. These issues are relevant to the design and operation of nuclear control rooms and, consequently, HFE should have a significant role in the design process. This role has been recognised by the joint development of the Human Factors Engineering Program Plan (HFEPP) by AECL and Ontario Hydro, with support from the CANDU Owners Group (COG). The purpose of the plan is to define the human factors issues relevant to a specific project and how they will be addressed. Ideally, a complete HFEPP will address the following topics:

- the design of the facility;
- the organisational structure;
- the selection and qualification of its personnel;
- the function and design of interfaces;
- the training of personnel;
- the examining and testing of personnel in "authorised" operating positions;
- the development and use of operating, maintenance, and test procedures; and
- the standards to be used for assessing operational performance.

In practice, the HPEPP is intended to be tailored in scope and size to suit the specific project.

3.3. General Design Considerations

A Canadian standard for the design of control rooms, does not exist at this time. there are, however, international standards which cover the subject. The most current standard is IEC 964, prepared for the International Electrotechnical Commission [1]. This covers the subject in detail and is a useful source document.

Design of control rooms should reflect one very key point; they are people places. This fact should be reflected in their design by giving a high priority to the needs of the human occupants. The purpose of all control room equipment is to support the operator function. Human-machine interfaces should be unambiguous and should not impose any additional workload on the operators. Thus, they should be "transparent" in the interaction between the human and machine. This ideal is not always realized in practice, but it is a sound objective when compromises are being considered. Some important design considerations follow:

Human Factors Engineering

Human factors engineering, described in Section 3.2, has application in the following stages of the design:

Formal System and Task Analysis

System and task analysis provide a formal method of identifying the functions that will be performed in the control room. They are also used to ensure that the interface between operator and system is adequate for the functions to be performed, and to examine the physical and cognitive workload. Task analysis are performed for various plant operating scenarios.

Detailed Design

HFE theory and practice is used, in this phase, to ensure that the control room design is consistent with human capabilities. The desired result is a consistent application of methods for the selection and presentation of data, and the input of control actions. This is achieved by the early selection of standards and guidelines, and a strict interpretation of them, both, during this phase and continuing throughout the plant life cycle. Standards and guidelines will include topics such as:

- Layout conventions;
- Anthropometric data;
- Interface hardware selection;
- Control panel conventions;
- Coding methods (shape and colour);
- CRT display formats and hierarchies;
- CRT display call-up methods;
- Control and setpoint input techniques; and
- Alarm annunciation presentation and methods.

Verification

This process is used to determine whether the control room meets the technical requirements that were specified for it. As defined in IEC 964, the approach implies a check of the individual components of the control room against plant engineering criteria, human factors engineering criteria, and operating and functional requirements.

Validation
Validation, in this context, is the process that determines whether the actual design solution is adequate to meet the functional and performance requirements. In other words, can the operators perform all required functions and tasks using the control room as designed?

Degree of Automation
This is an important consideration in the design of the control room. A standard definition related to CANDU automation is the 'Fifteen Minute Rule'. This requires that there will be sufficient automation of critical systems to ensure that no operator action is required in the first fifteen minutes of any event that has been included in the safety analysis for the plant. The degree of automation will impact on many of the decisions taken with respect to the HMI, the devices used for control and monitoring, the physical size, and the control room personnel requirement.

CANDU plants have been highly automated since their inception. Early plants (NPD, Douglas Point) used analogue control technology. Introduction of digital control computers (Pickering A) has resulted in a gradual transfer of automated functions from analogue to digital methods. At Darlington, all of the closed-loop control is performed in the control computers. The proposed CANDU 3 design will take this further to include computer-based automation of the logic controls for pumps, valves, etc.

The situation with respect to CANDU automation is that the control strategies for reactor and other processes have become more sophisticated, resulting in increased complexity in the implementation of the automation.

These situations, CANDU and world-wide, have led to concerns about the ability of the operators to understand what is occurring in the automated processes during normal and abnormal situations. Therefore, the division of responsibility between the operator and the automated control functions must be addressed in the design process.

Role of Procedures
The relationship between the HMI and procedures is a relevant consideration in the design process. Ideally, the procedures should be based on the functions of the plant systems and the interface requirements would emerge during their preparation. However, the preparation of procedures early in the design phase has not been practiced to-date. The common approach is to prepare the procedures after the HMI design is essentially complete, but future design strategies should consider the early production of draft procedures.

Safety Separation
The design of the control rooms (MCR and SCA) are subject to the rules imposed

by the safety separation philosophy. In post-Pickering A CANDU plants the systems are divided into two groups — Group 1 and Group 2. The systems within each group has varied with each generation of plant design, but Group 1 always includes Shutdown System No.1, and Group 2 includes Shutdown System No.2. The separation philosophy for a specific plant will detail the requirements that must be met in the design. For the purposes of this lesson it is sufficient to understand the following:

1. The control centre, including the MCR, is a Group 1 facility,
2. The SCA is a Group 2 facility,
3. The primary location for Group 2 controls and instrumentation equipment is the SCA,
4. Group 2 interfaces and equipment that are placed in the MCR for convenience of operation must be buffered (electrically) to ensure that any damage occurring in the MCR cannot propagate to the SCA and impair the functioning, in any way, of the Group 2 systems,
5. Group 1 and Group 2 (non-buffered) equipment and cabling must be separated by a minimum prescribed distance or approved physical barrier,

Environmental Requirements

The requirements for the environment in the control room (MCR and SCA) must address the needs of both the operators and the equipment. Major considerations are:

- temperature and humidity must be controlled to the specified limits for human occupation and equipment needs;
- the lighting must be adequate for all tasks;
- noise levels should be kept at a level consistent with those of a normal office;
- the building structure and the internal fittings in the MCR must be qualified to a degree that will minimise the operating personnel during and after a seismic event; and
- the control room design must protect against possible damage or impairment by fire, tornado missiles, turbine missiles, the ingress of water, steam, toxic gas, and smoke, and possible radiation fields in a post LOCA environment.

3.4. The CANDU Control Room Design

A standard approach, that has evolved through several generations, is evident in the design of CANDU control rooms. The basis of the design is the provision of two facilities. A primary control centre, containing a number of component areas including the MCR, and a secondary location containing the SCA. Significant design features are as follows:

Main Control Room

The principal design objective for the main control room is to provide a facility from which the plant can be controlled in a safe and efficient manner. This objective must be met for all operational states and accident conditions.

All possible activities that will be performed from the MCR must be identified.

These include commissioning, normal and abnormal operation, maintenance outages, and rehabilitation functions. Each may require different uses of the available equipment and space. It is often necessary to install extra equipment in the MCR for the duration of a particular job (e.g., retubing) and space must be available for future additions to the HMI and support equipment. Conceptual control room designs often show minimum proportions that will not meet the demands of a plant life cycle. This occurs because control rooms are often compared with aircraft cockpits. They are not the same. A cockpit is designed for a relatively simple set of well-defined functions —i.e., flying the plane.

The MCR in a CANDU plant will contain the interfaces for the control of the generating unit(s), the common equipment (in a multi-unit plant), and the fuel-handling system. It will also provide adequate work station facilities for the operating staff. Easy retrieval of operating and emergency procedures must be ensured, and space for group consultation should be available.

The MCR should be a pleasant place to work in, with an environment controlled to meet the needs of both the humans and equipment.

The control centre will include other areas that are provided to support the operation of the actual control room. These areas are:

Control Equipment Rooms:
These will contain equipment that supports the interface between the control room and the plant. The equipment will include control and display computers, power supplies, logic controls, and panels containing controls and information required on an occasional basis by the operators.

Work Control Area:
The ongoing maintenance of a plant requires an efficient means of controlling all work activities. Planning and clerical functions associated with the issuance of work permits are done in this area. Also, the personnel who require the permits can obtain them without disturbing the control room operators.

Offices:
As a minimum there will be office space for the shift supervision staff.

Amenities :
A kitchen and washroom facilities for the use of the control centre staff are normally included in the layout.

Secondary Control Area
The secondary control area (SCA) is required to withstand specified Design Basis Events (DBE). The most severe of these is an earthquake. Therefore, the SCA must be located in a secure area of the plant in; a seismically qualified structure.

It should also be protected against common-mode events such as a main steam line break. The SCA will become the primary point of unit operations if the MCR is rendered inoperative. Therefore, the HMI in the SCA must contain sufficient monitoring and control equipment to enable the reactor to be shutdown, cooled, monitored, and maintained in a safe shut down condition. All of the equipment contained in, and operated from, the SCA must be qualified to DBE Category B.

Originally the concept of the SCA was to provide a limited alternative to the MCR. In practice it has become another, albeit smaller, control centre containing the control logic, instrumentation, and documentation for the Group 2 systems. The SCA will likely be the initial point of contact with the off-site emergency agencies, thus requiring the necessary communications equipment and documentation.

General considerations that apply to the design of the MCR also apply to the SCA. However, there are additional human needs to be considered in its design. Operators must be able to reach the SCA under all circumstances if they have to evacuate the MCR. Therefore, a secure route, free of possible obstruction from seismic-induced debris, must be provided between the MCR and SCA. In a major emergency, the operators may be required to spend a long time in the SCA before relief is available, so food, water, and toilet facilities must be provided.

4. Human-Machine Interface Design

4.1. Introduction

Control of a CANDU plant is a highly automated function, using a combination of digital and analogue methods to perform the monitoring and control functions. Operators, however, are responsible for monitoring the performance of those functions. In addition, they initiate startup and shutdown activities, direct power manoeuvres, co-ordinate maintenance activities, initiate system tests, and intervene in the control function in the event of equipment failure or abnormal incidents. An extensive human-machine interface (HMI) is required to support the human role in the monitoring and control function.

4.2. A Review of the CANDU Human-Machine Interface

Before proceeding, it is necessary to introduce the concept of parallel and serial interfaces:

A purely parallel interface has the monitoring and control functions arranged in a single form of presentation —i.e. it is on view at all times. A control panel is the prime example of a parallel interface.

A purely serial interface is usually computer-based, and the monitoring and control functions are normally presented in small “chunks” on CRT display formats.

HMI has been evolving gradually, from the parallel, to the serial form of presentation. The majority of interfaces in current use in CANDU plants are in an intermediate, or hybrid, mode.

CANDU plant interfaces evolved through several generations of plant design:

First generation:
This generation of control room was primarily parallel in nature. The HMI consisted of discrete components (handswitches, indicator lights, indicating meters, chart recorders, annunciator windows, etc.) mounted on a control panel. A small amount of data was presented in serial format from a data logger or plant computer. HMI’s from this generation were used at NPD, Douglas Point, and Pickering A.

The design and operating experience with these plants demonstrated that the parallel interface, despite being popular with operators, was a large , complex, and costly structure. Human factors studies were suggesting that large parallel interfaces were possible contributors to operator error in high stress situations. The serial interface approach, based on emerging computer display technology, was proposed as a way of assisting the operators in coping with the increasing quantity of information.

Second generation:
This generation employs a hybrid presentation of parallel and serial interface methods, and it introduced computer-based display technology into the control room. Control panels still use many of the discrete components of the previous generation but the computer-based control function are performed via CRT displays and keyboards. CRT‘s are also used to display data in tabular, bar, and trend formats. Annunciation is a combination of CRT and window methods. This generation is used at Bruce A&B, Pickering B, and AECL 600 MW plants.

This generation has now been in service since the mid 1970’s and has accumulated many hours of operation. Introduction into the Bruce A design was probably based more on technical availability than on any study of functional need. At the time it represented a method of keeping the size of the HMI within reasonable bounds, and in bringing some flexibility into the presentation of the monitoring and control functions.

The display technology, at the time, was still immature and this resulted in some limitations in its application. Design was device-oriented in that the majority of the display formats were basically duplicates of hardware instruments. Aside from a few text based and special function formats, trend displays replaced chart recorders, and bar-charts replaced indicators.

The display call-up method was also limited, using arrays of push buttons, each button being related to a fixed display page. It was apparent that a different method would be required if the CRT based interface was to include many more pages.

The significant weakness from a human factors perspective is the need to call-up, and remember, information (page-by-page) in order to obtain the overall status of a system. This adds to the operators' short term memory workload.

Some of this feedback was provided during the commissioning and early operation of the Bruce A plant and was used in the development of the next generation.

Third generation:
Computer-based serial interface became the primary source of information in this generation. The outcome was a reduction in the size of the control panel in comparison with earlier designs. Hardware devices (handswitches, lights, push buttons) were retained for the control of the discrete logic drives for pumps, valves, etc., and to provide a minimum backup control and monitoring capability in the event of a dual computer outage.

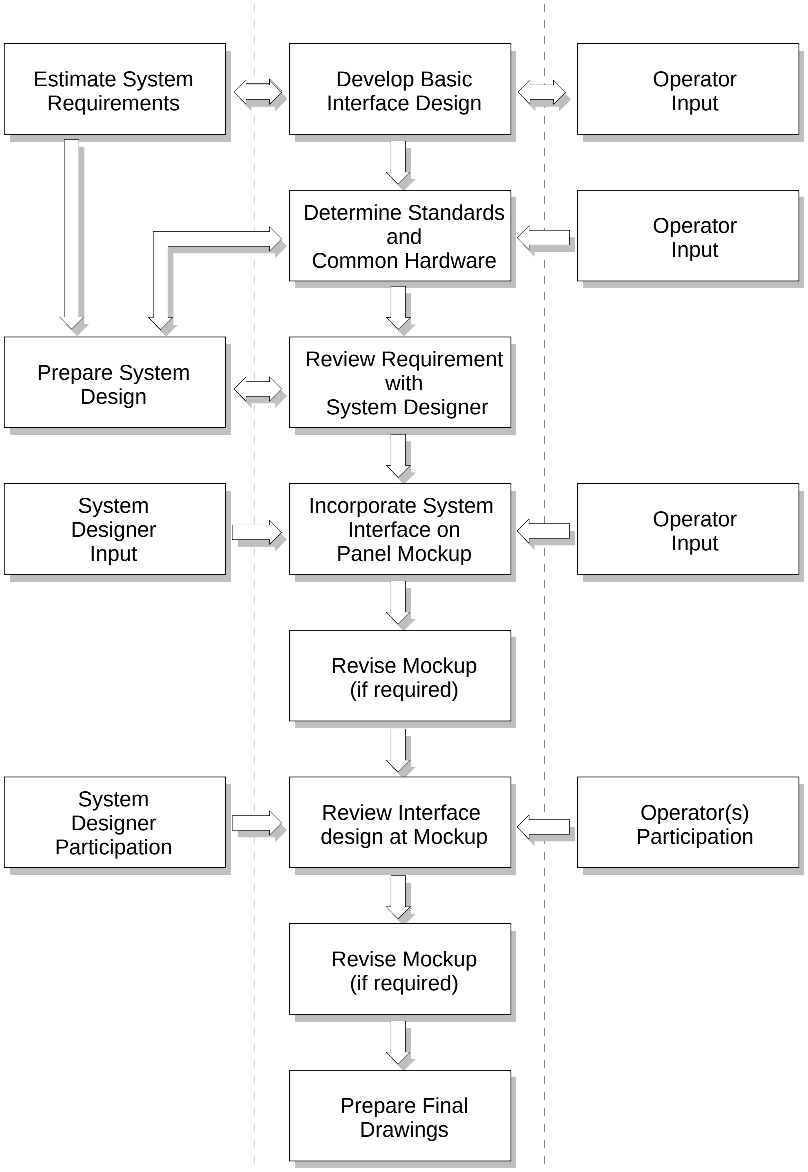
The quantity of CRT formats was expanded and arranged into system-based hierarchies to enable ease of selection. The concept of function-based formats was introduced wherein data were assembled from various systems into a single display to support a specific task. Display selection and control interaction was provided via on-screen (lightpen) and keyboard methods. The monitoring and test interface for shutdown systems was implemented on CRT's. Annunciation features were provided to give more analytical capability to the operators. Darlington NGS has the only example of this design.

This design is still in an early stage of operation and cannot be considered as a mature approach. However, its operation to-date and the advances in display technology has encouraged designers to move to the next generation. Also, the trend, in other industries, to eliminate hardware-based control panels has resulted in a rapidly diminishing supply of suitable control panel equipment.

Fourth generation:
This generation is still in the design stage. It will make maximum use of the computer-based techniques for control and monitoring of most plant systems. Control of both closed loop and discrete logic systems will be via the CRT screens. Use of function-oriented displays will be expanded to provide a context-sensitive interface for information and control. Hardware backup of essential safety-related functions will be provided, and large plant overview

Figure 1

Flow of Design Information in Conventional Design



displays are being proposed for the design. This generation will consist mainly of serial presentation with a small parallel component.

4.3. The Human-Machine Interface Design Process

Procedures for design of a HMI have been changing in recent years, moving in the direction of a formal, documented process. However, almost all of the plants that are operating today were designed using a conventional, and less rigorous, method in which the HMI design was biased by the technical requirements of the plant systems and equipment. Operator needs, while being considered, did not generally receive equal treatment when compromises were being made. In the new approach, a function-based perspective ensures that both technical and human needs receive due consideration.

The Conventional Method

HMI's in the currently operating CANDU plants were designed using a system-based design method. General flow of design information is shown in Figure 1. Control and measurement needs for a system were determined by the responsible process designer. These were taken by I&C designer(s) who would develop control strategy, determine the hardware/software requirements, alarm annunciation, and, in the case of a control panel, possibly select the interface devices. This work was done with reference to project standards but there was a range of possible results. Decisions relating to the automation strategies did not include any formal review of the operator function.

It was the task of the control room designer to derive a coherent HMI from a range of differing system requirements and equipment. Input would be obtained from operating personnel, usually a commissioning engineer. For some plants, a full-size mockup of the HMI would be constructed. This is a valuable tool in achieving consistency across the many systems that appear on the interface. A mockup also provides a common focus for designers, operators, and others (including regulators) when systems are being reviewed. The process is iterative and relies heavily on the ability of the control room designer to balance the often conflicting demands of the various designers, and also of the operating staff.

A deficiency in this approach is that the choice of information and controls to be included in the HMI is highly dependent on the judgement of the individual design and commissioning engineers who tend to have a system oriented perspective. This does not lead to an examination of the operators' task from an overall perspective. A plant does not normally operate as a collection of systems working in isolation. It operates as a set of systems working in unison. Thus, operators generally use the systems to perform functions, e.g. control reactor power. This is an important concept on which the newer design methods are based.

Human factors issues were not dealt with in a complete fashion. Ergonomic aspects such as layout, coding, labeling, and hardware design were addressed. However, issues such as an assessment of operator workload during normal and abnormal situations, and the impact of a loss of automatic functions received a limited qualitative treatment.

The topics of verification and validation (V&V) were also addressed in a rudimentary manner. A limited form of HMI verification was achieved by use of the mockup facility, and through the formal design review process. There was no attempt to conduct any meaningful form of validation in the design phase. Validation was, in effect, conducted on-the-job. First indications of any user problems were usually identified during the plant commissioning phase and subsequent operation.

The incident at Three Mile Island in 1979 had a major impact on the design of nuclear control rooms. Although much of the cause was attributed to issues of organisational management, operator error, training, and procedures, it was also determined that the HMI was a contributing factor. This prompted a regulatory demand in the U.S., which was followed by other countries, for a formal re-examination of control room and interface designs. The method that was developed to assist this process was the Control Room Design Review (CRDR). This process was primarily a human factors review, and examined the design and operation of control rooms. A full description of the program can be found in a series of documents produced by The Institute of Nuclear Power Operations [2]. From this initial work a more formal approach to control room and human-machine interface design has evolved.

The Function-Based Method
In this approach, the HMI is treated as a component of an integrated control room system and the control room is given the status of a plant system instead of just being a location where the plant systems are brought together. One treatment of the method is given in IEC-964 [1] and a CANDU approach is under development. Basic steps in the process are shown in Figures 2A & 2B.

The design process consists of two main phases:
1. Functional design

- This phase establishes the functional basis for the control room and HMI. It consists of:
- a statement of the functional goals to be achieved. These are based on the plant performance goals;
- the identification of the functions that are to be monitored and controlled from the control room;
  Note:
  This a hierarchical process. At the top of the function tree would be e.g. “Control Reactor Power” or ”Maintain Heat Sink”. These functions would then be decomposed until the lowest level of sub-function is reached.
- the assignment and verification of function to the operator or to the machine (automation);
  Note:
  This is done by developing a task structure for each function. The tasks are then analyzed to determine which functions should be performed by the operators and which should be assigned to the machine. The relative capabilities of both the operators (physical and cognitive) and the control systems, and the demands placed on them are key elements of this process.
- production of functional requirements for both human and machine based assignments;

- the functional integration of the control room system
  Note:
  This step is taken to ensure that the human and machine assignments are appropriate within the overall context.
- the validation of the allocation of functions.
- The functional design process also identifies requirements for the plant procedures, staffing needs (crew size), and training.

Figure 2A

Function-based Design of Control Rooms

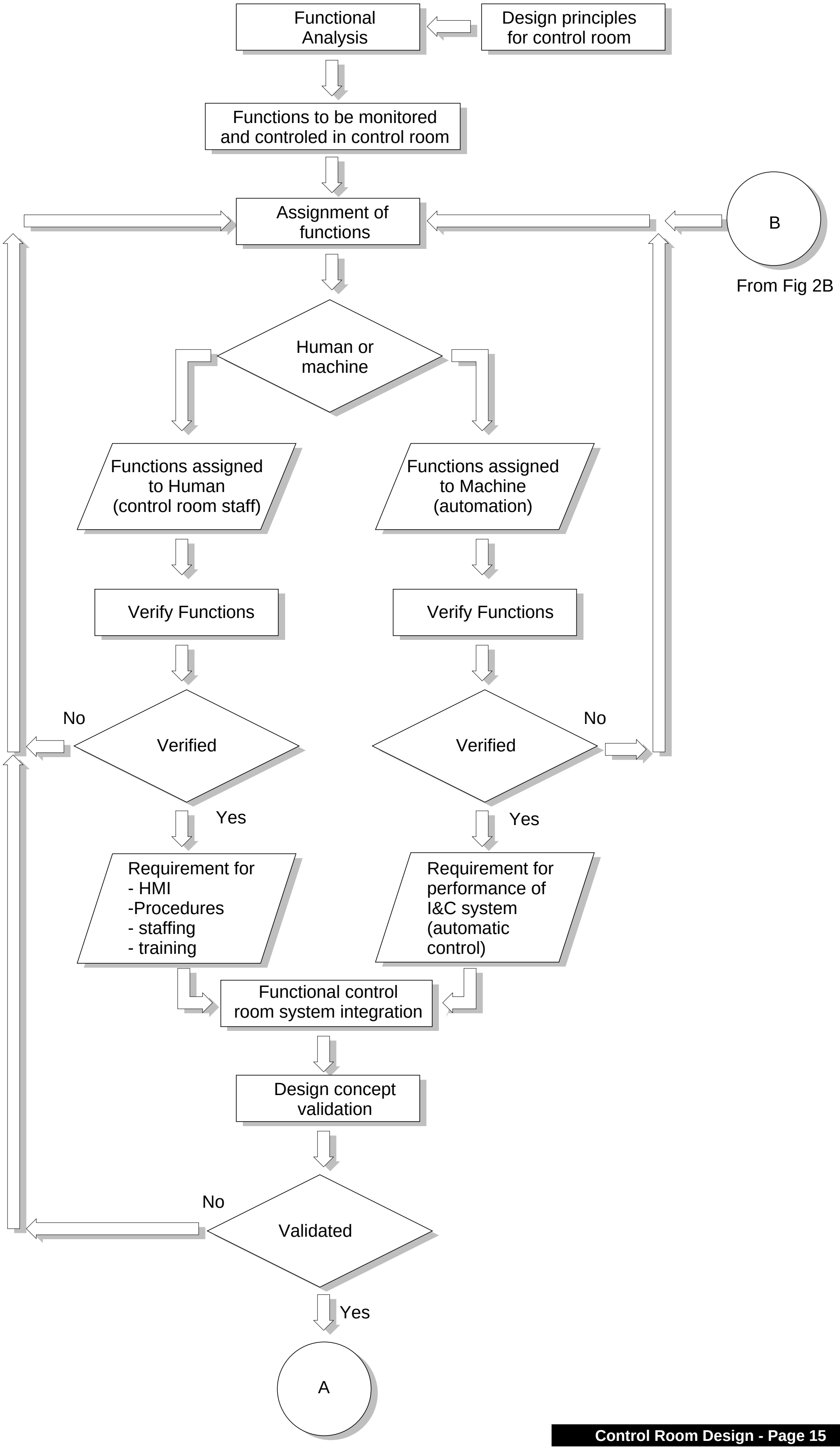
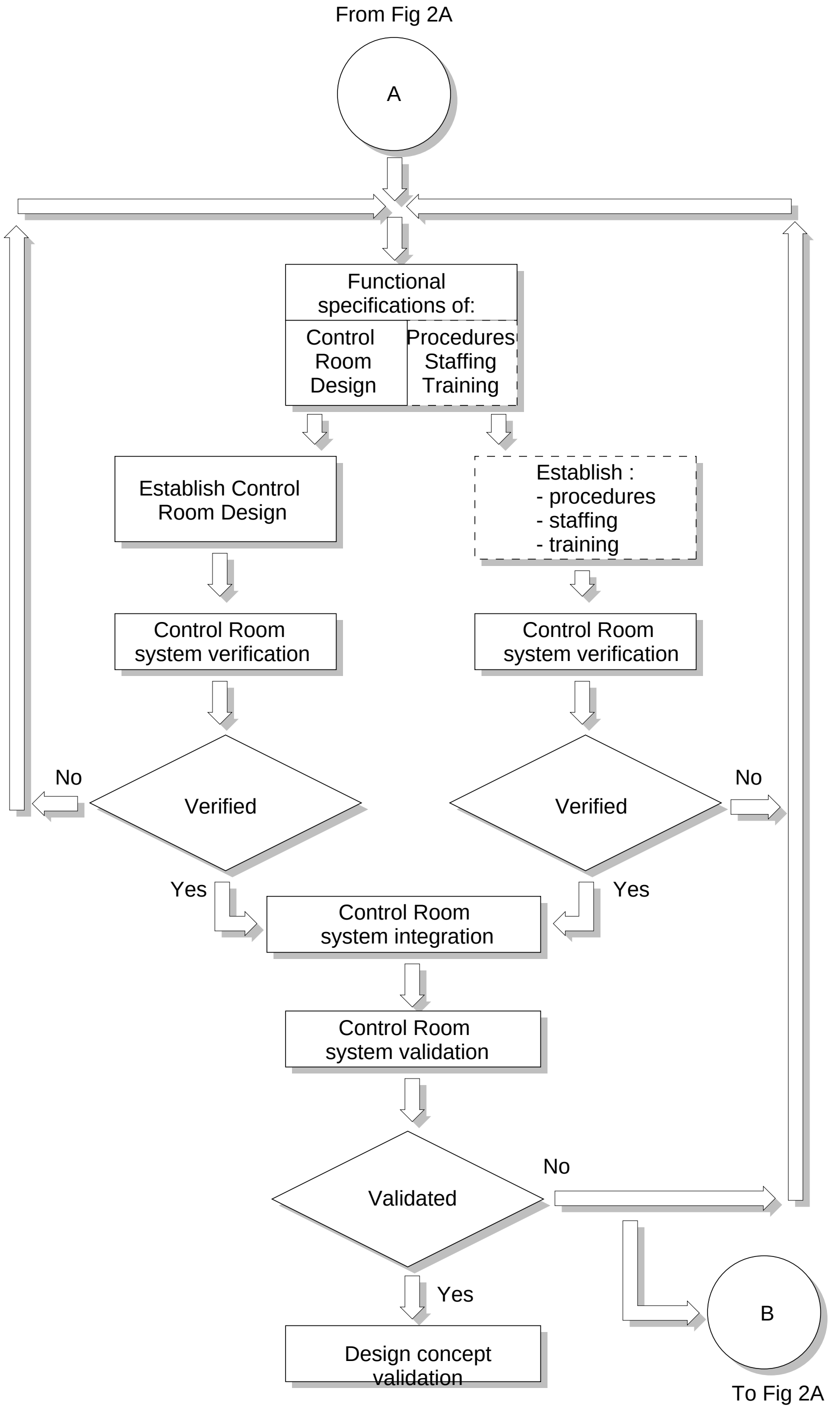


Figure 2B

Function-based Design of Control Rooms



2. Detailed Design

This phase takes the results of the functional design step and incorporates its findings into the detailed design of not only the HMI but also the process systems, and their control and monitoring equipment and methods. The information will aid in the following areas:

- the identification of monitoring information (including alarm annunciation) and controls required in the control room;
- the plant conditions for which a particular information or control signal is used. e.g. normal, abnormal, safety, or post accident monitoring;
- the range and accuracy required for each signal, and ensuring its adequacy for all intended uses;
- the design of function-based formats on CRT displays or control panels;
- selection of the methods to be used in presenting information.
  e.g. trend, indication, textual, coding, labeling;
- the development of hierarchical structures for the display of CRT based information;
- a formal verification and validation of the HMI

The final product from this process will be a control room (MCR or SCA) and HMI design that is based on a thorough analysis of its functional role in management of the plant systems.

The shortcomings of the conventional method did not prevent the design of many, very adequate, human-machine interfaces. Control rooms for CANDU plants were designed with due concern being given to the human factors knowledge available at the time. Most of this was in the ergonomic area, but there was a lack of information on the cognitive aspects that could be applied directly to the design task. However, control rooms were designed and operators, being resourceful people, were able to work safely in them. Deficiencies were corrected as necessary.

5.0 Specific Topics in HMI Design

There are some areas of the HMI design process that are worthy of some additional explanation:

5.1. Alarm Annunciation

The primary function of the alarm annunciation system is to advise the operators that an abnormal condition is present in the plant processes. In addition, it is often used to advise the operators of conditions other than alarm states. These uses include information regarding confirmation of automatic action, equipment status, test conditions, jumper status, and logging of operator actions. It is these secondary functions that have made the design of annunciation systems a difficult task.

Design of annunciation systems, like the rest of the HMI, has been changing from a parallel form of presentation to a partially serial one. Parallel format was implemented used the lighted window box, using various types of visual and audible signals to alert the operators. These windows are to be found in the upper portions of the control panels. Windows have proved to be a good method of performing the annunciation task, and operators have always expressed a desire to retain them. The main advantages are that they are always in sight, give a powerful indication of plant status, and operators, through familiarity, can quickly identify plant conditions. The move to the serial, CRT-based presentation is due to the vast increase in the quantity of alarm and other signals that the modern annunciation system is expected to handle.

For example:
Douglas Point GS had 650 alarm windows, and 28 printed alarm messages from its data logger for a single unit.

A single unit of Darlington GS 'A' plant has in excess of 8000 signals into its annunciation system.

The plants designed in the period between these two have a gradually increasing quantity.

The major difficulty in the design of the annunciation system is the large quantity of information to be handled. In a major event the flood of alarms can overpower the system and make identification of the cause a difficult task. Information can pass-by on a CRT screen at speeds beyond human comprehension and operators are often reduced to waiting for and reading printed output. Numerous solutions have been tried on the problem. These include techniques such as alarm conditioning and major/minor classification. Work is also proceeding, world-wide, in the application of artificial intelligence and expert systems to the problem. There have been analytical features added such as alarm summaries, high capacity information buffers, "freezing" of the CRT output, and some primitive attempts at alarm recognition strategies, but these have not solved the primary problem.

The main purpose of this section is to direct attention to the difficulties inherent in designing the annunciation systems. Experience to date with the computer-based methods have not been good and operators are highly critical of them. However, there is a lot of interest in the problem, and work is proceeding worldwide (including Canada) to develop solutions.

5.2. CRT Display Design

Designing a computer-based HMI differs in several ways from designing a control panel interface. The control panel is essentially a parallel configuration of hardware devices for the input of control actions and the presentation of information. The level of sophistication in the interface is low. Information displayed can ,in most cases, be considered as “raw”— i.e. does not undergo much processing, and the control inputs (handswitches) are mainly direct, with minor logic processing. Therefore, as long as the panel layout is in some logical format, the devices are adequate for the task, the labeling is consistent, and the procedures are compatible, the operators can use it to operate the plant.

It is an approach, while not ideal, that has served reasonably well in many applications including nuclear power generation. Its success is totally dependent on the human ability to process information and to convert it into actions. A major concern is that each operator will not have the same mental model of the plant and may use the displayed information differently to achieve the same result. The freedom of choice that is inherent in a parallel HMI permits variations from the model that was used to train the operators. The model can be shaped by experience and, because plants are in continuous operation, operators will not be exposed to the same events or conditions. Movement towards almost total reliance on CRT-based interfaces transfers this freedom of choice to the control room designer.

The concepts of device, system, and function-oriented display formats were introduced in Sect. 4.2. Development of these methods has made it imperative that the control room functions be clearly identified and examined before the HMI is designed. No longer is the operator being fed with “raw” information. It is now being arranged, or “massaged” into the discrete “chunks” that will be displayed on the CRT screens in a context-sensitive manner. Therefore, it is now the control room designer who must interpret the mental model that operators will be trained in. This is not a simple task, and input from plant operators and training staff should be an integral part of the design process.

The use of the formal method, described in Sect. 4.3., will assist in ensuring this particular change in the control room and HMI design process will be accomplished in a successful manner.

5.3. Safety System Interfaces

HMI for safety systems has evolved from the purely hardware type to a CRT-based system as used in the Darlington design. A safety system interface has two main functions:

- 1. To present the status of safety parameters.
- 2. For periodic testing of the safety system.

Operation of a safety system is generally automatic so the main operator interaction is with the testing function. The testing function should be given the same functional scrutiny as for the other plant functions, and the decisions re. manual, semi-automatic, and automatic testing should come from this analysis. Provision should also be made for flexibility in the test routines to minimise the imposition of non-standard actions on the operators.

5.4. Secondary Control Area and HMI

The SCA is a key part of the plant safety concept and the quality of its design must be, as a minimum, equal to that of the MCR. The design method must be as rigorous - the functions performed in the SCA must be completely identified, and all equipment and documentation required for their execution provided. There should be no attempt to produce an “economical” design based on an assumption that the facility might never be used in practice.

That operators will not be familiar with this HMI is important. They will have spent many hours in the MCR (and its simulator) and will have a good understanding of its use in both normal and abnormal conditions. Operators will receive periodic training in the SCA but, as it is not simulated, they will not be experienced using it in high workload, emergency situations.

Some process systems, for operating convenience, have an interface in the SCA and the MCR. In these cases, both should be as identical as possible. However, the primary interface for such systems must be in the SCA, and the schemes for the transfer of control between the two locations must guarantee that control is always possible from the SCA— i.e., it must not be possible for the MCR interface to block the SCA interface.

HMI will likely be a combination of panel and CRT based methods. All equipment must be qualified to the requirements of the plant DBE’s, and the low availability of qualified interface devices may put constraints on the HMI design.

5.5. Operator Support Systems

These are systems which are provided in a control room, as part of the HMI, to assist the operators. Those that have been designed to date have generally been in response to a need in a particular function. There are “on-line” versions using real-time data, and “off-line” versions using stored data and/or rules. Some of the functions being performed by systems, either in operation, or development, are:

- Safety System Monitor;
- Reactor Overpower Trip Status;
- Equipment Status Monitor;
- Critical Safety Parameter Monitor;
- Emergency Operating Procedure Entry Condition; and
- Fuelling Machine Systems Advisor.

An operator support system can be implemented as a stand alone device or it can be a function within a larger computer system e.g. the control computers. Systems are often prototyped on a personal computer platform and are proposed, by their originators, to be placed on the operators desk. As this type of system is proliferating, this is clearly an inappropriate location. A major concern of the HMI designer is how the system will be integrated into the overall control room concept.

These systems are usually described as giving decision-support or advice to the operators on the basis that the advice can be rejected by the recipient. The acceptance or rejection of this form of information is not a simple concept, and it is a major concern about the application of such systems. It is particularly important in systems using artificial intelligence or expert system technology. The design and integration of such operator support systems must be subjected to a rigorous functional analysis, and a comprehensive verification and validation process.

6.0 Control Room Rehabilitation Projects

Control rooms and the HMI are subject to change throughout their lifetime. The changes may involve the addition or removal of a function or device, a change in the control or presentation method, or a major rehabilitation of the facility. These activities must be undertaken within the bounds of the existing control room and HMI design philosophy. A series of, seemingly, small changes can be as destructive as a major rehabilitation.

Decline in the quality of a control room often starts with small changes to the control panels. The most common instance is the gradual separation of functional relationships. This will happen when, for expediency, a new control or display device is located in the nearest available space instead of in the correct functional relationship. If this practice has been occurring over a number of years, the spaces are further apart, and information gathering becomes a more complex task, leading to an increase in operator workload. If the effect is gradual the danger is that problems may not be apparent until identified by a study of seemingly unrelated events.

A major rehabilitation project can create a similar situation in a much shorter time period. Rehabilitation may occur after a plant has been operational for 15-20 years. During this time period the operators will have become very familiar with the control room and HMI, including any deficiencies, and will be confident in their ability to cope with most situations. Operator capability can be eroded if the changes are made without due consideration of the established operating methods.

The problems can be minimised if all changes to a control room and HMI are made in accordance with a planned strategy:

There should be a procedure in place to cover the minor modifications that are needed on a regular basis. This will reference the appropriate technical standards, documentation requirements, and human factors engineering considerations; and

A major rehabilitation activity should be treated as an independent project and have a formal design process in place. A Human Factors Engineering Program Plan should be prepared to cover all aspects of the changes. This will be "tailored" within the limits set by the project goals, schedules and costs, but should still contain the required elements as described in Sect. 3.2.

Rehabilitation projects are undertaken to improve the performance of a plant. Changes made to a control room should not diminish operator performance.

7.0 Safety and Licensing Aspects

Within the safety and licensing process the position of the control room is not well defined. The control room and its HMI have not been treated as an integrated system but rather as a place where all the plant systems are brought together. This is similar to the situation that existed in the conventional design process.

There are no regulatory guides that cover the control room performance requirements in a complete fashion. Although CSA standards do exist for safety system and post-accident monitoring interface requirements there is no Canadian standard for the design of control rooms. Designers, however, have developed in-house standards and guidelines, and international standards have been produced in several countries.

In the absence of a formal requirement, there should be evidence of the following points in any review of the control room facilities:
1. The control room(s) (MCR and SCA) should be treated as a fully integrated system.
2. The design should be based on a complete functional analysis of the plant processes.
3. There should be Design Requirement and Design Description documents for the facility.
4. The requirement for human factors engineering, as applicable to the control room design, should be defined in an HFEPP.
5. The design should be done in accordance with a formal documented design process.
6. A human factors engineering review should be included in the selection process for all devices used in the HMI.
7. The standards and guidelines (including any non-compliance) should be clearly referenced.

8. A formal program of design verification and validation should be required.
9. There should be a traceable document path covering all activities.

Design of a control room requires the melding of many separate parts into a fully integrated facility. This can only be achieved by the definition of good design practices and a consistent approach to their use over the lifetime of the plant.

8.0 Some CANDU Design Examples

It is not the intent in this document to provide detail design information for specific plants and their control rooms. Specific plant design manuals should be consulted for this type of information.

There have been a variety of control centre, control room and HMI designs in the CANDU family. These include single, dual, and four unit layouts. Some examples of those in current operation are given below:

Single Unit Configuration

Plants:	AECL 600 MW units.
Basic Layout:	MCR with an adjacent control equipment room. SCA located near reactor building.
Automation:	Computer control of reactor and process systems. Some analogue based control.
HMI type:	Power systems: Hybrid. Composite of panel and CRT-based methods. CRT information presented in device oriented manner. Safety systems: Panel based interface. Fuel-handling: Hybrid. Composite of panel and CRT-based methods.
Annunciation:	Hybrid combination of window and CRT-based methods.

Dual Unit Configuration

Plants:	There are six CANDU plants in India that use this configuration.
Basic Layout:	Single MCR containing the HMI, with an adjacent control equipment room, for each unit. SCA: Status unknown.
Automation:	Early units used analogue control for reactor and process systems. Current status unknown.
HMI type:	Power systems: Panel based interface. Safety systems: Panel based interface.

Fuel-handling:
Panel based interface to solid state logic system.

Annunciation: Window alarm system supported by data logger.

Four Unit Configuration

Plants: Pickering A&B, Bruce A&B, Darlington A.

Basic Layout: Pickering A&B:
Single rectangular MCR containing the HMI, with an adjacent control equipment room, for each unit.
SCA (B plant only) for each unit located adjacent to reactor building.

Bruce A&B:
Similar to Pickering with the addition of separate control equipment rooms for control computers, common control equipment, and fuel-handling systems.
SCA (B plant only) for each unit located adjacent to reactor building.

Darlington A:
Differs significantly from Pickering and Bruce. A single linear configuration is used for the MCR as a result of seismic requirements, and operator input from Bruce A experience. Each unit has a control computer room and a small control equipment room located behind its control panel. The fuel-handling and the electrical and common equipment control panels have a similar arrangement. Control equipment that is not required to be in the control centre complex is located in each units field area.

SCA for unit is located adjacent to its reactor building.

These contain the interface and the control equipment for the unit Group 2 systems. A fifth SCA is located in the central service area to provide for the control and monitoring of the Group 2 systems that are common to all units. These include Containment, ECI Delivery, EPS, and EWS. Provision was also made for the co-ordination of emergency communications.

Automation: The implementation of the automation strategy has been a gradual evolution from analogue to digital methods. Pickering 'A' had digital control of the reactor and boiler pressure. Darlington has digital control of all of its major closed loop control systems. Analogue backup control is provided where

HMI type:	Power systems:
	Pickering A&B: Panel based with CRT interface for computer-based control functions. CRT information presented in device oriented manner.
	Bruce A&B: Panel based with greater use of CRT interface for information display. CRT information presented in device oriented manner.
	Darlington A: CRT based interface with panel for discrete logic drives and backup systems. CRT information presented in system oriented manner.
Safety systems:	Pickering A&B: Panel based interface.
	Bruce A&B: Panel based interface.
	Darlington A: CRT interface for information display and testing.
Fuel-handling:	Pickering A&B: Hybrid. Composite of panel and CRT-based methods.
	Bruce A&B: Simple CRT interface with panel backup.
	Darlington A: Extensive CRT interface with panel backup.
Annunciation:	All plants: Hybrid combination of window and CRT-based methods.

9.0. References

1 International Electrotechnical Committee
IEC-964
Design For Control Rooms of Nuclear Power Plants
2 Institute of Nuclear Power Operations
Control Room design Review
(Series of Documents)

Reactor Assembly

Training Objectives

On completion of this lesson the participant will be able to:

- name each part of a drawing of a CANDU reactor assembly and state its function;
- state the material used for the major parts of the reactor assembly and explain the reason for its use;
- Describe the various types of reactivity control units, their orientation in the reactor, their function and the criteria which govern their design'
- Describe the major differences between the CANDU 6 design and Bruce "A";
- Outline the radiation hazards associated with the reactor assembly.

Table of Contents

1 Overview

At the outset it will be useful to have a simple diagram of a CANDU nuclear-electric power plant. Although it may only be a series of boxes joined by lines it provides a set of “pigeon holes” into which the information, you will receive in the following lessons can be stored. In fact, you may want to add to the diagram as the course progresses. Such a diagram is shown in Figure 1.

There is, of course, a heat source, the reactor, in which heat is generated by a controlled fission reaction. The reactor vessel holds the fuel, the moderator and the fission or reactivity control mechanisms.

Heat is transported to the steam generators by the heat transport system. It is the heat transport system in the CANDU plant that is pressurized and this has a major impact on the design of the reactor vessel (it does not have to be a pressure vessel). In addition, because the steam and feedwater systems are not in direct contact with the fuel they are free of radioactive contamination as are the turbine and all other related equipment. This also has considerable impact on the plant design and maintainability. The heat transport system is a closed system in which the coolant is circulated.

The steam and feedwater system is also a closed system. The steam drives the turbine and electrical generator and is exhausted to the condenser and then the condensate is pumped back to the steam generator.

This lesson deals with the nuclear steam supply system which includes the reactor assembly, the fuel handling systems, the moderator systems, the heat transport systems and the steam and feedwater systems. Figure 2 shows a more detailed diagram of a CANDU steam supply system.

2 Reactor Assembly

The reactor assembly brings together the fuel, the moderator and the reactivity control mechanisms to produce heat through a controlled fission reaction. In the CANDU reactor the fuel is supported in horizontal tubes, pressure tubes, which themselves run through other tubes, known as calandria tubes. The calandria tubes run axially inside a horizontal, cylindrical tank called the calandria. The moderator circulates outside the calandria tubes within the calandria. Mechanisms for controlling reactivity are located vertically and horizontally across the calandria between the calandria tubes.

Figure 1:
Simplified CANDU Nuclear Electric Power Plant

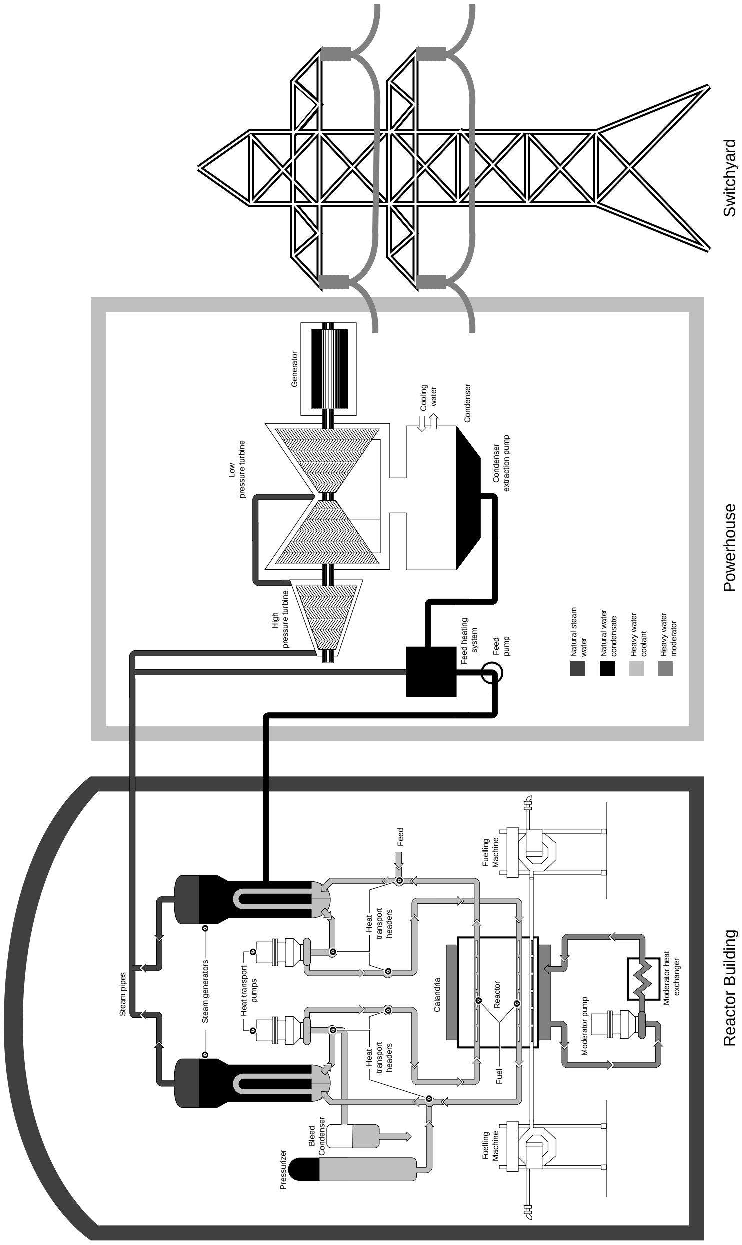
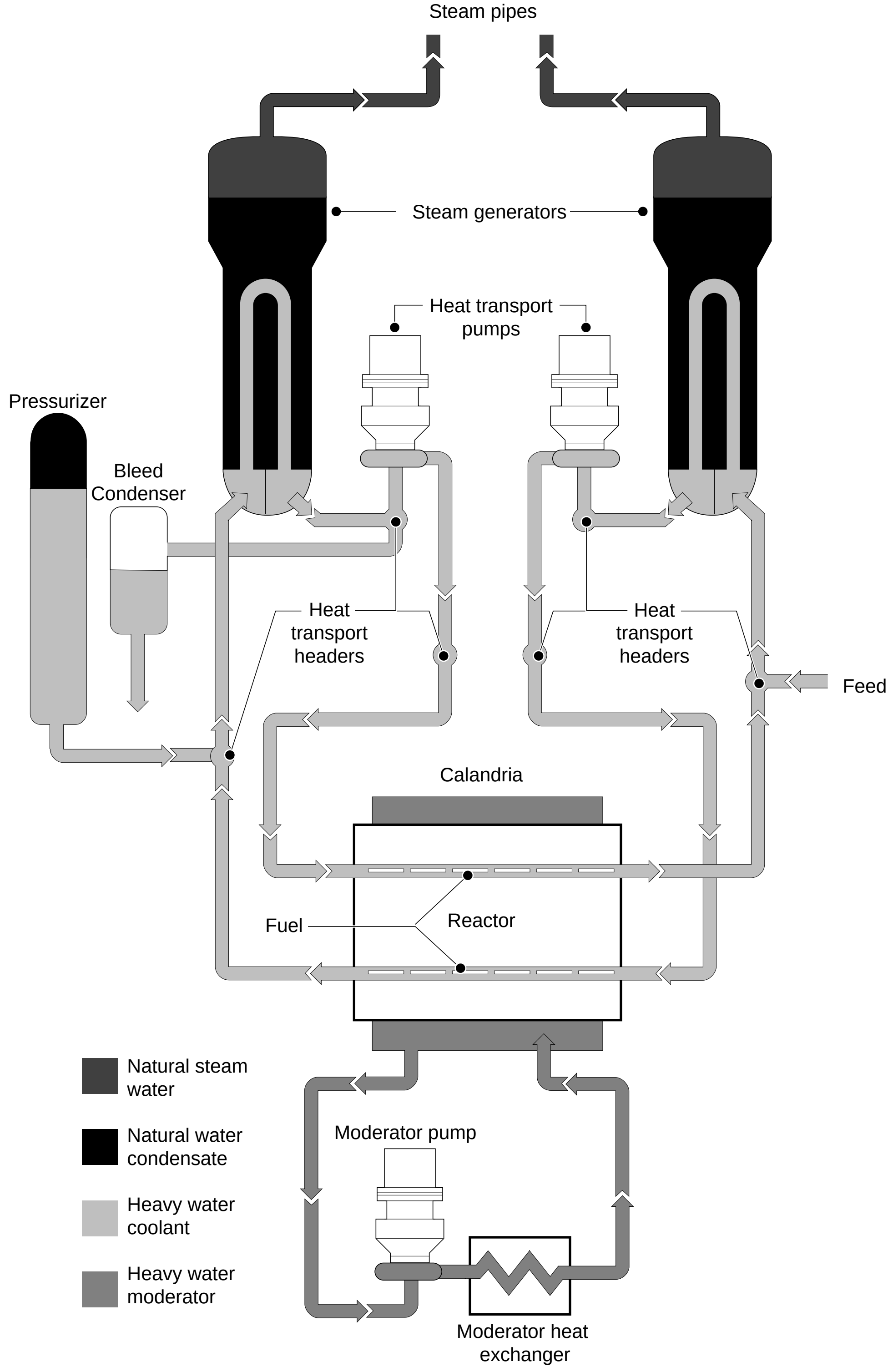


Figure 2
CANDU Nuclear Steam Supply System



2.1 Functional Requirements

The functional requirements of the reactor assembly are:

- to provide support for the fuel channels and in-core components of the reactivity control mechanisms;
- to contain the heavy water moderator surrounding the fuel channels. The moderator is a relatively cool supply of D_2O and will act as a heat sink in an emergency condition;
- to provide radioactive shielding for the fuelling machine area during normal operation and shutdown;
- to provide access and support for reactivity control mechanisms and process piping;
- to provide overpressure protection during normal operation and a postulated pressure/calandria tube rupture;
- to provide structural support for the complete assembly within the vault.
- that all components must be designed to withstand all operating conditions.

2.2 Calandria

The calandria is a horizontal, cylindrical, single-walled, stepped shell enclosed at each end by tubesheets and spanned by calandria tubes (380 for CANDU 6) installed parallel to the horizontal axis.

The stepped shell is formed from a main shell and two subshells of smaller diameter at each end. The main shell and the subshells are connected by annular plates welded to each of them which form flexible diaphragms to accommodate differential expansion between the calandria tubes and the shell. This is illustrated in Figures 3 and 4.

The calandria shell and tube sheets are made of austenitic stainless steel plate.

The calandria shell, tubes and tubesheets form the pressure boundary for the moderator.

The calandria tubes are arranged in a square pitch. The ends of the tubes are rolled into the calandria side tubesheets together with stainless steel inserts forming high integrity joints (Figure 5). The tubes are made of seam welded, annealed Zircaloy 2 alloy.

The calandria tubes provide access through the calandria for the fuel channel assemblies and also support the pressure tubes by means of four garter springs per channel (Figure 5). The gap between the fuel channel pressure tubes and the calandria tubes is known as the annulus and it serves as a means to insulate the hot pressure tubes from the relatively cool moderator.

Guide tubes for the reactivity control units penetrate through nozzles in the calandria shell passing between the calandria tubes and locking into

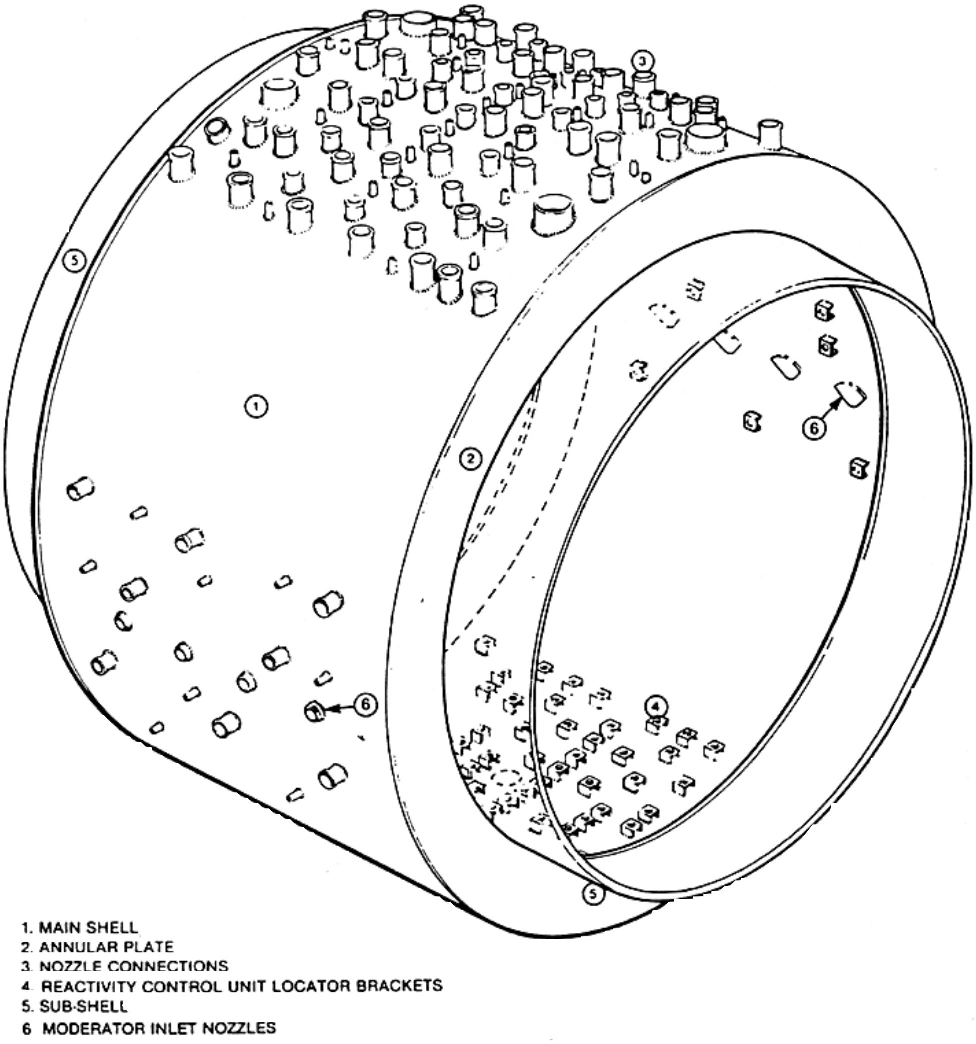


Figure 3:
Calandria Shell

Figure 4:
Calandria Vessel Pressure Boundary

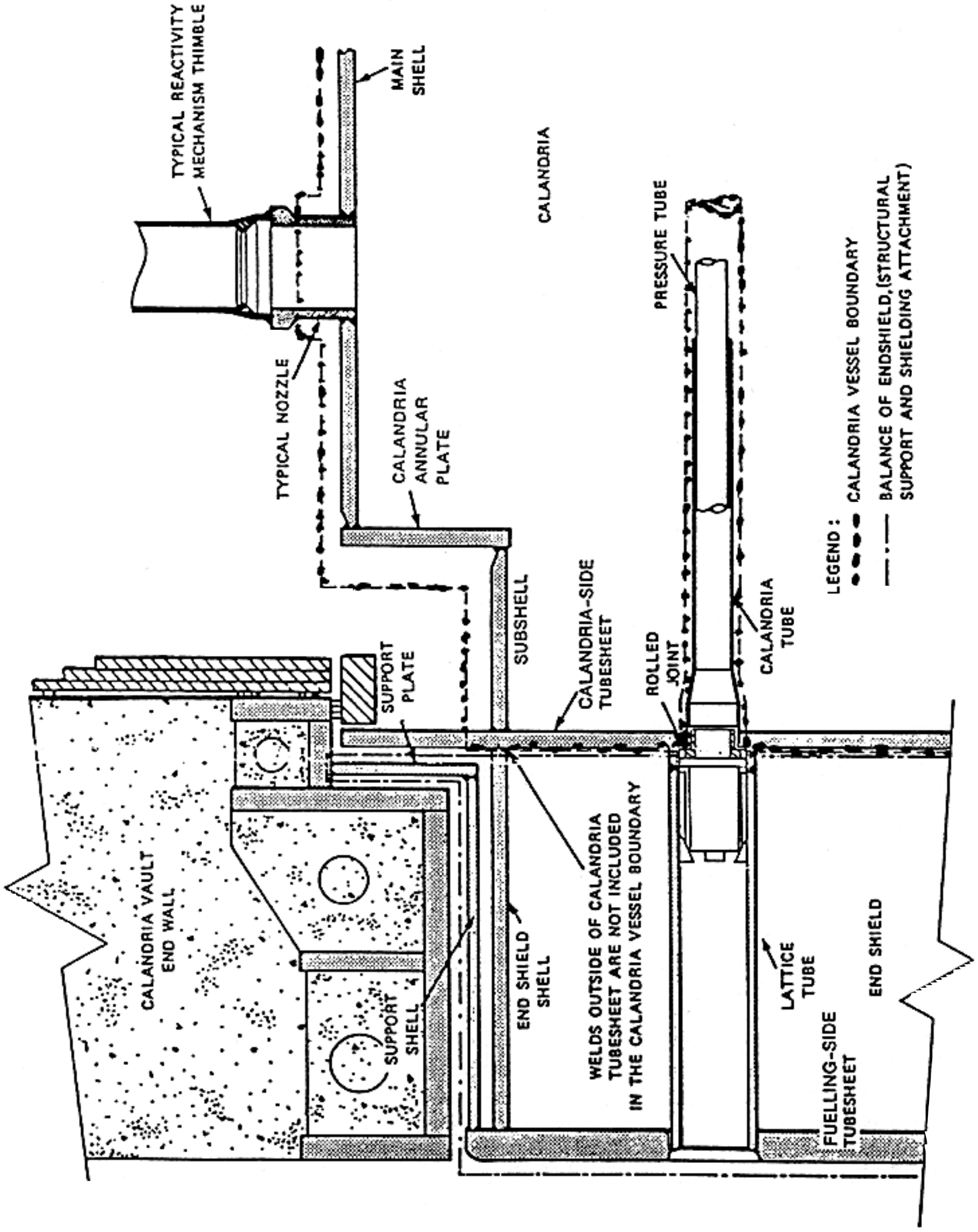
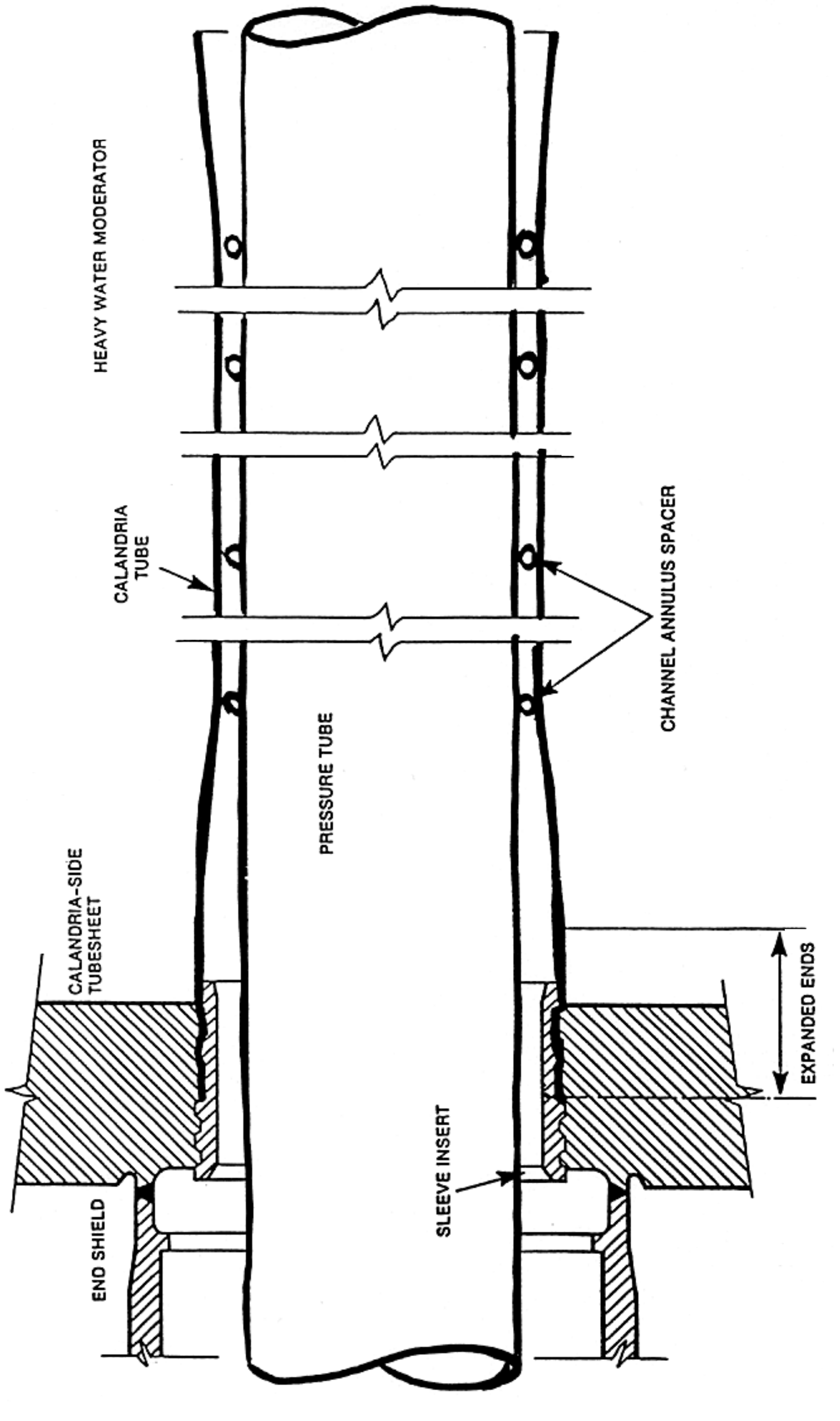


Figure 5:
Schematic of Installed Calandria Tube



locators on the opposite wall of the calandria. The weight of each control unit is supported by a stainless steel thimble which is welded to the calandria nozzle and extends either vertically to the reactivity mechanism deck or horizontally to the outside of the vault wall in the case of CANDU 6 or the shield tank at Bruce NGS 'A'. Figures 6,7 and 8 illustrate these placements.

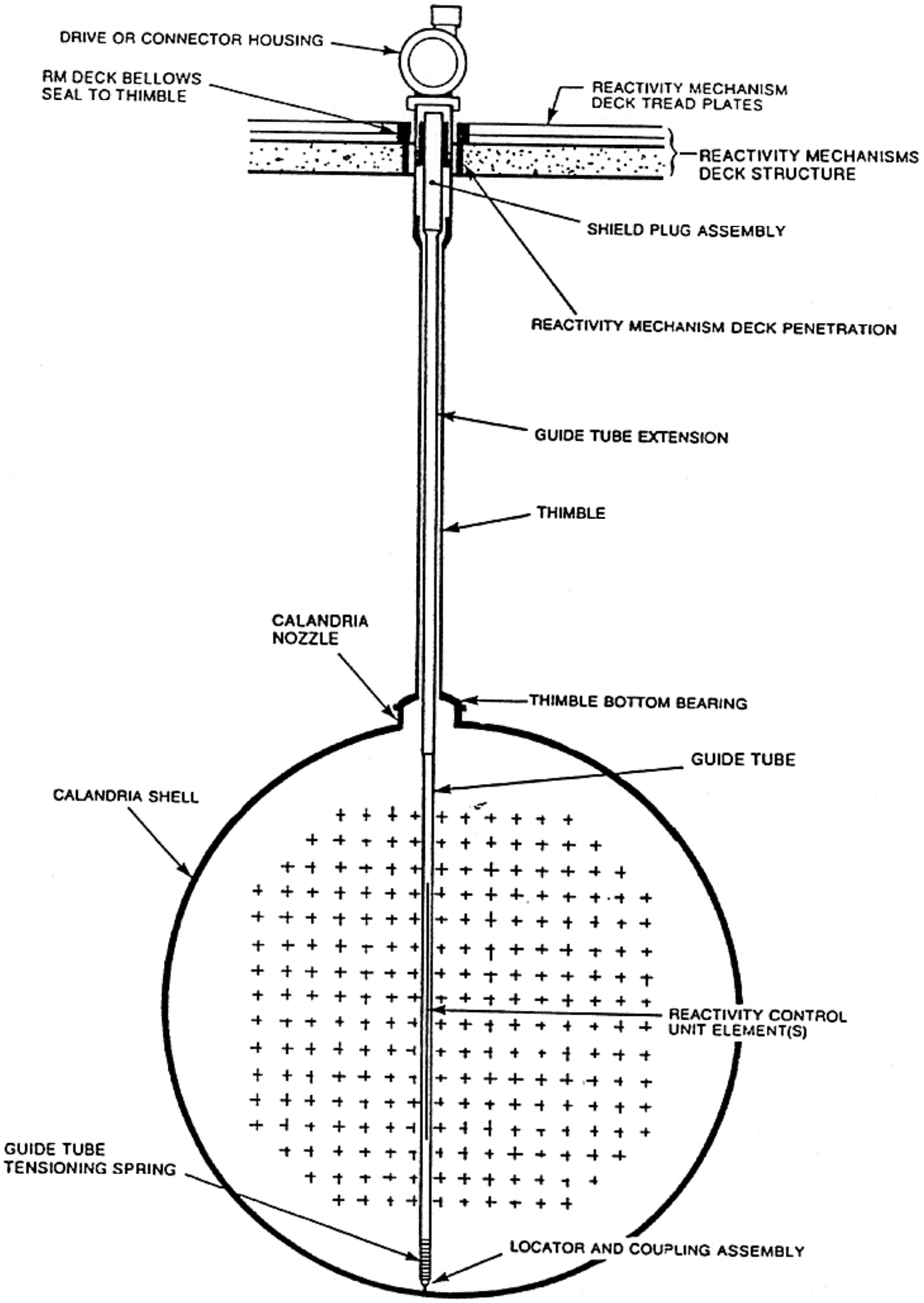
Moderator enters the calandria through nozzles placed to avoid direct impingement onto the calandria tubes and to promote good flow to all parts of the calandria. It exits through nozzles at the bottom of the calandria for purification and cooling.

The calandria is equipped with four pressure relief ducts which exhaust into the reactor building and are sealed with rupture discs. They are designed to provide adequate area for the discharge of heavy water in the unlikely event of a pressure tube/calandria tube rupture and to protect the calandria structures from the effects of overpressure.

A helium cover gas is provided above the moderator which is maintained at a level a few centimetres above the top of the calandria in a head tank. This is to provide pressure regulation in the moderator during normal operation and gives a means of limiting the concentration of deuterium in the pressure relief ducts.

The calandria is classified as a pressure vessel according to the terms of the ASME Boiler and Pressure Vessel Code, Section III, class 3. All materials and procedures conform to the code and in those areas where the code is silent the materials and procedures have been selected to comply with the intent of the code.

Figure 6:
General Configuration of Vertical Reactivity Control Units



2.3 End Shields

The purpose of the end shields is to reduce the direct radiation from the reactor core in the fueling machine vault both during operation at high power and during shutdown and/or during maintenance.

The two end shields form part of the calandria assembly. They are vertical cylindrical shells closed at each end by a tubesheet, one of which is the calandria tube sheet. The calandria tubesheets are shared by both the calandria and the end shields, (Figures 4, 9 and 10).

Between the tubesheets is an array of lattice tubes (380 for CANDU 6) through which the fuel channel assemblies penetrate. The lattice tubes are welded to the calandria tubesheet and are joined to the other tubesheet by a combination of rolling and welding. These joints are designed to eliminate the problem of crevice corrosion.

The space between the tubesheets and around the lattice tubes is filled with carbon steel balls and is cooled by demineralized, light water. This provides the necessary biological shielding for work in the fuelling machine space during shutdown.

Heating in the end shields is produced by the absorption of radiation energy and by conduction from the heat transport system feeders and end fittings.

The end shields are fabricated from austenitic stainless steel. They do not form part of the pressure boundary and are therefore not regarded as pressure vessels however the materials conform to the requirements of the ASME Boiler and Pressure Code, Section III, Class 1.

Figure 7:
Horizontal Flux Detector Unit (CANDU 6)

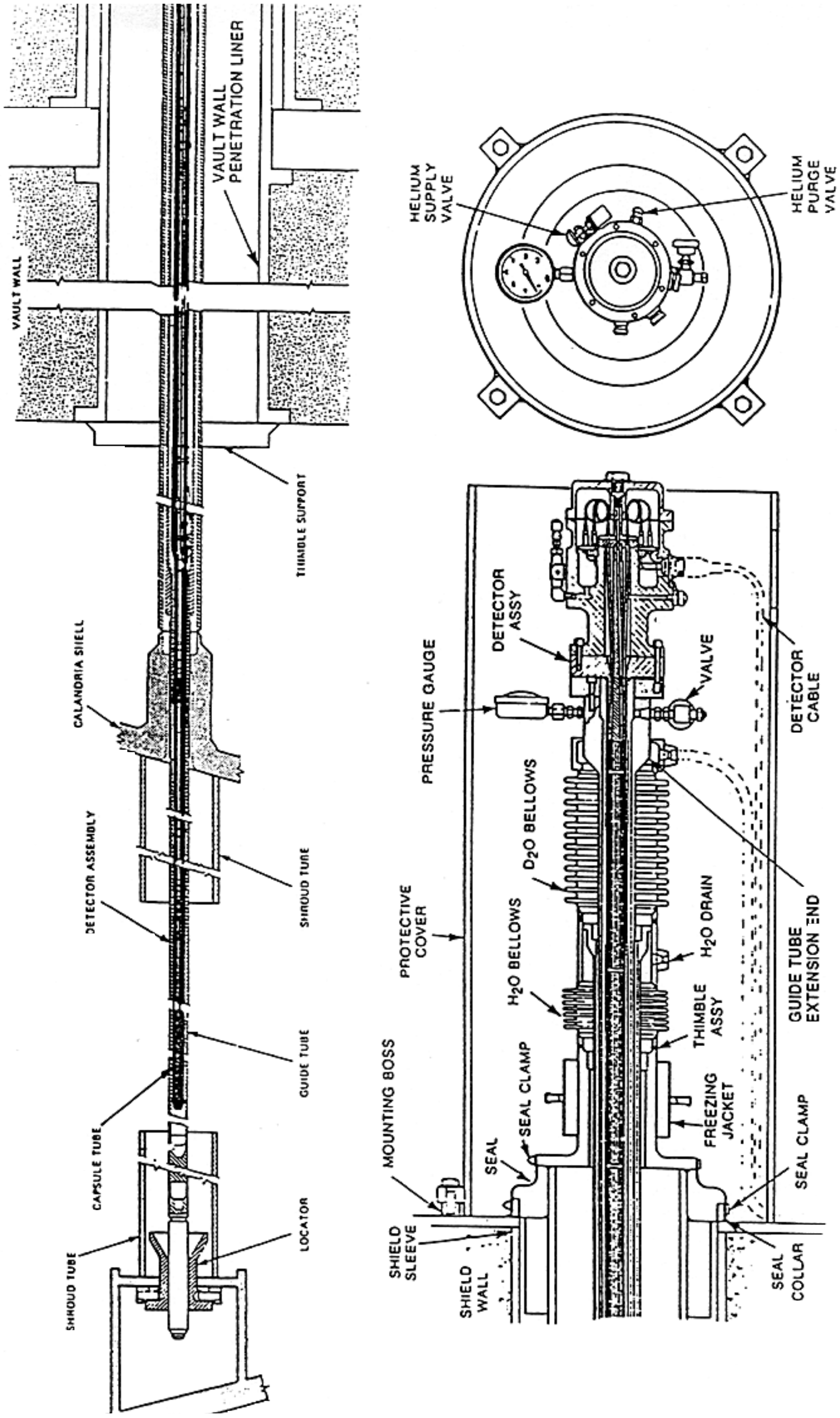
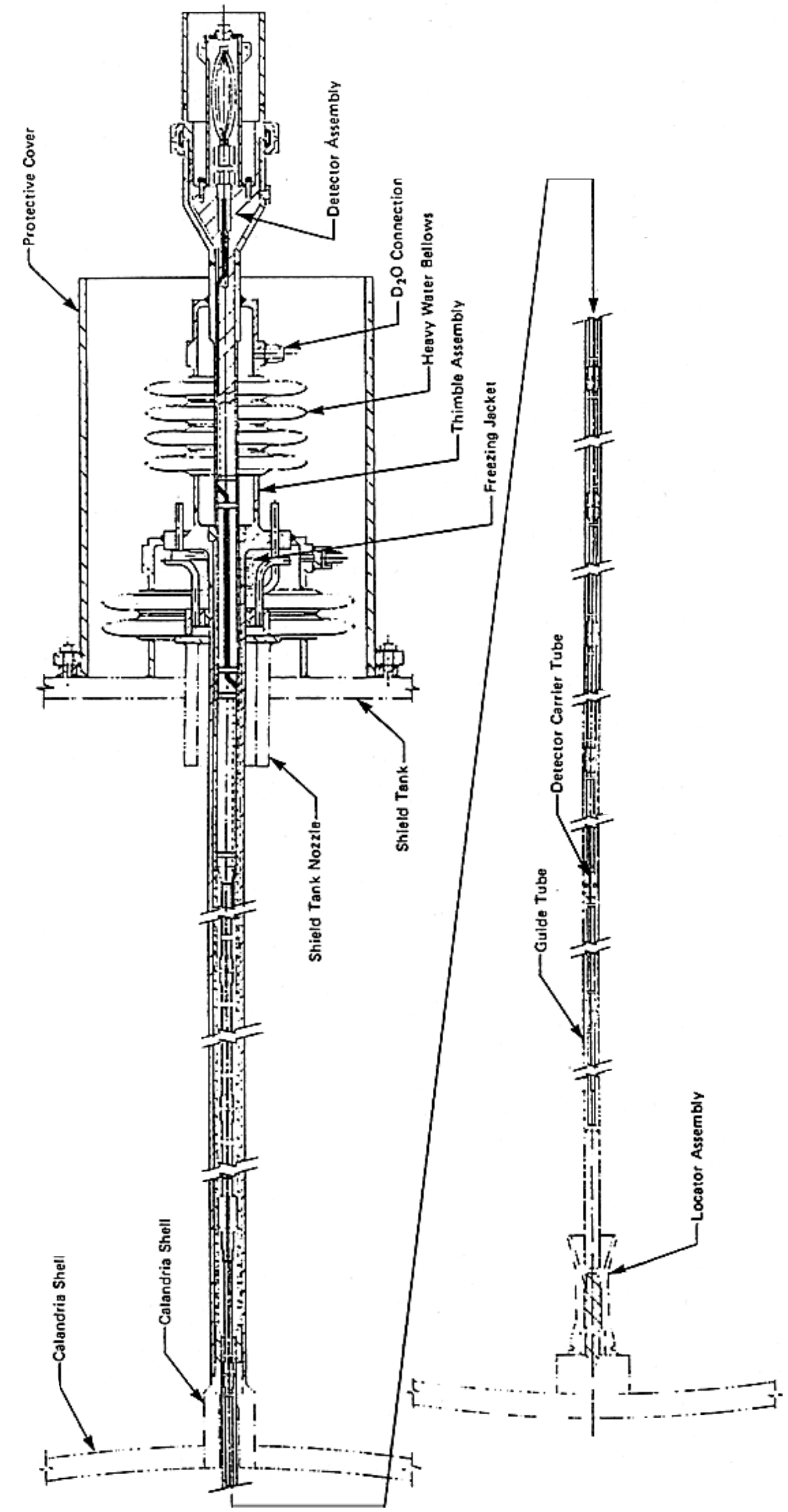


Figure 8:
Horizontal Flux Monitor Unit (Bruce A)



2.4 Calandria Support Structure

The ways of supporting and enclosing the calandria have evolved. Two of the approaches are discussed in this lesson, the CANDU 6 and Bruce NGS 'A' designs. The prominent difference between them is that the CANDU 6 calandria is installed in a reinforced concrete vault and the Bruce one in a carbon steel vessel. What follows are some details of each design.

2.5 CANDU 6

The CANDU 6 reactor assembly is installed inside a reinforced concrete vault which is lined with carbon steel (Figures 9 and 11). The end shields protrude through two sides of the vault and support the calandria within. This concrete box sits on the floor of the reactor building.

Around each end shield is a concentric, welded, support structure consisting of a cylindrical shell and an annular plate. The outer edge of the plate is welded to an embedment ring, so called because it is grouted into the opening of the concrete vault. The annular plate serves the same function as the one built into the calandria, to accommodate differential thermal expansion, in this case between the calandria and the vault. One of the support plates is fixed, or rather its flexibility is restricted, by means of bolts around the circumference tying it rigidly to the embedment ring. This ensures that the calandria/end shield structure responds as a unit in the event of a seismic event.

The concrete within and in the immediate vicinity of the embedment rings is heated by nuclear radiation from the calandria side and by radiant and convective heating from the feeder pipes. To protect it from overheating the concrete is provided with pipes for cooling water around the complete circumference, (Figure 12). Also illustrated in Figure 12 are the carbon steel slabs installed on the calandria side of the concrete to give further protection from heating. They are called the curtain and support ring shielding slabs and are bolted directly to the embedment rings. Each is approximately 152 cm thick.

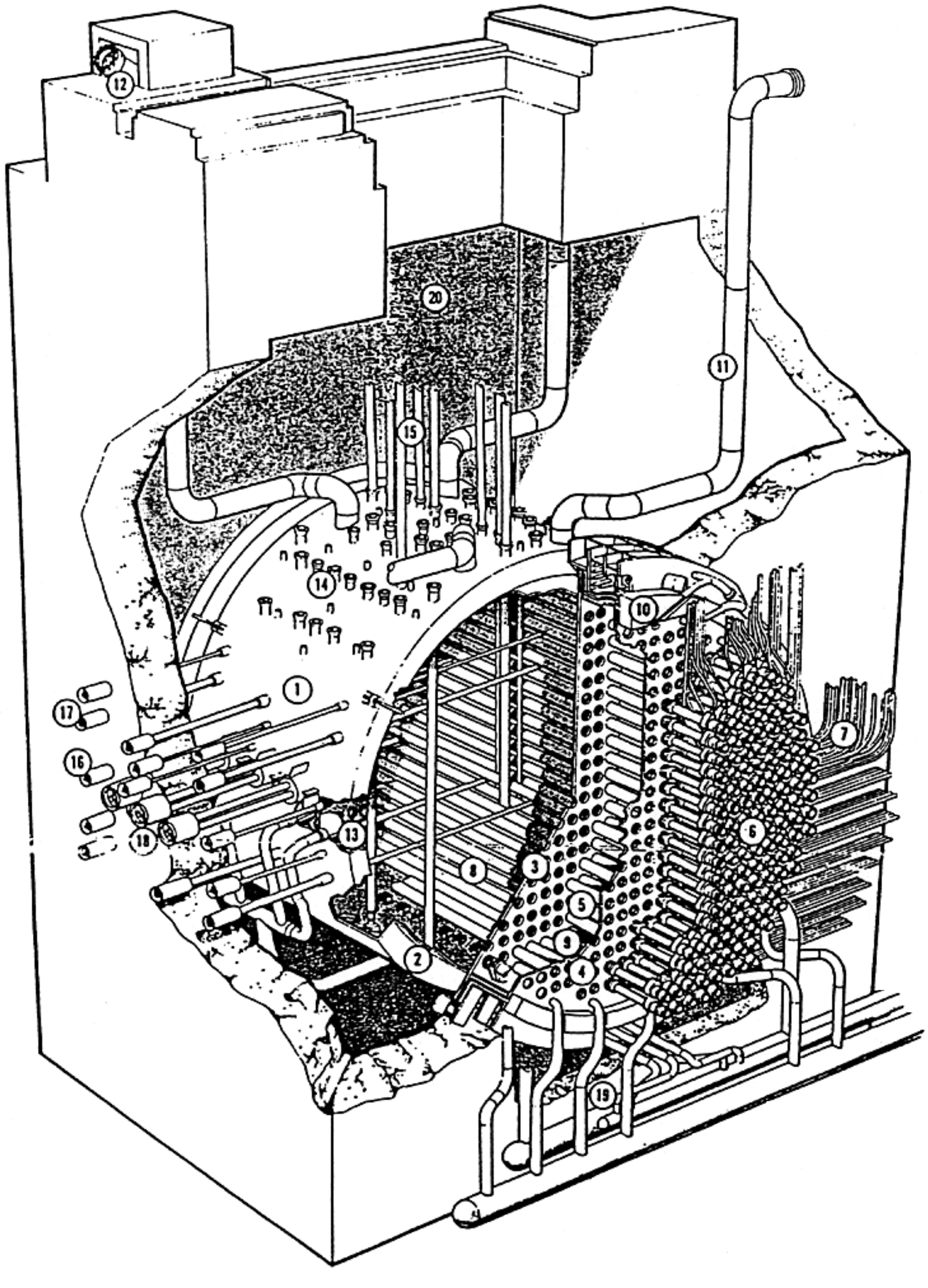
The space between the embedment ring and the support shell is packed with alternate layers of lead and stainless steel wool to reduce the possibility of radiation streaming through it into the fuelling machine vault.

The support shells and the annular support plate are fabricated from austenitic stainless steel. The embedment rings, the cooling water pipes and the shielding slabs are all made from carbon steel.

The top of the concrete vault has a rectangular stepped opening which is closed by the reactivity mechanism deck,(Figure 13). The deck which, as its name implies, supports the vertical reactivity control units (RCUs), is a concrete filled steel box suitably stiffened. There are holes through it to provide access for the RCUs. More details of these installations will be given in a discussion of the RCUs. The reactivity mechanism deck is mounted so that differential thermal expansion is accommodated at the same time as providing a stable platform, which is designed to handle all operating and seismic loads.

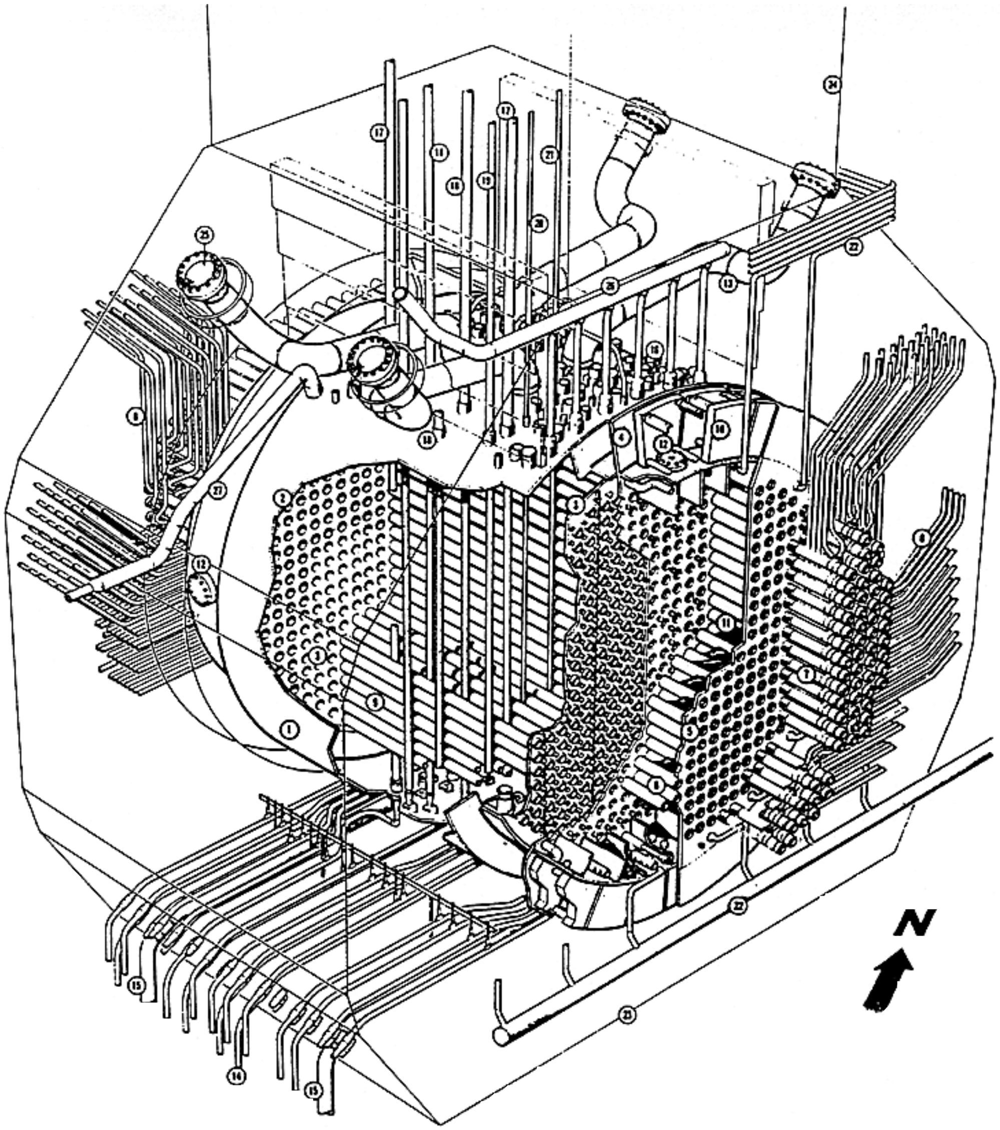
Figure 9:

Reactor Assembly CANDU 6



1	*Calandria*	2	*Calandria Shell*
3	*Calandria Side Tube Sheet*	4	*Fuelling Machine Side Tube Sheet*
5	*Lattice Tubes*	6	*End Fittings*
7	*Feeders*	8	*Calandria Tubes*
9	*Steel Ball Shielding*	10	*Annular Shielding Slab*
11	*Pressure Relief Pipes*	12	*Rupture Disc*
13	*Moderator Inlets (4 Each Side)*	14	*Reactivity Control Nozzles*
15	*Reactivity Control Devices*	16	*Horizontal Flux Detectors (9)*
17	*Poison Injector Nozzles (6)*	18	*Ion Chamber Housing (3 Each Side)*
19	*End Shield Cooling Piping*	20	*Calandria Vault (Light Water Shield)*

Figure 10:
Reactor Assembly Bruce A



1 Calandria
2 Calandria Shell
3 Calandria Side Tube Sheet
4 Baffle Plate
5 Fuelling Machine Side Tube Sheet
6 Lattice Tubes
7 End Fittings
8 Feeders
9 Calandria Tubes
10 Shield Tank Solid Shielding
11 Steel Ball Shielding
12 Manhole
13 Pressure Relief Pipes
13 Moderator Inlets
13 Moderator Outlets
16 Reactivity Control Nozzles
17 Booster Rod
18 Shut Off Rod
19 Zone Control Rod
20 Flux Monitor
21 Flux Monitor and Poison Injection
22 End Shield Cooling Piping
23 Shield Tank
24 Shield Tank Extension
25 Rupture Disc Assembly
26 Moderator Inlet Header
27 Moderator Overflow

Figure 11:
Calandria Vault CANDU 6

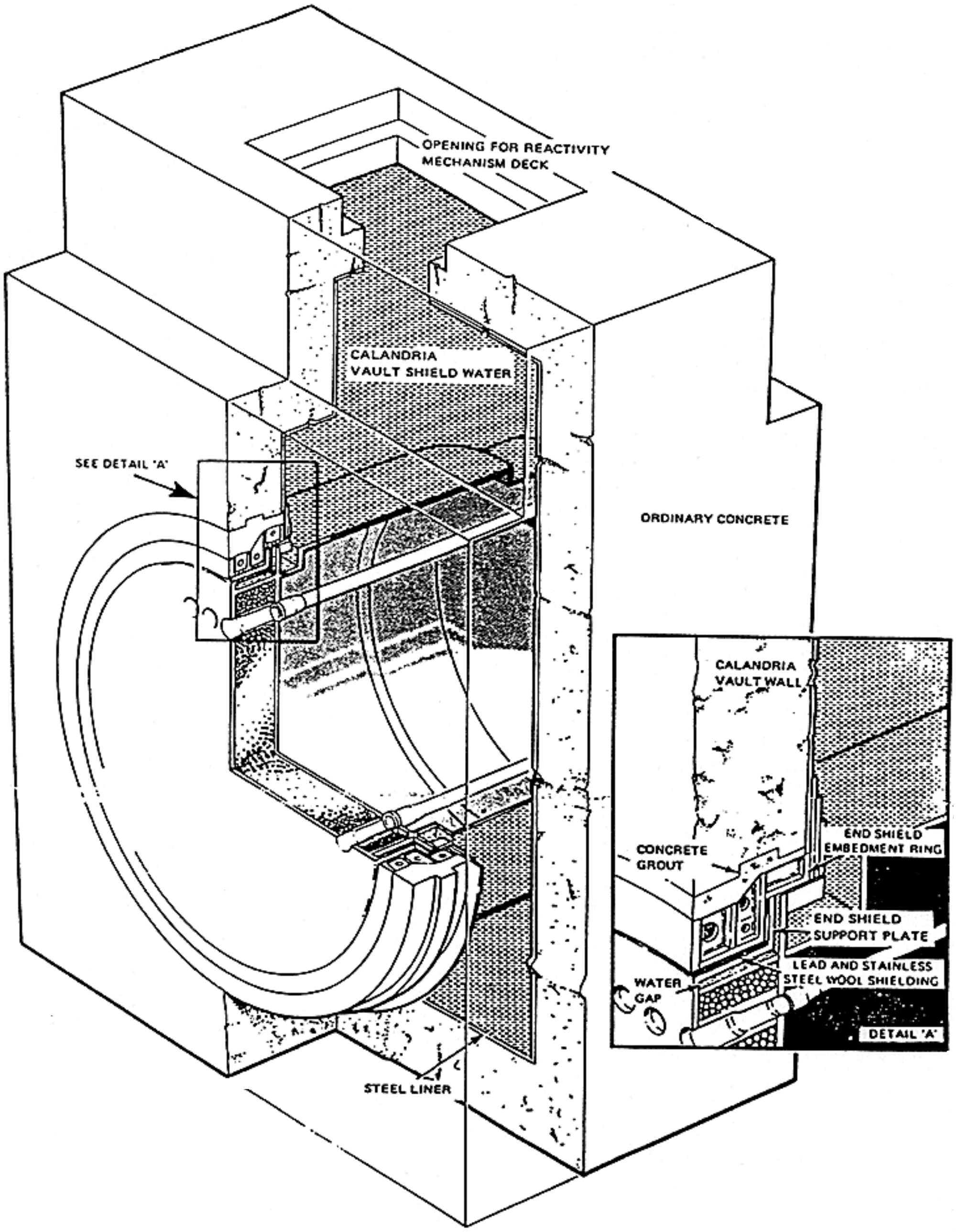


Figure 12:
Embedment Ring CANDU 6

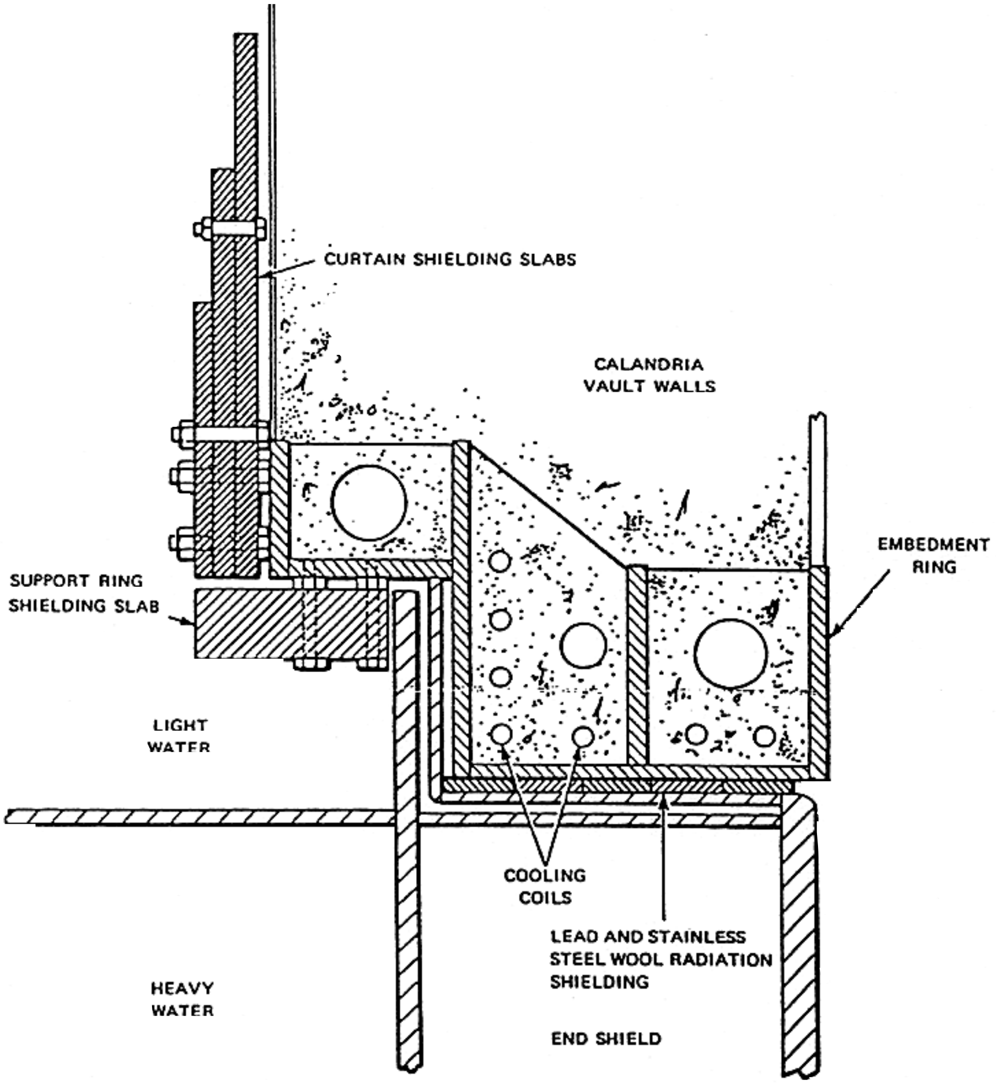
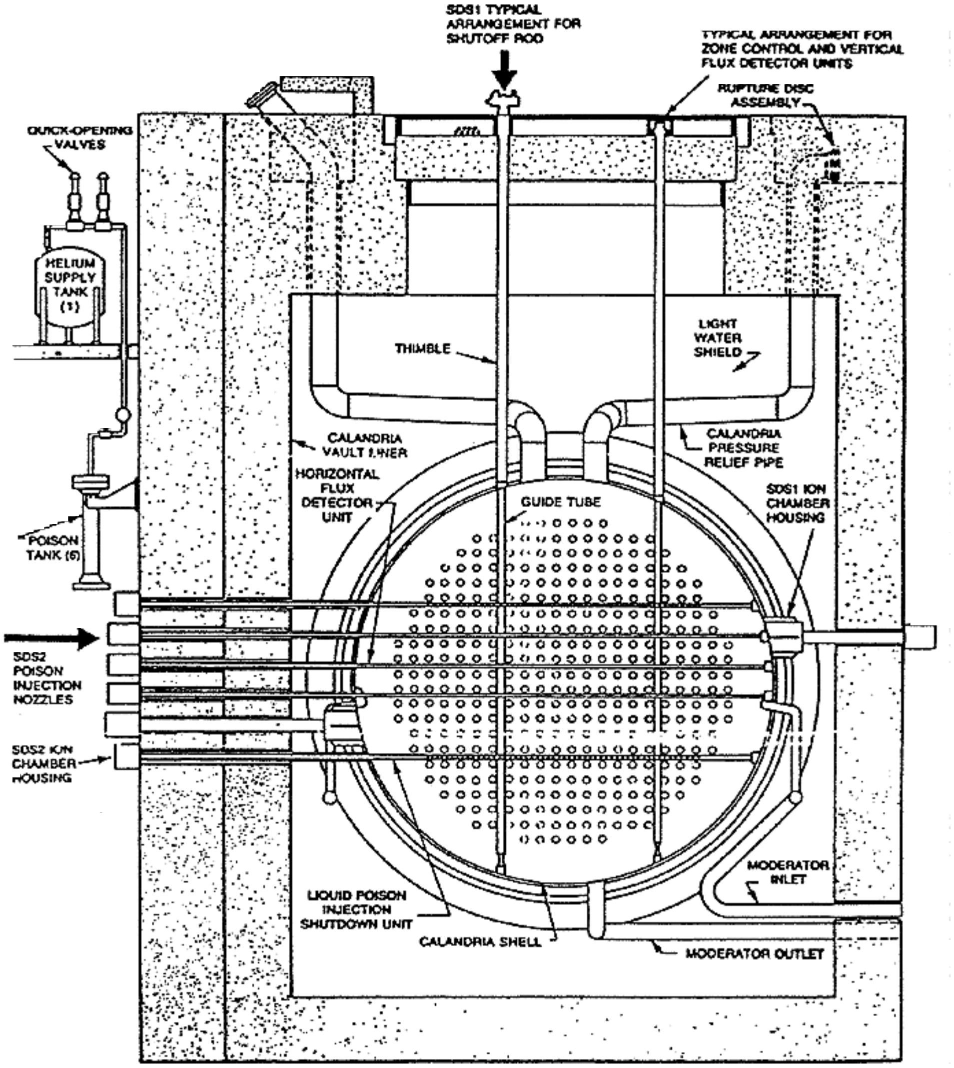


Figure 13:
Reactor Layout



The other two walls of the vault are also provided with openings to accommodate the horizontal reactivity control units, the moderator piping and the piping for the shield water.

The space between the calandria and the vault is filled with demineralized light water. The piping mentioned in the previous paragraph is used to circulate the water for the purpose of treating and cooling it. The water level is maintained at 0.3 m below the bottom of the reactivity mechanism deck and has a nitrogen cover gas above it which serves as a cushion against an increase in pressure in the shield water.

The large pressure relief ducts, mentioned in the discussion of the calandria, penetrate through the top of the vault.

2.6 Bruce 'A' Nuclear Generating Station

In the Bruce 'A' design the calandria is installed inside a welded, carbon steel tank called a shield tank. Figure 14 shows some of the details of the shield tank. It is stiffened by a variety of internal welded members. This is to ensure that it retains its shape when the space between it and the calandria is filled with demineralized light water. The end walls are double and through them are installed and welded the end shields. This is illustrated, in cross-section, in Figure 15. In this diagram can be seen steel, shielding slabs installed around the end shield.

A rectangular opening extends from the top of the shield tank providing access to, and support for, the reactivity mechanism deck (Figure 16). The deck seals the shield tank and is designed to provide radiological shielding for staff performing maintenance. It has a lower thick steel plate which forms the containment boundary for the shield tank and an upper platform, constructed of steel and heavy concrete which provides additional shielding permitting access to the reactivity mechanisms during reactor operation. There are penetrations through the deck to provide access for the reactivity control units. The walls of the extension are shielded by steel plates and steel balls.

The light water in the shield tank and also the end shields is constantly circulated through a cooler to remove the heat from the calandria.

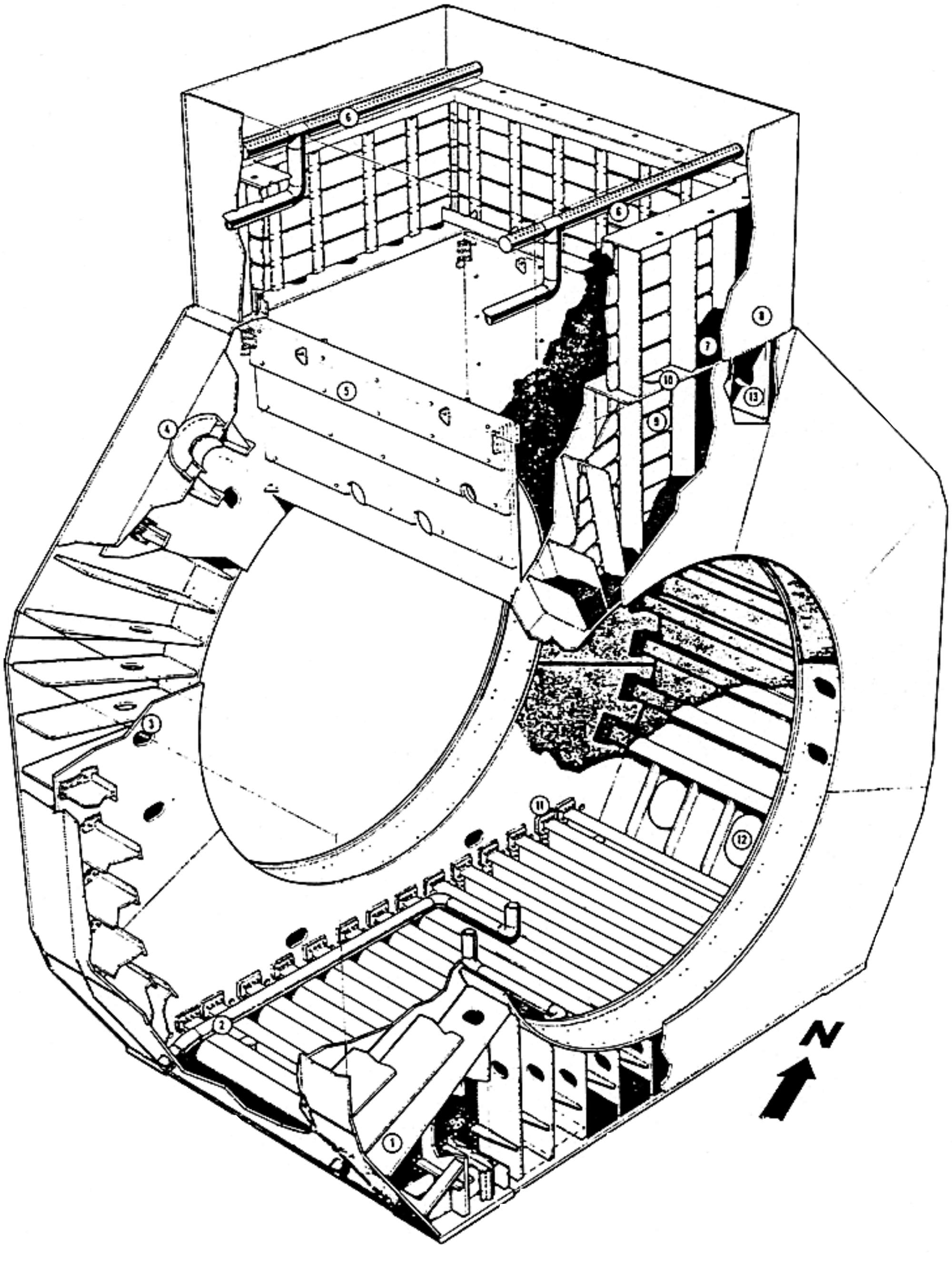
Similar to the CANDU 6 vault the shield tank is penetrated by pipes for the moderator system, the pressure relief ducts and the horizontal RCUs.

The shield tank sits on bearings at each corner which support the complete reactor assembly.

2.7 Fuel Channels

The fuel channel assemblies are supported in the calandria tubes and kept separate from them by four garter springs. The channels are an integral part of the heat transport system and form part of the pressure boundary to contain the hot pressurized heavy water coolant which is pumped through them. There is an end fitting at each end of the pressure tube, (Figure 17) . It connects the feeder pipes of the heat transport system to the pressure tube

Figure 14:
Bruce A Shield Tank



1	*Radial Stiffening Plates*	*2*	*10″ Moderator Outlet*
3	*Manholes for Construction*	*4*	*Pressure Relief Connection Opening*
5	*Curtain Shielding Slab*	*6*	*Cooling Light Water Return Pipes*
7	*Steel Shot Shielding*	*8*	*Shield Tank Extension*
9	*Steel Shot Retaining Plates*	*10*	*Manhole Cover Plate*
11	*Bolted Connection of Beams*	*12*	*Manholes Covers (for Construction)*
13	*Lifting Lugs*		

Figure 15:
Typical Section Through Calandria and End Shield

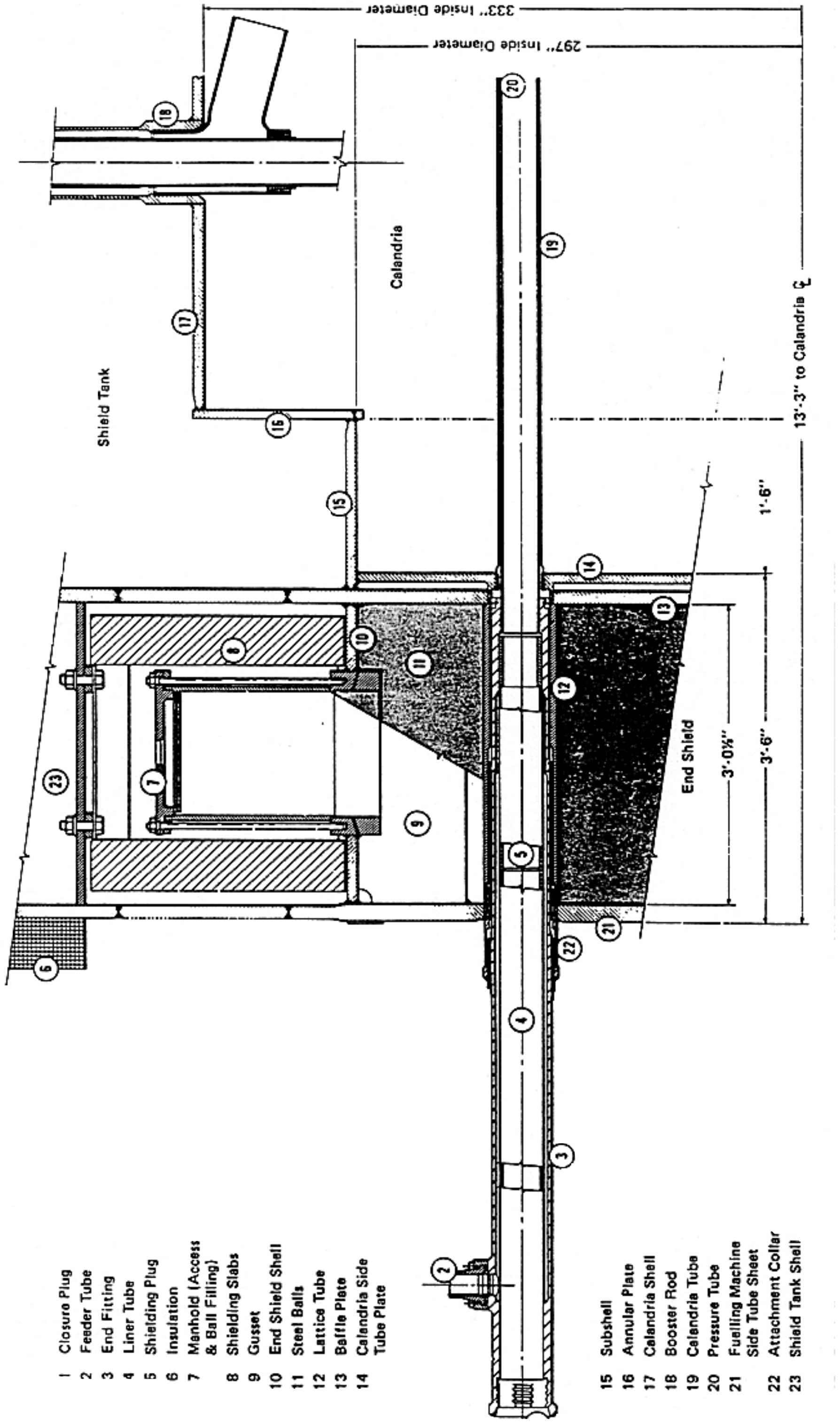


Figure 16:
Reactivity Mechanism Platform

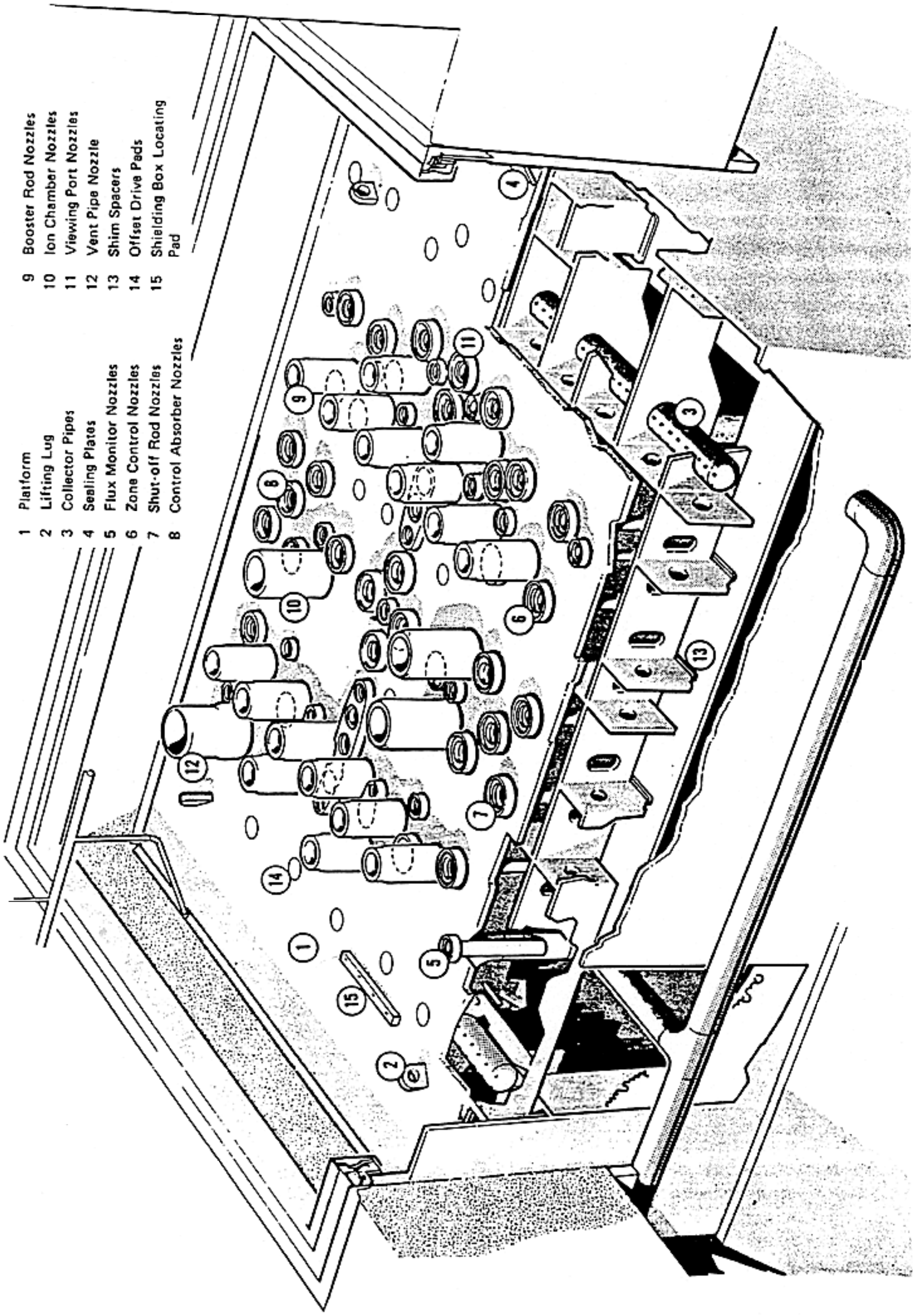
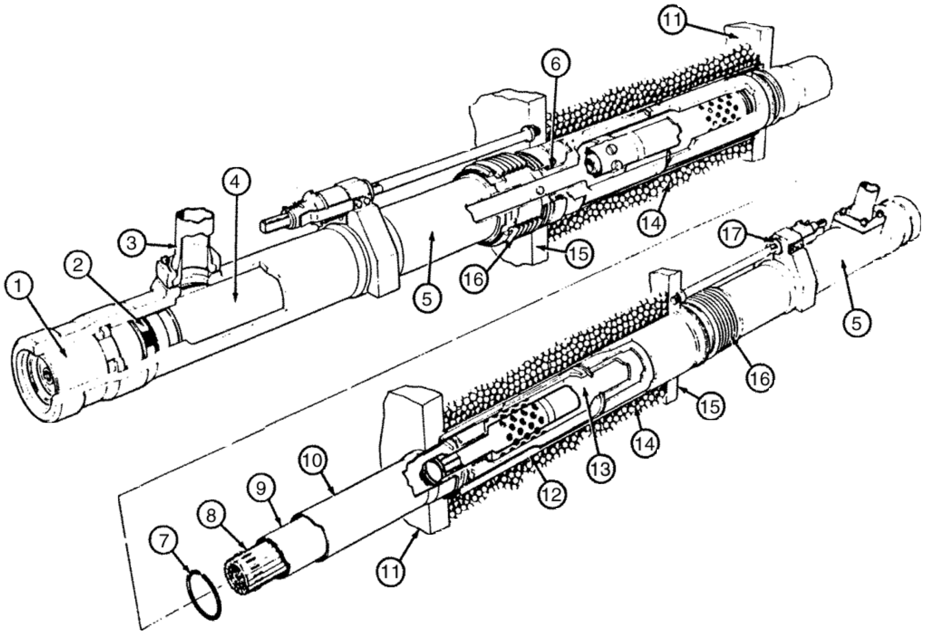


Figure 17:
Fuel Channel Assembly



1 Channel Closure
2 Closure Seal Insert
3 Feeder Coupling
4 Liner Tube
5 End Fitting Body
6 End Fitting Bearing
7 Tube Spacer
8 Fuel Bundle
9 Pressure Tube
10 Calandria Tube
11 Calandria Side Tube Sheet
12 End Shield Lattice Tube
13 Shield Plug
14 End Shield Shielding Bearings
15 Fuelling Machine Side Tube Sheet
16 Channel Annulus Bellows
17 Channel Positioning Assembly

and also permits access to the fuel in it. During fuelling, the fuelling machine seals on to the end fitting (becoming part of the heat transport system pressure boundary), removes a shield plug from it, stores the plug temporarily and so accesses the fuel. On completion of fuelling the seal plug is replaced restoring the pressure boundary and the fuelling machine can move off the end fitting.

Carbon dioxide gas is circulated in the gaps between the pressure tubes and the calandria tubes. A gap or annulus insulates the moderator from a hot pressure tube. This system is known as the 'annulus gas system'. The fuel channels and the annulus gas system are dealt with in detail in other lessons.

2.8 Reactivity Control Units (RCUs)

The reactivity control units are the in-reactor sensor and actuator portions of the reactor regulating and shutdown systems. There are neutron flux measuring devices, flux detectors and ion chambers; reactivity control devices, adjuster rods, control absorbers and liquid zone control units; shutdown devices, shutoff rods for shutdown system 1 (SDS 1) and liquid poison injection tubes for SDS 2.

Although the RCUs are parts of the reactor regulating system and the two shutdown systems they are also very much a part of the reactor assembly. As such the functions, construction, operation and location of these elements which are either in or on the calandria will be discussed here. The overall design and functions of the systems will be discussed in detail in other lessons.

Most of the discussion applies equally to the CANDU 6 and Bruce 'A' reactors. Significant differences will be noted.

2.9 RCU, RRS and SDS Requirements

- RCUs must be simple and require little maintenance
- They must survive a design basis earthquake and therefore are required to be rugged.
- All RCU drive mechanisms and power and instrumentation connections should be accessible on-power.
- In the case where a reactor regulating device fails, its failure must not cause an increase in reactivity, i.e. it must be fail-safe.
- The reactor regulating system (RRS), SDS 1 and SDS 2 must be independent of each other so that failure of one system will not affect the operation of the others. [NOTE: Measuring devices dedicated to each of these systems share the same housings but are physically and operationally independent.]
- The RRS must be able to detect and measure deviations in the local or bulk power levels and act to correct the deviation and so maintain the reactor at the desired power level.
- The RRS must be able to raise or lower the reactor power in a controlled manner including normal startups.
- There must be two fully independent shutdown systems, SDS 1 and SDS 2.
- Each SDS must be able to measure local and global flux levels and the rates of change of the levels during all normal, transient and accident conditions in the core.
- Each SDS must be able to insert into the core, neutron absorber with sufficient negative reactivity and at a sufficient rate and distribution to shut down the fission reaction under all specified accident conditions so as to prevent unacceptable damage to the reactor equipment and systems.
- Each SDS must have sufficient negative reactivity to ensure that the reactor core remains sub-critical for an indefinite period of time for all possible conditions.

2.10 Reactor Regulating System (RRS)

The purpose of the Reactor Regulating System is to control power at set point, control flux distribution and to reduce reactor power when a process system failure occurs.In addition, RRS must be able to provide negative reactivity to shut down the reactor and provide positive reactivity to override the xenon build up that occurs during a reactor power reduction.

The elements of the reactor regulating system associated with the reactor assembly are of two types, measuring units and operating units. The measuring units are flux detectors and ion chambers. Those which operate are adjuster units, liquid zone control units and mechanical control absorber units. Major design differences exist between CANDU 6 and Bruce ‘A’ RRSs. Instead of adjuster units Bruce ‘A’ has booster units, that is, its reactivity may be controlled by inserting enriched uranium control rods into the reactor core. On all other CANDU reactors the control is by removing or withdrawing a neutron absorbing rod. Bruce ‘A’ boosters will be described in more detail later in the lesson however they are slated to be removed because of safety related operational concerns.

With the exception of the ion chambers all of the RCUs for the RRS are installed vertically. The ion chambers on the Bruce ‘A’ reactors are installed vertically but on the CANDU 6 they are installed horizontally.

2.11 Shutdown Systems (SDS)

The purpose of the SDS to detect any failure that is beyond the process system (RRS) capability and quickly insert a large amount of negative reactivity to reduce the reactor heat production to a safe shutdown level.

The shutdown systems also have flux detectors and ion chambers as measuring units. SDS 1 uses mechanical shutoff rods which are dropped rapidly into the core to shutdown the reaction. SDS 2 has injection nozzle tubes through which a neutron-absorbing solution is injected into the reactor to shut it down.

The flux detectors and shut off rods for SDS 1 are installed vertically. The ion chambers for CANDU 6 are installed horizontally for both SDS 1 and SDS 2. The Bruce ‘A’ design placed the ion chambers for SDS 1 on a vertical axis on top of the calandria.

The flux detectors, ion chambers and liquid injection nozzle tubes for SDS 2 are horizontal.

2.12 Location of Reactivity Control Units

The location of the RCUs are shown in Figures 18 and 19 for CANDU 6 and in Figures 20 and 21 for Bruce ‘A’. In general they are located to provide the best overall control of the power level.

2.13 RCU Access To Core

Except for the ion chambers, all RCUs must have access to the reactor core. This is provided by guide or access tubes which span the calandria between the calandria tubes, and which are extended up through the reactivity mechanism deck for vertical units or out through the side walls of the vault (CANDU 6) or the shield tank (Bruce ‘A’) for horizontal units. Figure 6 shows the general set up for these installations.

To provide the access the calandria shell is manufactured with nozzle openings and opposite each opening, on the inside of the shell, is a locator bracket. Figure 3 shows these details.

A thimble is welded to each nozzle which extends the calandria boundary through the Reactor Maintenance (RM) deck for vertical units or through the wall of the vault or shield tank for horizontal units. It is sealed to the deck opening using a flexible metal bellows which completes the boundary between the vault nitrogen cover gas and the atmosphere above the RM deck. The opening into the calandria is sealed by an elastomeric sealant between the end of the thimble and the RCU drive or connector housing which is bolted directly to it.

Guide tubes are inserted through the thimble and are fastened at the bottom to the locator brackets. At the top they are supported in a seat in the thimble, close to the entrance to the calandria. These tubes are made of light zircaloy and are highly perforated. At the lower end the tubes are maintained in tension by a spring which is compressed during installation.

A stainless steel guide tube extension is used to continue the bore through the thimble to the RM deck. It is seated on the top of the guide tube and is located at the top by a bearing and a light spring. A shield plug is seated on top of the assembly to prevent neutron and gamma radiation from streaming up from the core of the reactor and ensures that the deck remains safely accessible.

2.14 Flux Detectors

The purpose of the flux detector units is to provide the RRS with linear flux signals for bulk and spatial power control when the reactor is above approximately 15% FP and to provide linear flux signals for the reactor high power trips of SDS1 and SDS2.

Flux detector units consist of a capsule tube, of the length of the guide tube, divided into twelve wells into which are inserted individual, self-powered detectors (Figure 22). Each detector is replaceable and the unit is known as a straight, individually replaceable (SIR) flux detector. The capsule tubes are inserted into the guide tubes giving the detectors access to the reactor core. Figure 23 is an example of a vertical flux detector unit and Figures 7 and 8 are examples of horizontal units.

Figure 18:
Location of Vertical Reactivity Control Units (CANDU 6)

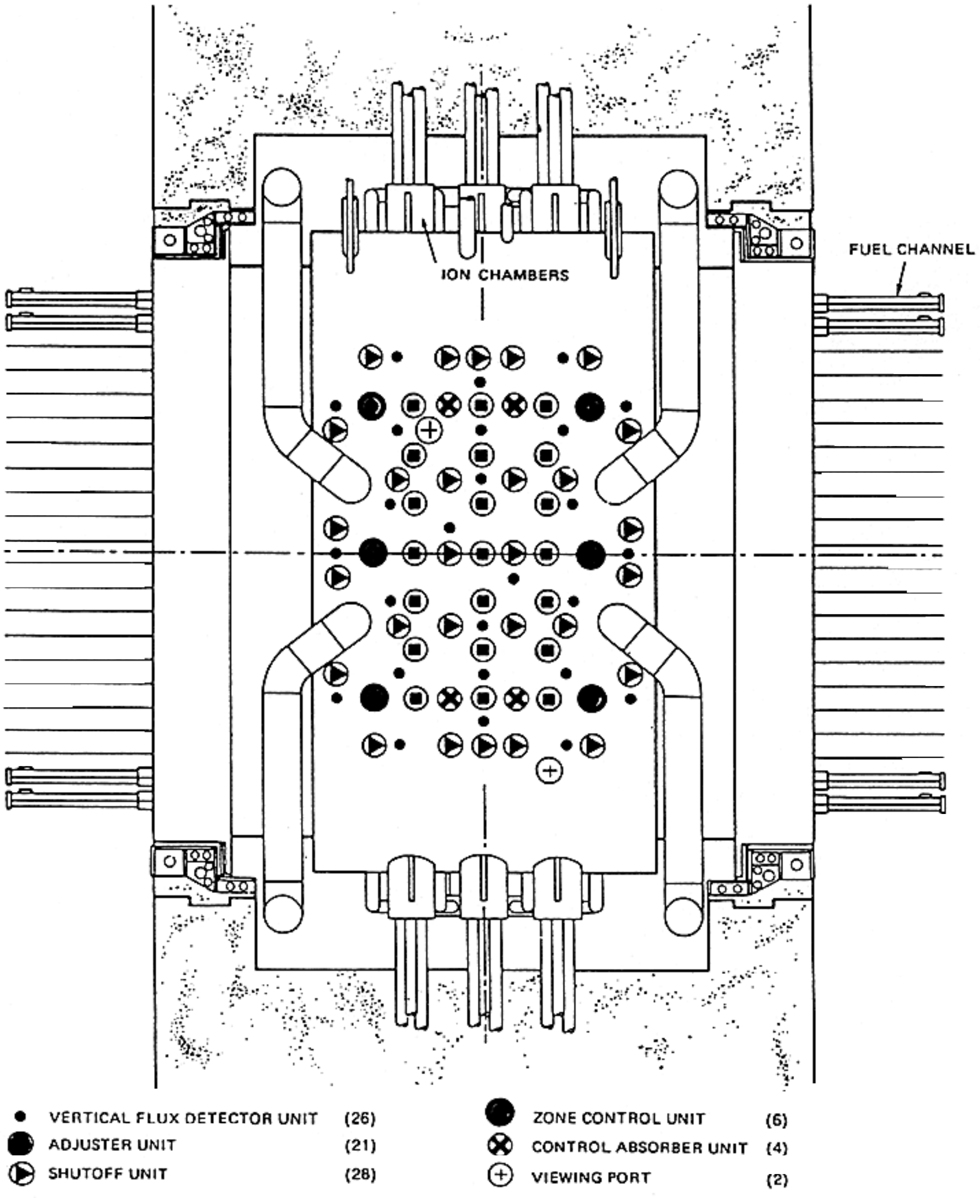


Figure 19:
Location of Horizontal Reactivity Control Units (CANDU 6)

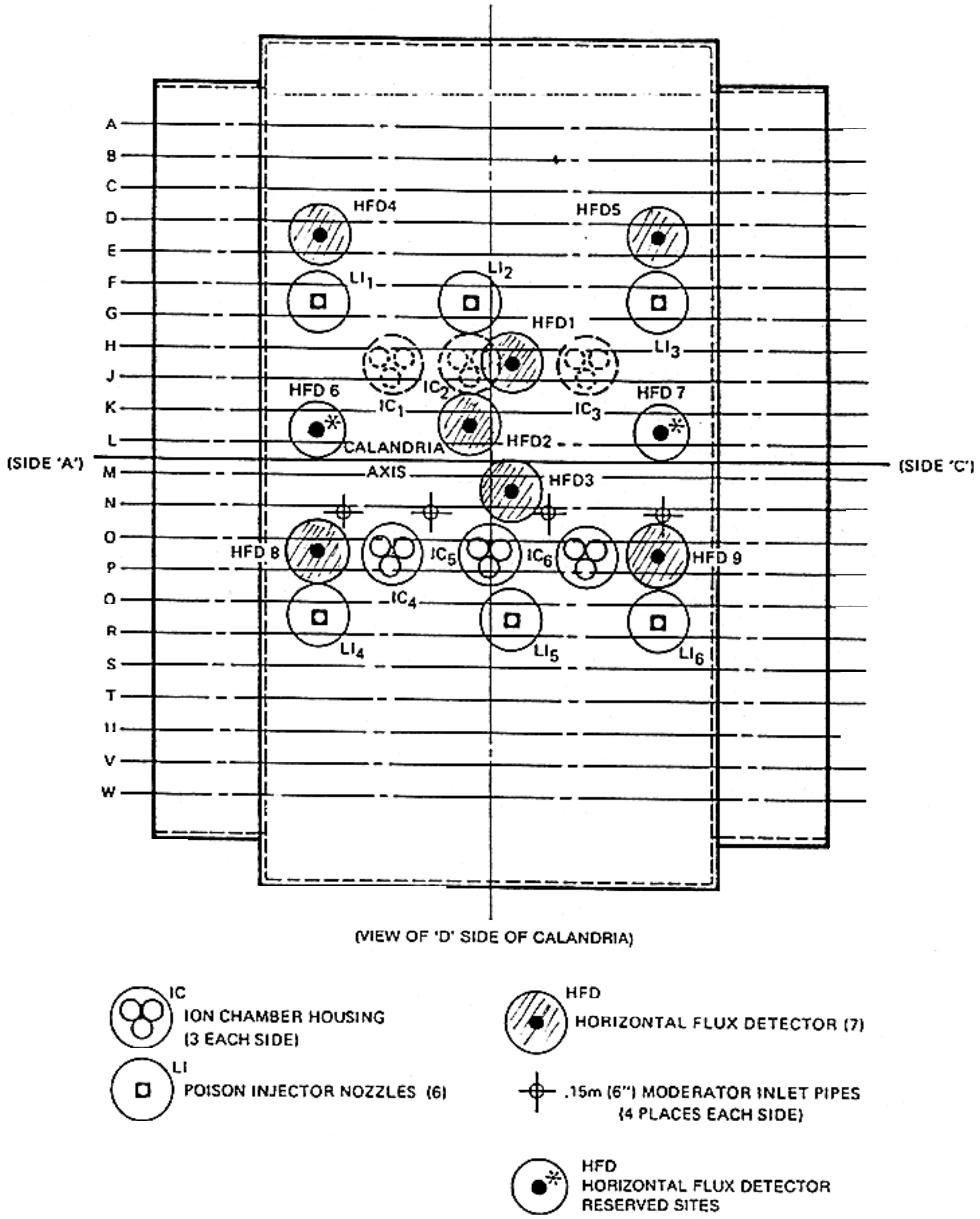


Figure 20:
Location of Vertical Reactivity Control Units (Bruce A)

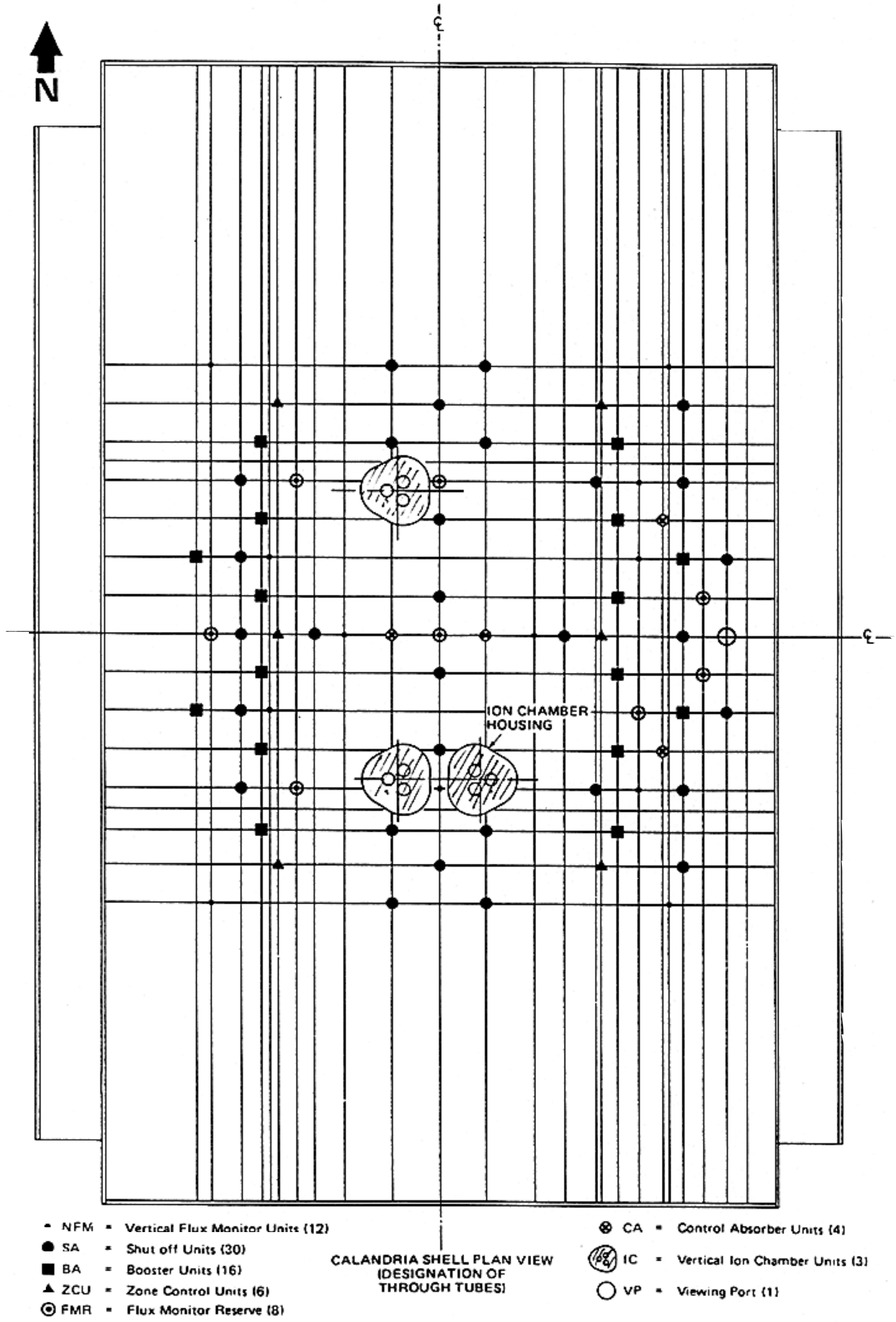


Figure 21:
Location of Horizontal Reactivity Control Units (Bruce A)

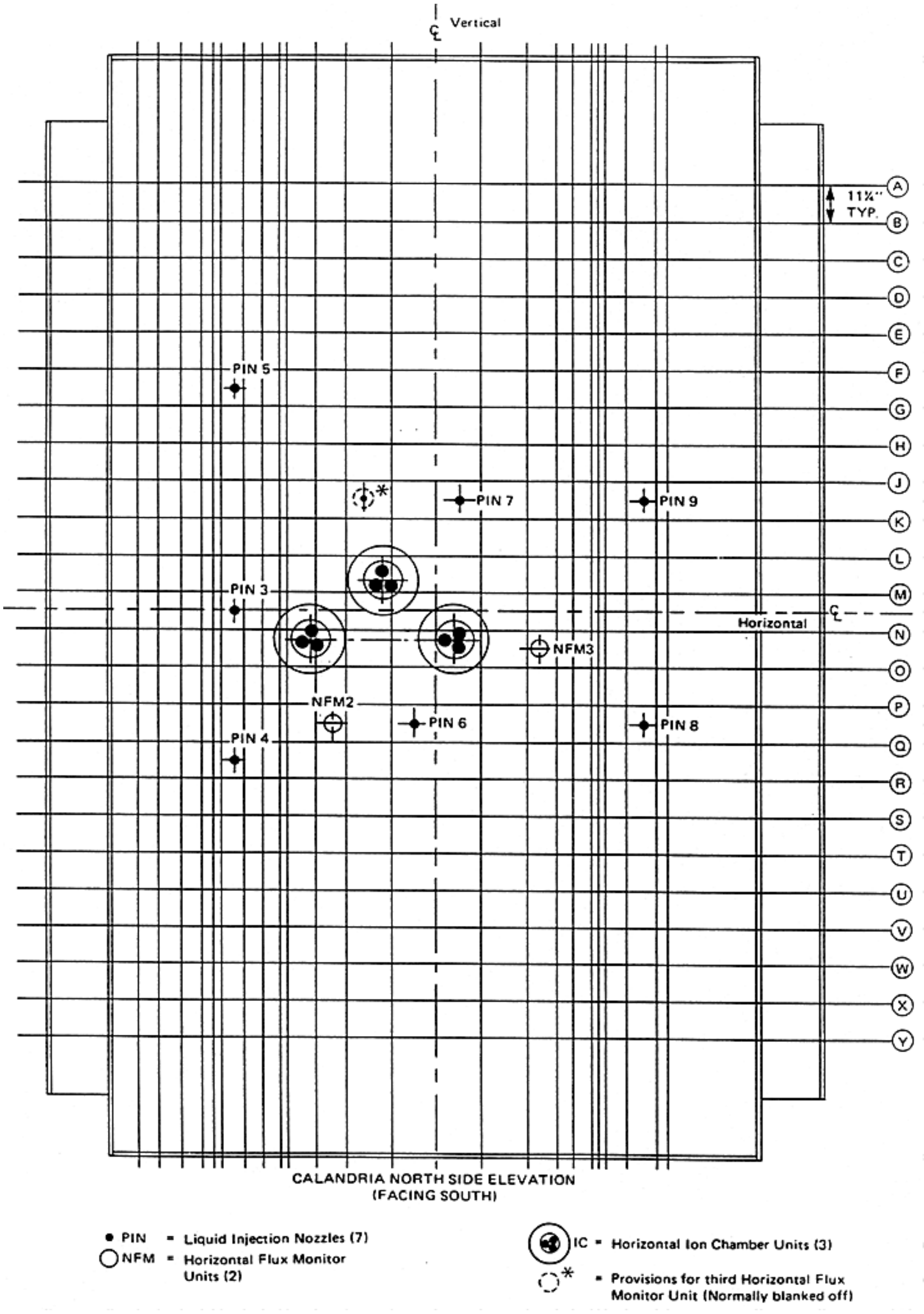


Figure 22:
Vertical Flux Detector Assembly

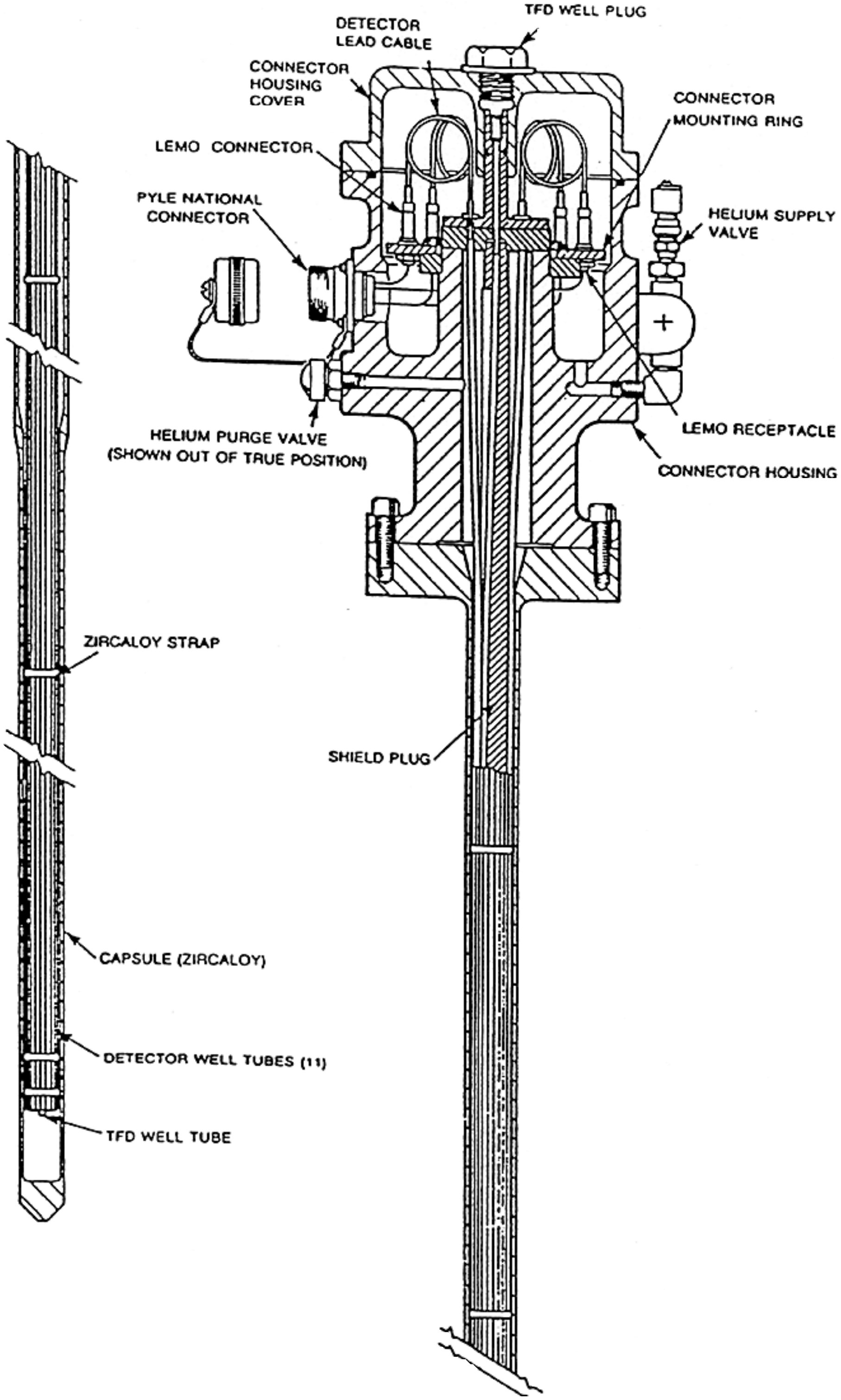
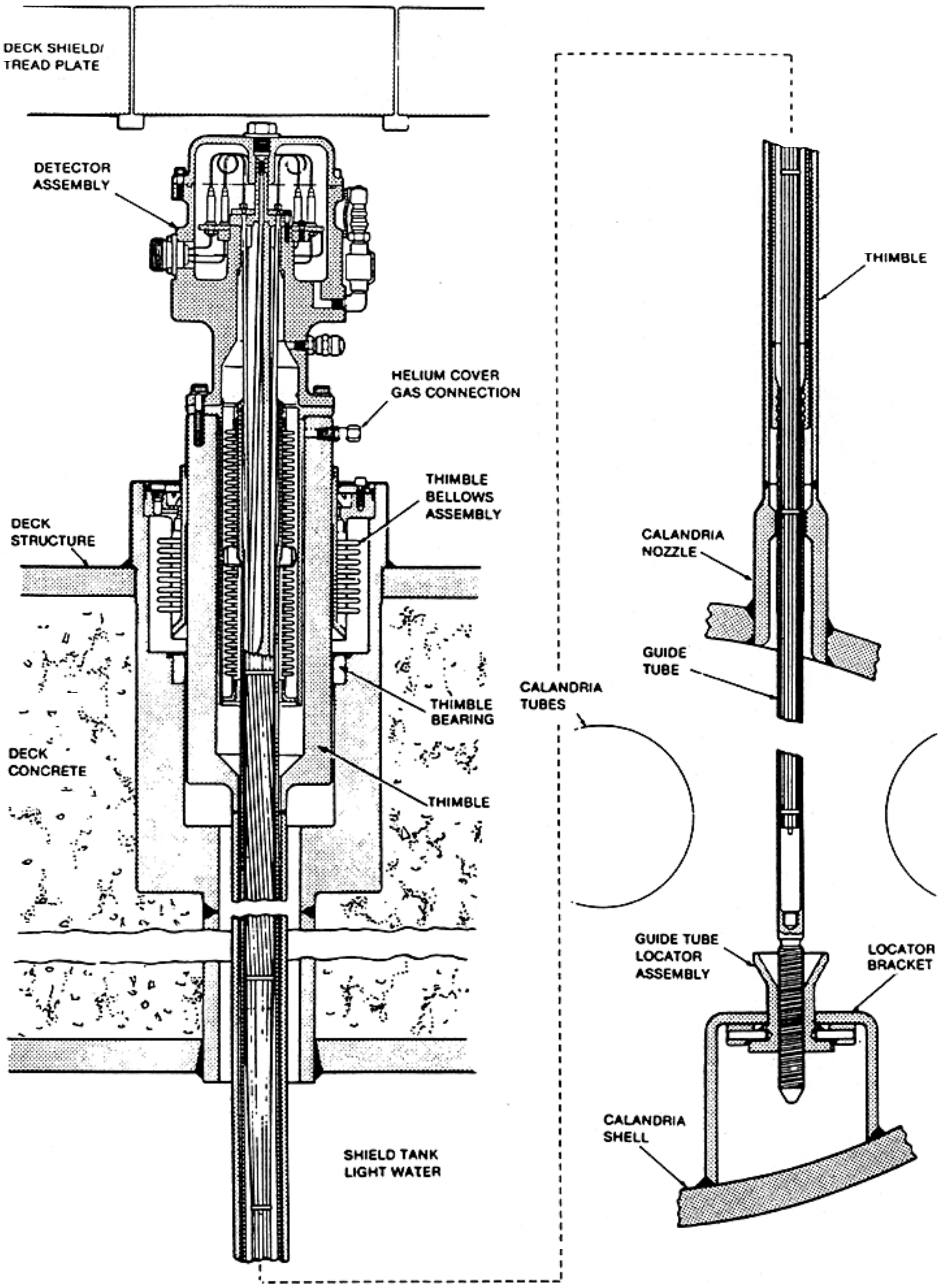


Figure 23:
Vertical Flux Detector Unit



Two types of detectors are used in the units. One responds quickly and is sensitive to both gamma and neutron fluxes, the other is slower and sensitive to only neutron flux. Each produces a continuous signal proportional to the fission rate in the reactor. Fast response detectors are used for control and safety. The slower detectors are used for flux mapping and long term control.

Each flux detector element consists of an emitter which has an inconnel core clad with platinum. This is surrounded by a magnesium oxide insulator and sheathed in inconnel. A lead of similar construction conducts the signal to the housing connection (Figure 24).

The vertical flux detector units contain detectors dedicated to either the reactor regulating system or to shutdown system 1. The horizontal flux detector units are dedicated to shutdown system 2. This arrangement is the same for both CANDU 6 and Bruce ‘A’. Units 3 and 4 of Bruce ‘A’ use detectors of a different design than described here.

2.15 Ion Chambers

Ion chamber units consist of lead-shielded housings mounted on the outside of the calandria shell in which ion chamber instruments and calibration shutters are installed. The lead attenuates gamma radiation so that the ion chambers primarily measure neutron flux. The purpose of the ion chamber is to measure both the bulk flux level and the rate of change of the level and provide inputs to both the RRS and the SDSs. Figure 25 illustrates a typical horizontal ion chamber installation.

On the CANDU 6 reactors the ion chambers are mounted horizontally. Those for SDS 2 on the accessible side. In the CANDU 6 design an additional concrete wall is constructed against one wall of the vault. It attenuates the radiation sufficiently that the instrumentation which penetrates through it is accessible while the reactor is at power,(Figure 13). The ion chamber housings on the inaccessible side contain elements dedicated to the RRS and SDS 1.

Bruce ‘A’ has ion chambers mounted horizontally, which provide inputs to SDS 2, and vertically, which provide dedicated inputs to the RRS and SDS 1.

2.16 Adjuster Units (AA)

The functions of the AA rods is to provide excess reactivity for xenon poison override and fuelling machine failures and to shape the neutron flux to optimize reactor power production.

Adjuster units are operating elements of the RRS in a CANDU 6 reactor.

An adjuster unit consists of a tubular, stainless steel, neutron absorbing element which is raised and lowered within a guide tube. It is suspended from a stainless steel cable which is wound around the sheave of a drive mechanism. The drive mechanism is bolted to the top of the thimble and, because it forms part of the pressure boundary of the calandria, it is sealed. There is a shaft seal to isolate the cable sheave cavity from the rest of the drive mechanism, (Figures 26 and 27).

The stainless steel absorbers are normally kept in core in a fixed normal position. Their function is to depress the flux somewhat in the centre of the core to provide a more uniform level of flux across the reactor.

Figure 24:
Platinum Clad SIR Flux Detector Element

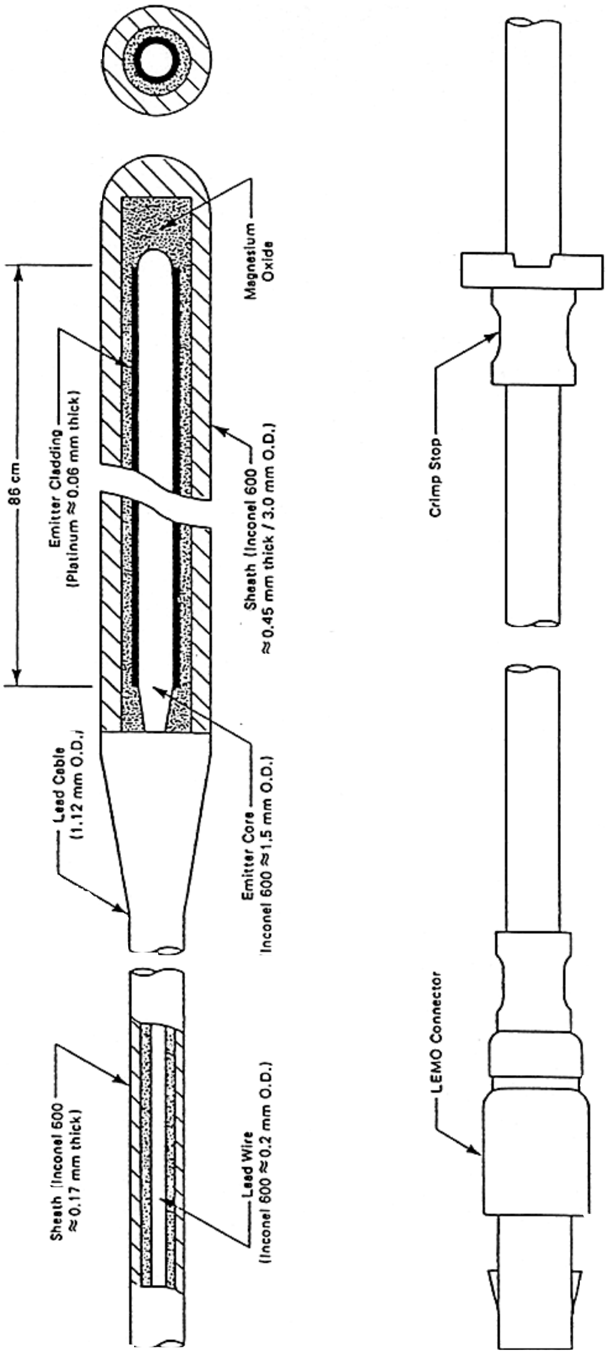


Figure 25:
Horizontal Ion Chamber Mounting

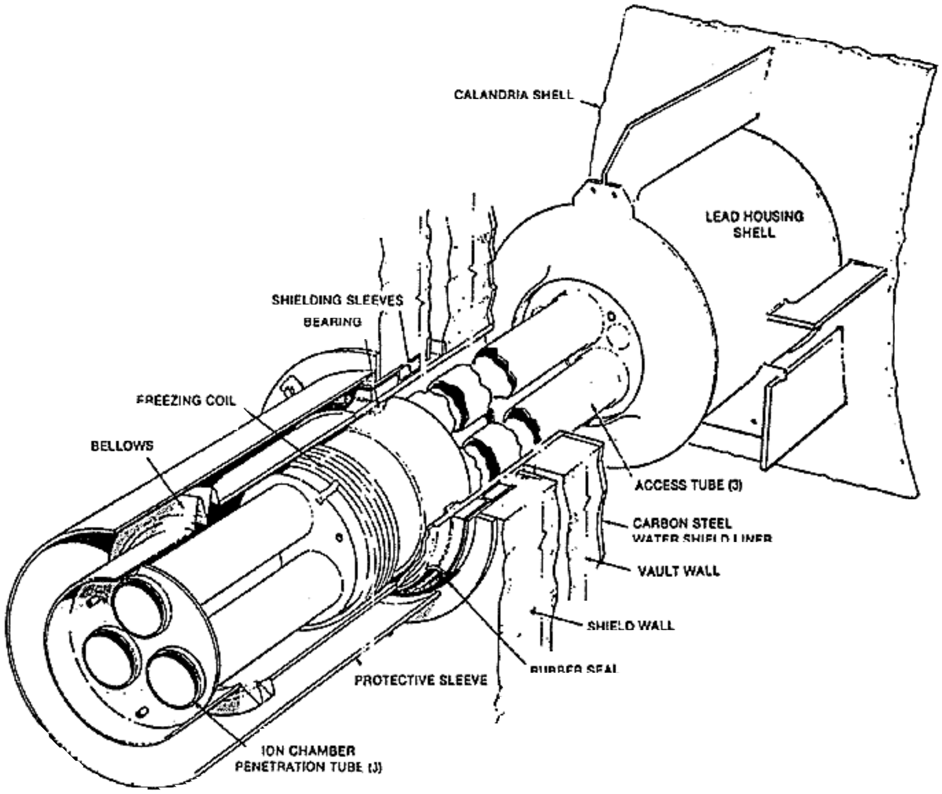


Figure 26:
Adjuster Unit (CANDU 6)

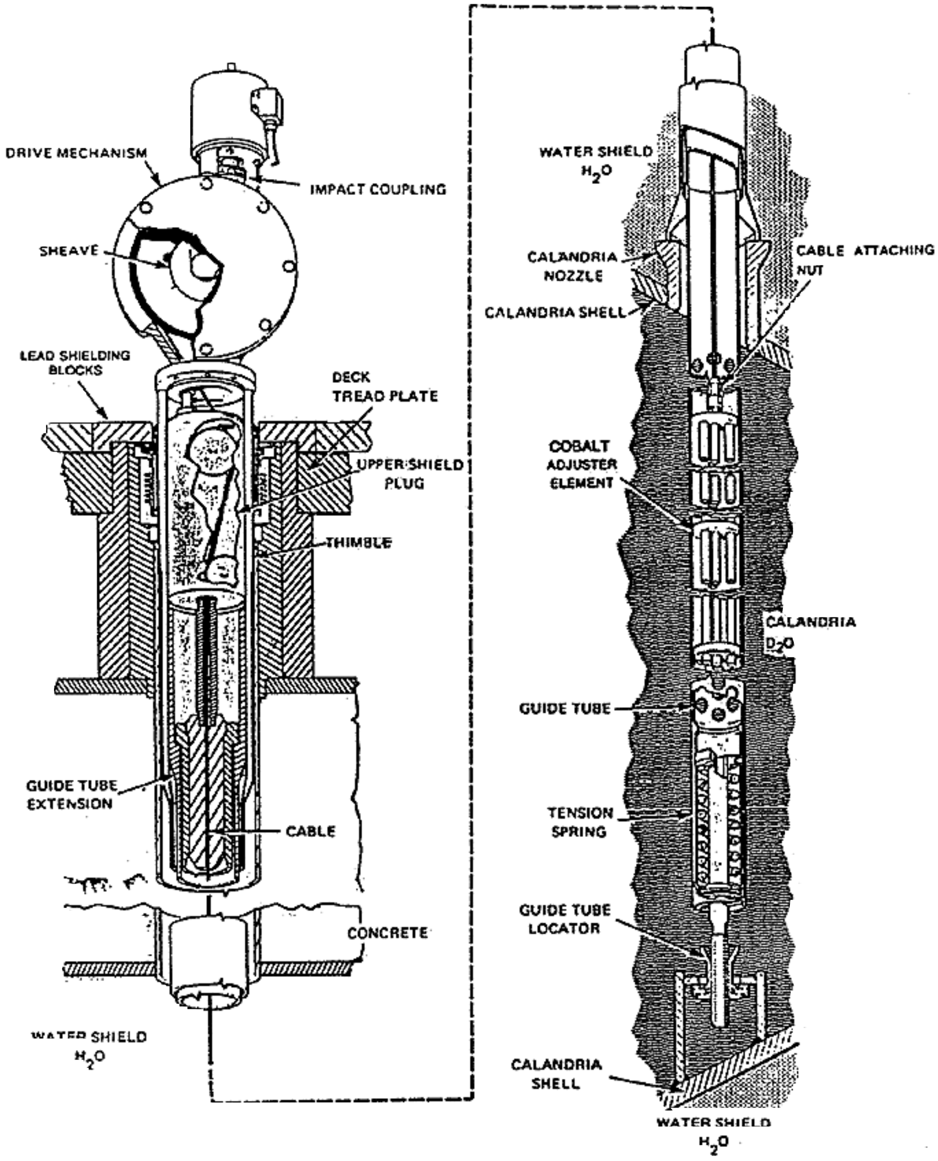
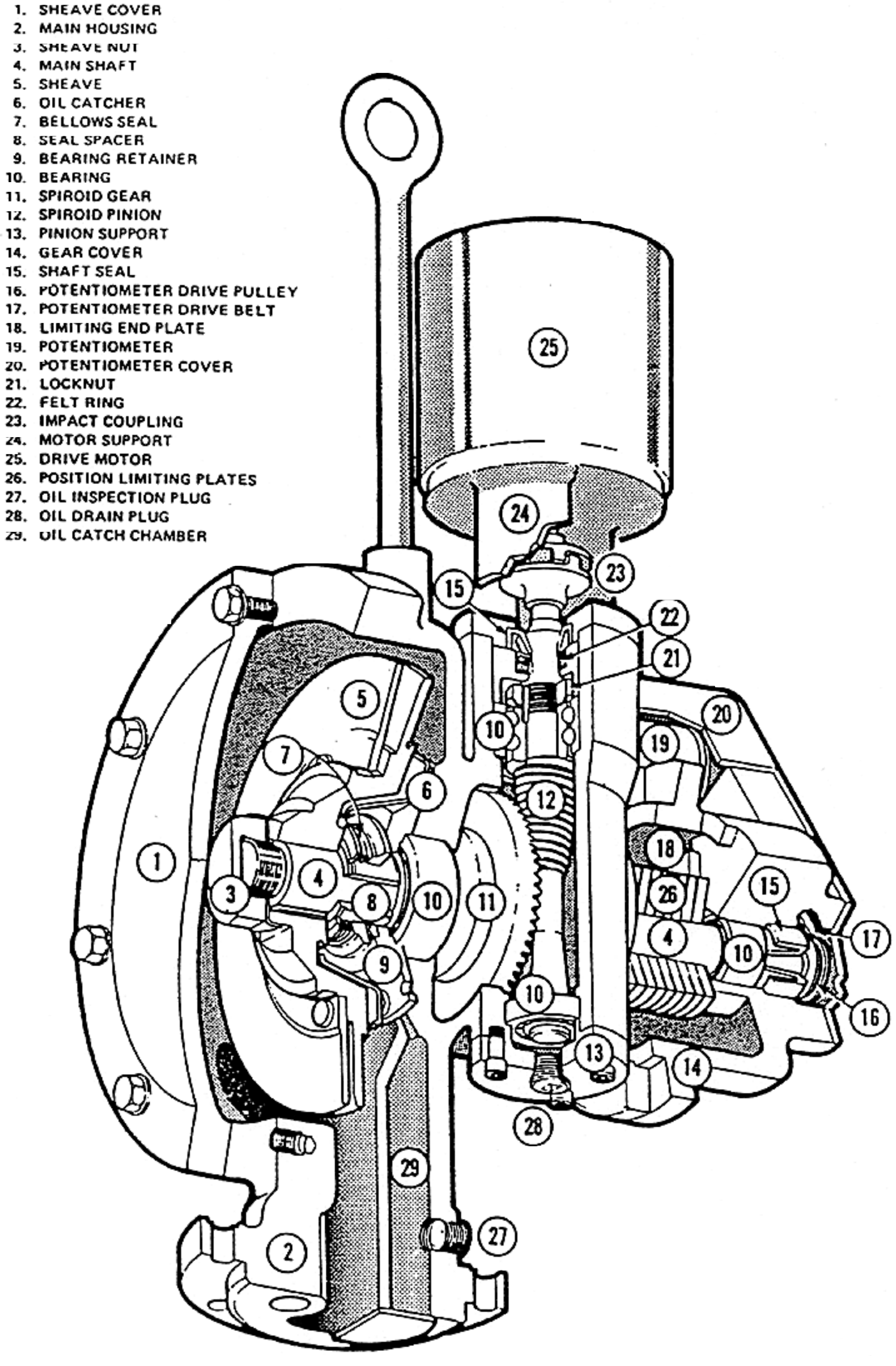


Figure 27:
Adjuster Drive Mechanism (CANDU 6)



1. SHEAVE COVER
2. MAIN HOUSING
3. SHEAVE NUT
4. MAIN SHAFT
5. SHEAVE
6. OIL CATCHER
7. BELLOWS SEAL
8. SEAL SPACER
9. BEARING RETAINER
10. BEARING
11. SPIROID GEAR
12. SPIROID PINION
13. PINION SUPPORT
14. GEAR COVER
15. SHAFT SEAL
16. POTENTIOMETER DRIVE PULLEY
17. POTENTIOMETER DRIVE BELT
18. LIMITING END PLATE
19. POTENTIOMETER
20. POTENTIOMETER COVER
21. LOCKNUT
22. FELT RING
23. IMPACT COUPLING
24. MOTOR SUPPORT
25. DRIVE MOTOR
26. POSITION LIMITING PLATES
27. OIL INSPECTION PLUG
28. OIL DRAIN PLUG
29. OIL CATCH CHAMBER

This causes all the fuel to be fissioned at about the same rate and permits the core to produce more power when the general flux level can be raised everywhere to a level corresponding to the peak rated power for the fuel.

2.17 Booster Units (Bruce 'A' Only)

The RRS at Bruce 'A' has booster units instead of adjuster units.

Although the booster units in the Bruce 'A' reactors are to be removed at the upcoming rehabilitation a brief description is given here.

A booster unit consists of rod of enriched uranium fuel bundles which can be lowered and raised within a guide tube by a drive mechanism. The 'rod' is shown in Figure 28 and a complete unit in Figure 29. The fuel bundle rod is attached to the rack of a rack and pinion drive which gives very positive control of its motion and location.

The booster fuel rod is normally parked in the guide tube extension outside of the core. To keep this enriched fuel cool moderator coolant is pumped up through the guide tube and extension where it exits through ports at the top into the annulus between the extension and the thimble and returns to the calandria through a nozzle at the bottom of the thimble. As in the case of the adjusters the drive mechanism is sealed to prevent leakage of the coolant.

The function of the booster unit is to provide additional reactivity when needed either in overcoming increases in Xe-135 following a reduction in power or when refuelling equipment is not available. There are stringent regulations regarding the use of booster units which leads to their infrequent use and therefore the plans to remove them.

2.18 Liquid Zone Control (LZC) Units

The purpose of the LZC units are to provide the RRS with the primary reactivity mechanism - neutron-absorbing light water rods of continuously variable length for controlling both bulk and spatial power distribution in the reactor core.

Liquid zone control units are other in-core parts of the RRS.

They consist of a closed tube, divided into separate compartments along its length, which is inserted into the reactor core through a thimble tube (Figure 30). Each of the compartments can be filled, independently, to any level with light water. The light water is supplied by internal piping to each compartment from the piping terminal head which is bolted to the top of the thimble.

As can be seen from Figures 18 and 20 there are six LZC units. The two on the centre line of the reactor have three compartments each and the other four have two compartments. This provides an ability to adjust the flux level in any one of the fourteen zones illustrated in Figure 31 by controlling the level of water in the appropriate compartment.

Each compartment is supplied with a continuous supply of light water and the level is determined by controlling the outflow.

All of the in-core components are manufactured from zirconium alloy.

Figure 28:
Booster Fuel Assembly (Bruce A)

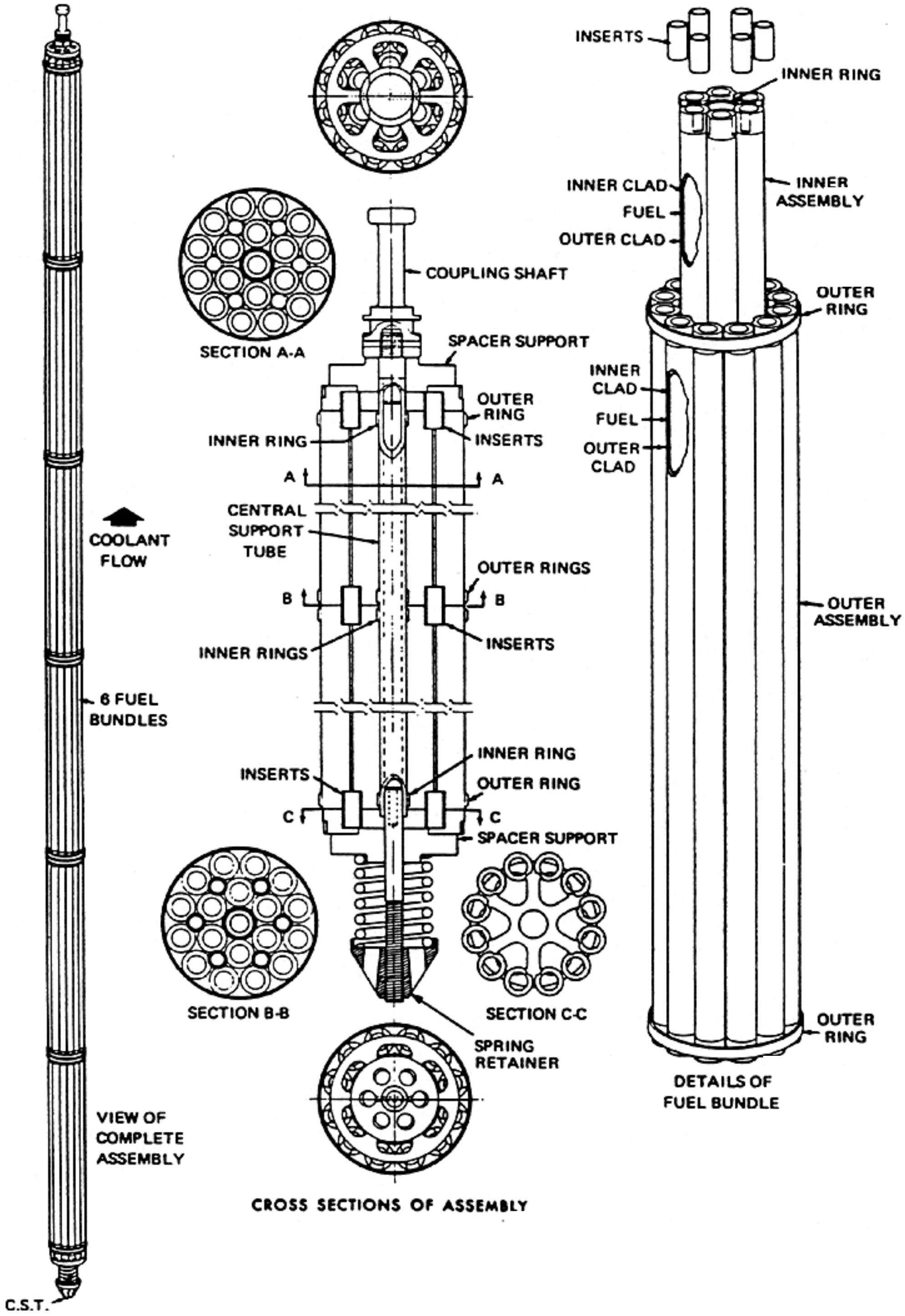


Figure 29:
Booster Unit (Bruce A)

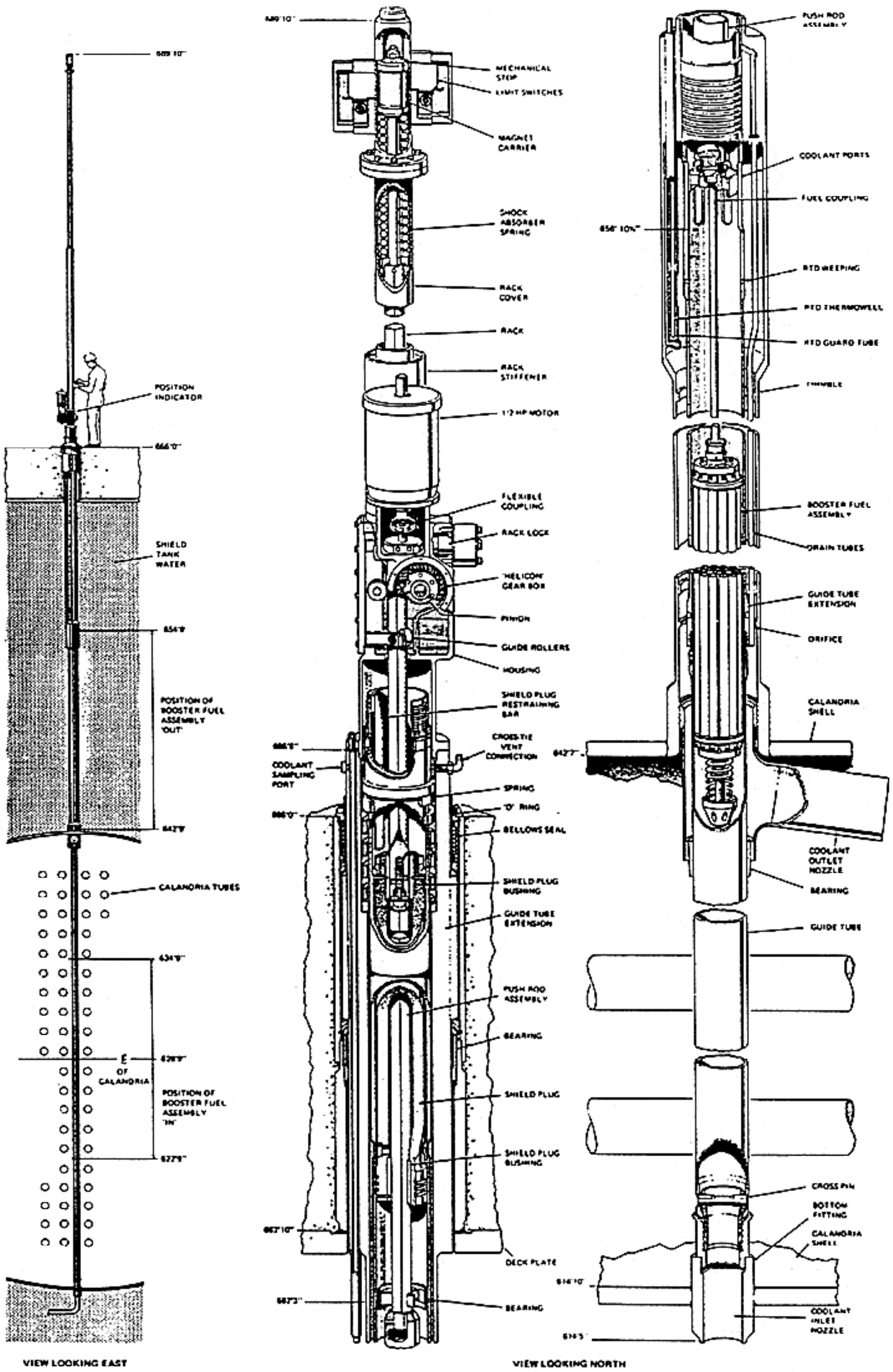


Figure 30:
Zone Control Unit

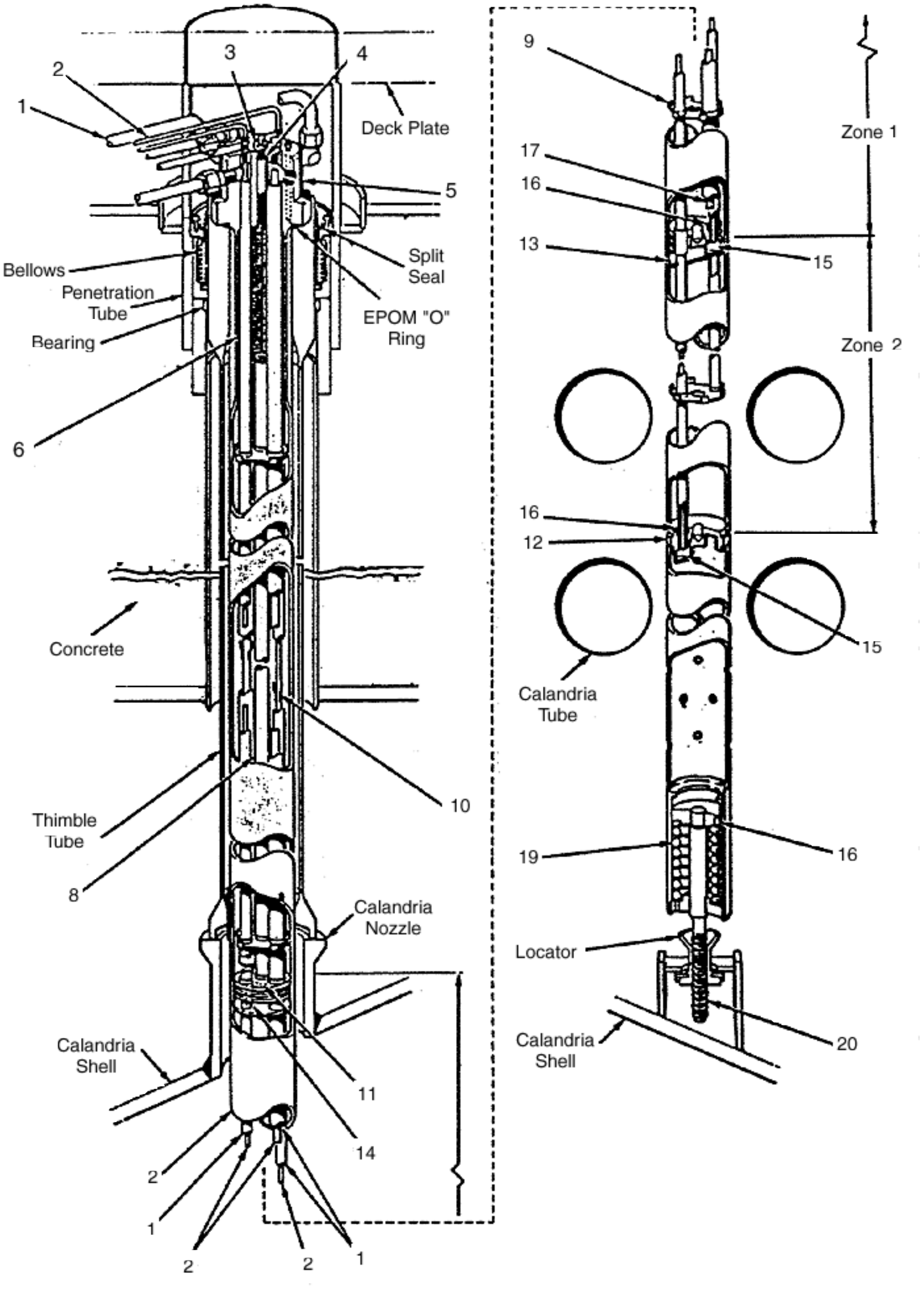
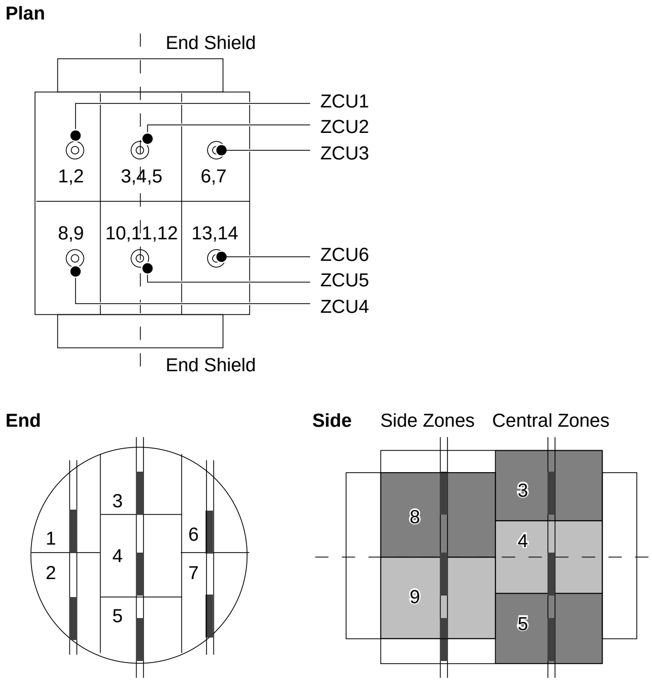


Figure 31:
14-Zone Core for CANDU 6



2.19 Mechanical Control Absorber Units (MCA)

The purpose of the control absorber rods is to provide approximately 8.2 mk of additional negative reactivity controlled by either the RRS or stepback system. This additional negative reactivity would be required in certain process system failures such as a loss of regulation.

These are the final control elements of the RRS installed in the reactor.

A mechanical control absorber consists of a hollow, open-ended tube of cadmium, sheathed in stainless steel, attached to a winch drive by a cable and installed in a guide tube. The drive mechanism seals off the thimble. The absorber tube has a support rod up the middle to which the drive cable is attached. An orifice is installed in the tube to slow down its entry into the core during a free fall (Figure 32).

To explain the last statement, the rod can either be lowered into the core by the winch or it can be dropped by releasing the sheave from the motor. For this purpose the drive is equipped with an electromagnetic, friction clutch between the sheave and the motor which can be de-energized from the control room.

Mechanical control absorbers are inserted into the reactor core, by the RRS, only on those occasions when a greater rate or depth of negative reactivity, than can be provided by the zone control units, is required. For example, to reduce the power level quickly the absorber rods can be inserted either by the motor drive or by releasing the clutch.

2.20 Mechanical Shutoff Units (SOR)

The purpose of the SORs is to insert a stated reactivity into the core in a stated time.

Mechanical shutoff units are the in-core operating parts of SDS 1.

They are virtually identical in design and operation to the control absorber units with several differences.

In the ready position the rod has compressed a spring which will accelerate it in the first part of its descent into the reactor. There is also no orifice plate to slow its drop.

The vertical position of each rod is measured by an electrical potentiometer sensor on the sheave shaft. In the CANDU 6 a second position sensor, the "rod ready" indicator, monitors the presence of the rod in the withdrawn position, to verify that it is ready to use. Magnetic reed switches mounted in the shield plug sense the presence of a permanent magnet mounted on the top of the rod.

The rods are dropped into the core when a shutdown trip signal de-energizes their clutches. They cannot be withdrawn by the regulating system until the trip is cleared.

2.21 Liquid Injection Shutdown Units

A schematic of the Liquid Injection Shutdown system is shown in Figure 33. The in-core actuation parts of SDS 2 are the liquid injection shutdown units.
They consist of zircaloy injection, nozzle tubes which are inserted, horizontally, through thimbles and are supported on the other end by a threaded locator on the opposite wall of the calandria. Each tube is perforated by rows of holes oriented to give the optimum dispersal of the
liquid poison. Stainless steel injection tubes, which are inside the thimbles, connect the inlet of the nozzles to the rest of the liquid injection system. Figure 34 shows an injection nozzle.

A schematic diagram of the liquid injection system is shown in Figure 33. Each nozzle tube is connected to a separate container of gadolinium nitrate solution. These containers are in turn connected to a common pressurized helium tank and are isolated from it by a set of quick opening valves. On receipt of a trip signal the valves open and the pressurized helium forces the neutron absorbing poison, gadolinium nitrate, into the moderator, shutting down the reactor.

The poison is removed from the moderator at the conclusion of the shutdown by the moderator purification system under the control of the control room operator.

Figure 32:
Shutoff and Control Absorber Unit

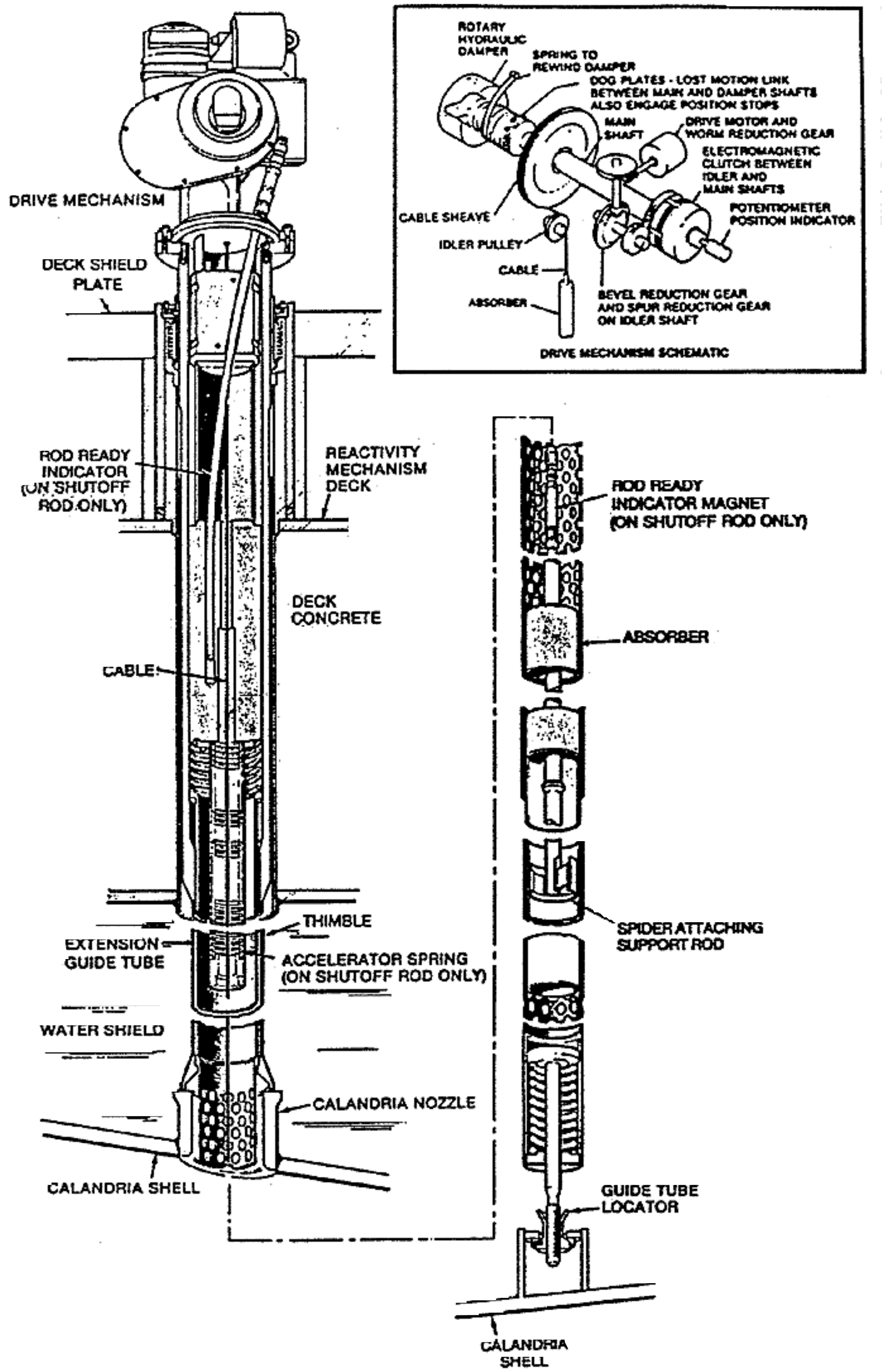


Figure 33:
Schematic of Liquid Injection Shutdown System

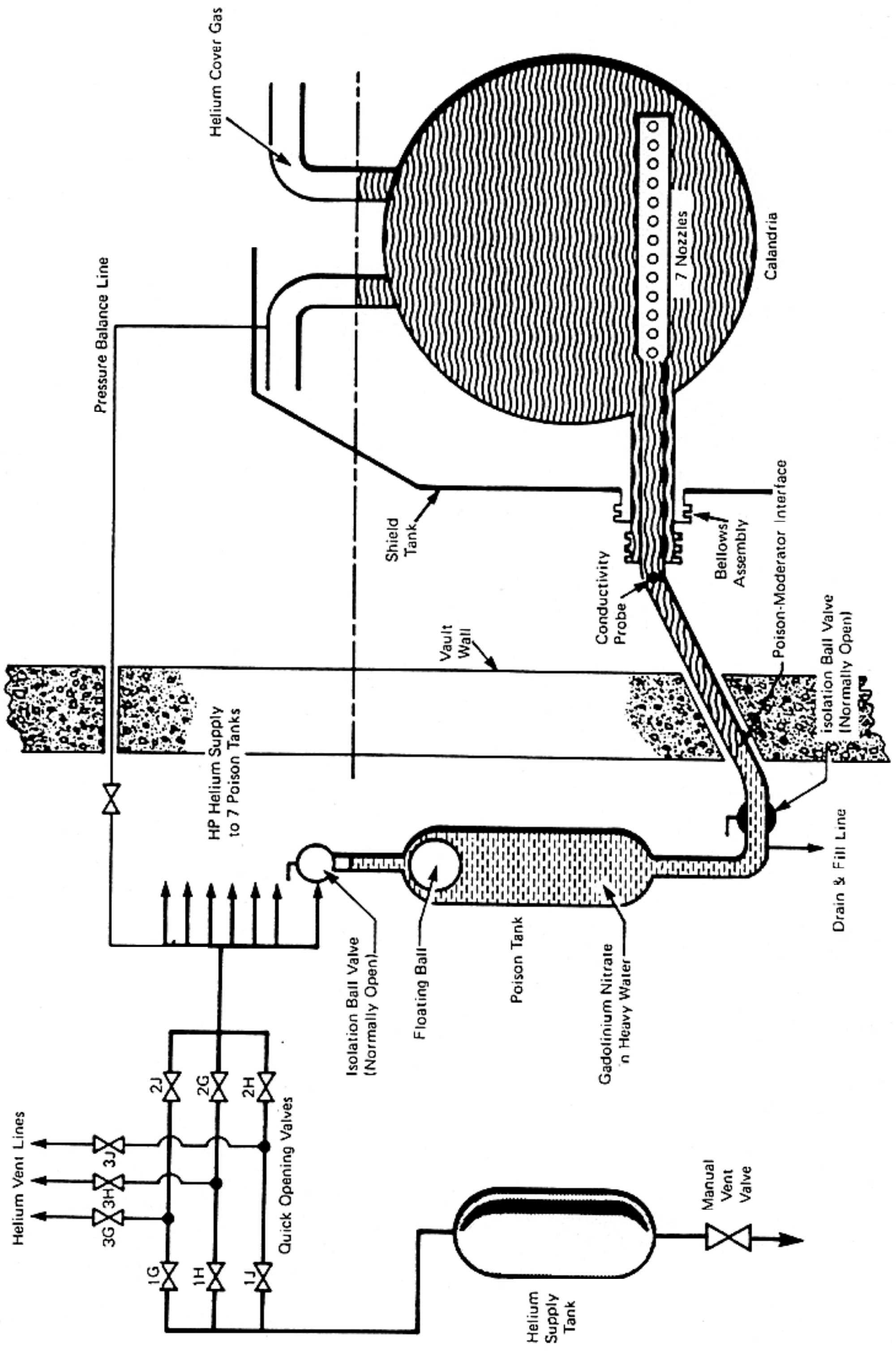
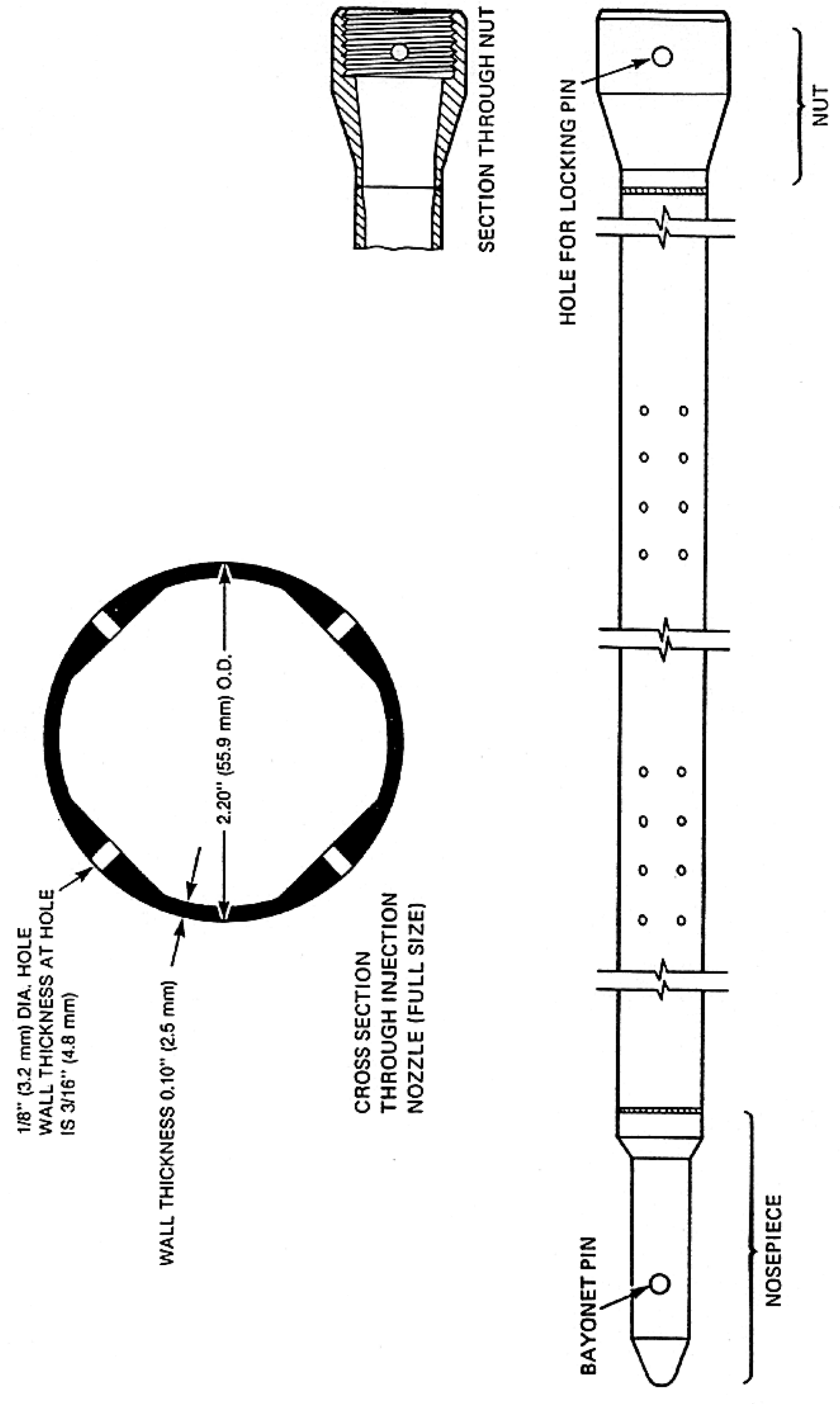


Figure 34:
Injection Nozzle



2.22 Summary of Reactivity Control Units

Table 1 gives a summary of the number of reactivity control units in the reactor regulating system and the two shutdown systems.

Table 1.

Summary of Reactivity Control Units

Reactivity Control Units	**CANDU 6**			**Bruce 'A'**		
	RRS	**SDS 1**	**SDS 2**	**RRS**	**SDS 1**	**SDS 2**
Vert.Flux Detectors	26	26		12	12	
Horiz. Flux Detectors			7			2
Vert. Ion Chambers				3	3	
Horiz. Ion Chambers	3	3	3			3
Adjusters	21					
Boosters				16		
Liquid Zone Control	6			6		
Control Absorbers	4			4		
Shutoff Rods		28			30	
Liquid Injection Units			6			7

NOTE:
Bruce 'A' has 12 VFDs, of the design discussed, in units 1 and 2 and 17 of a different design in units 3 and 4.

2.23 Summary of Reactor Assembly

The major structural features of the reactor assembly have been described: the calandria with its end shields and calandria tubes, which hold the fuel channels; The water-filled vault or shield tank; the penetrations into the core for the reactivity control units and the piping for other systems; the provisions for attenuating the radiation to allow work to be done.

2.24 Other Systems which Interface with the Reactor Assembly

Here is a list of the major systems that interface with the reactor assembly. They are covered in other lessons.

Fuel Handling System
Heat Transport System
Moderator System
Reactor Regulating System
Safety Systems
    Shutdown Systems
    Emergency Core Cooling System
Shield Cooling System
Cover Gas System
Annulus Gas System

2.25 Radiation Hazards to Staff

It is difficult to isolate the radiation hazards from the reactor assembly as opposed to the major interconnected systems. At power, the principal source of external radiation is the prompt gamma rays resulting from neutron capture in the calandria and tubesheets. At shutdown, the major radiation sources are gamma radiation from activation products, primarily cobalt-60, in the calandria and tubesheets, fission product decay gammas from the fuel and the corrosion and fission product deposits in the end fittings and feeder pipes. To reduce the external hazard there are limits on the cobalt content of all components subjected to neutron flux, for example the calandria shells, annular plates, tubesheets and reactivity mechanisms.

The principal source of internal radiation hazard at both power and shutdown is tritium. If tritium removal is not being carried out the equilibrium concentration of tritium in the moderator system is at least a factor of ten greater than the heat transport system. High concentrations can be encountered when these systems are opened for maintenance or when a spill of D_2O occurs.

The calandria is encased in a water-filled vault or shield tank the purpose of which is to reduce the external radiation dose in the areas outside these barriers to acceptable levels at both power or shutdown. However some areas such as outside the end shields and the fuelling machine vaults, are only accessible 24 hours after shutdown. Certain spaces such as the fuelling machine vaults are equipped with dedicated drier systems to reduce the tritium in air concentrations.

Main Moderator System

Objectives

On completion of this lesson the participant will be able to:

- List the functions and design features which the moderator system must provide;
- Sketch the moderator system including the pumps, the heat exchangers, the head tank, the purification system and the cover gas system;
- State the materials used for the various parts of the moderator system and why they are used;
- Describe the set up of the pumps and motors and the heat exchangers;
- Explain the control of temperature and level in the moderator system;
- Describe how the system and its components are protected against overpressure;
- List the sequence of automatic actions taken by the moderator system as a result of abnormal events;
- Describe the functions of the auxiliary systems;
- Describe the methods of protecting the plant personnel and the public from the radiation hazards originating with the moderator system;
- Compare the CANDU 6 moderator system with other systems.

Table of Contents

1 System Overview

The discussion which follows deals with only the CANDU 6 system. In one section there is a discussion of the major differences between it and the other stations.

The heavy water in the main moderator system performs the following functions:
- moderates (slows down) the high energy fission neutrons to the required thermal energy levels to permit further nuclear fission;
- removes the heat generated by the moderating process;
- serves as a medium for dispersion of chemicals to control reactivity in the reactor core;
- provides a heat sink for the reactor fuel in the unlikely event of a Loss of Coolant Accident (LOCA) coincident with the unavailability of emergency core cooling.

Heat enters the heavy water moderator by the action of slowing down of fast neutrons - the moderating action. This source of heat is only present during operation. Other sources of heat are:
- the conduction from the fuel channels which is present as long as the channels are at operating temperature;
- by gamma heating from the fission process which is greatly reduced when shutdown;
- by gamma heating from the decay of fission products which continues after shutdown.

The system is a low pressure, low temperature closed D_20 circuit and consists of two 100 percent capacity centrifugal pumps, two 50 per cent flow capacity heat exchangers cooled by service water, and valves, piping, instrumentation and associated control mechanisms. During normal operation of the reactor, heavy water is continually pumped from the bottom of the calandria and returned through the heat exchangers which remove the moderator heat. The cooled water enters the calandria through diametrically opposite nozzles located approximately at the horizontal midplane of the calandria. Discharge from the nozzles generates forced convection currents in the moderator inside the calandria. The level of moderator in the calandria is kept constant during normal operating conditions; minor temperature fluctuations in the moderator system are accommodated in a head tank above the calandria.

2. Summary of Functional and Design Requirements

2.1 Safety Requirements

The moderator system is required as a backup heat sink following;

- a LOCA
- a LOCA with loss of class IV power
- and a LOCA with loss of emergency core cooling.

To perform this function the system is designed to provide adequate;

- circulation
- pump NPSH
- subcooling and
- heat removal capability.

Provision is made to rapidly cool the moderator under LOCA conditions by increasing the flow of cooling water from the moderator heat exchangers.

2.2 Process Requirements

The heavy water in the moderator system provides a medium to slow down the high energy fission neutrons in the reactor core to the thermal energy levels required to promote further nuclear fission.

The system is designed to remove the heat generated in or transferred to the moderator and to control the bulk temperature in the calandria. The sources of heat are;

- a result of the fission process either from thermalizing neutrons or absorbing fission gammas,
- heat generated from the absorption of gamma rays from fission products and activated components of the core,
- heat transferred from the pressure tubes to the calandria tubes across the annular gap.

The heavy water in the moderator system will accumulate activation and corrosion products and in addition it is used as a medium for the dispersion of liquid poisons injected to control and shutdown the reactor. To restore and maintain its purity, the system design provides a flow of heavy water through a purification system.

Radiolysis of the heavy water produces deuterium and oxygen. To deal with this hazard the cover gas system is designed to include recombination units which limit the concentrations.

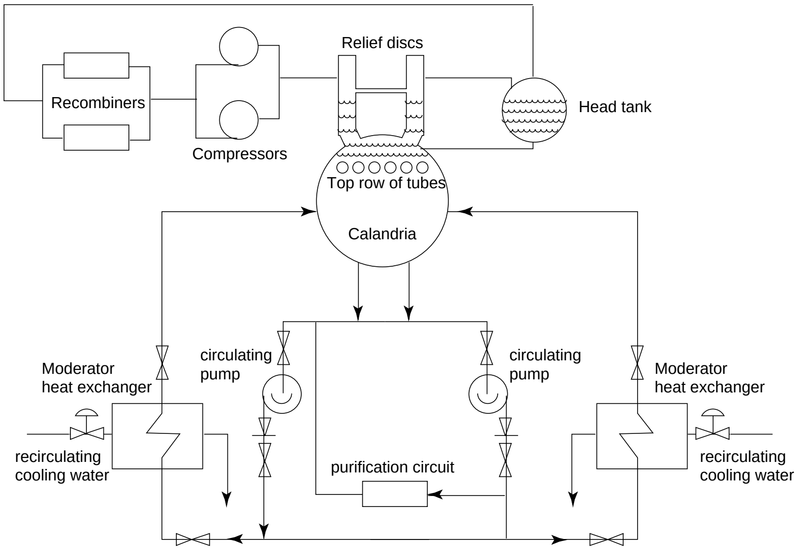
To limit the pressure on the lower calandria tubes during an upset the system is designed to control both the moderator level and the cover gas pressure.

3. System Description

Referring to figure 1, it can be seen that the main system has two circulating pumps and two heat exchangers. The heavy water is drawn through two pipes in the bottom of the calandria into a common header. From there it can be pumped by either one of the pumps into a common discharge header and then through the two heat exchangers and back into the calandria at about the midplane on diametrically opposite sides. The discharge nozzles are designed to prevent the impingement of cold heavy water directly onto the calandria tubes.

There is a head tank located above the calandria which maintains the moderator level nearly constant by accommodating the swell and shrink of the heavy water caused by minor temperature fluctuations.

Figure 1:
CANDU 6 Main Moderator System



Each pump is designed for 100% of design flow capacity and each heat exchanger for 50% of design heat removal capacity. This setup permits the flow from either pump to be divided equally between the two heat exchangers.

The main moderator system is connected to the moderator purification system, the cover gas system and the D_2O collection,supply and sampling systems.

The pumps, heat exchangers and valves are located within a space, with a D_2O vapour barrier, to the side of the calandria vault, (Figure 2). The pump suction and heat exchanger discharge lines are anchored to rigid penetration seals where they pass through vault concrete. This is to maintain the integrity of the vault and to eliminate loss of vault shielding water. All of the equipment in the system is accessible for isolation and maintenance when the reactor is shut down.

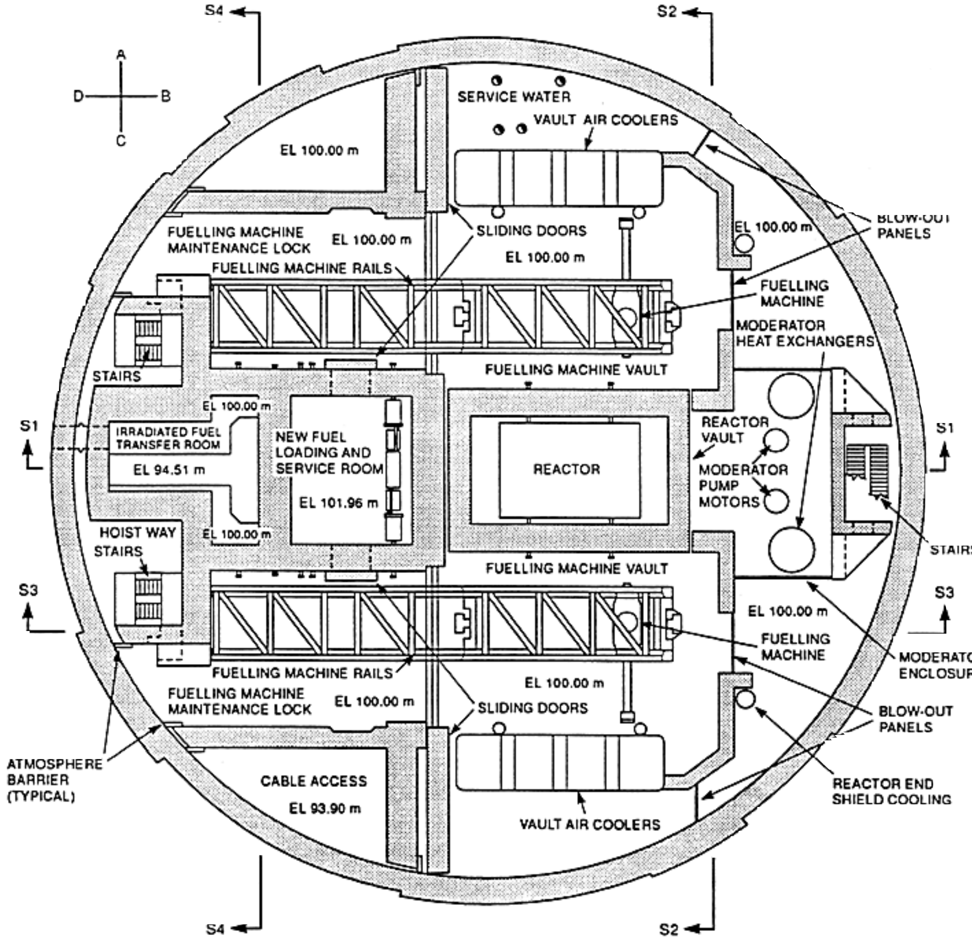
With the exception of the heat exchanger tubes and tube sheets, all components of the moderator system, outside the calandria, which are in contact with heavy water, including the piping, are made of austenitic stainless steel. The tubes are made of Incoloy 800 and the tube sheets are made of carbon steel with a stainless steel overlay on the shell side and an Inconel overlay on the other side.

The moderator system components are designed and built to higher standards than required in order to minimize the possibility of heavy water loss and to maximize reliability. For example, liveloaded, doublepacked stem valves are used on large valves and bellows stem seals are used on small valves to reduce maintenance and leakage.

The calandria is seismically qualified to earthquake level DBE category ‘A’ because it provides support to the heat transport system pressure boundary, the integrity of which must be maintained to prevent a LOCA during an earthquake.

The main moderator system is environmentally qualified to remain functional in the harsh environmental conditions caused by a LOCA plus a loss of emergency core cooling.

Figure 2:
Plan of a typical Reactor Building
Showing Location of Moderator Equipment



4. Major Equipment and Components

4.1 Pumps

As mentioned in the previous section the main moderator system is furnished with two pumps each capable of handling 100% of the design flow capacity. Each pump is provided with a main motor and a pony motor. They are vertically mounted, single stage, double suction centrifugal pumps with a rated capacity of 940 *l*/s at 600kPa(g) and a head of 55 m at 1190 rpm. The pumps are duplicated for reliability purposes so that during normal operation one pump is running and the other is on standby. If the operating pump motor fails the standby unit is started automatically. Failure of both main motors automatically starts the pony motors.

The moderator pump shafts are sealed with two multiple mechanical seals installed in series. Each of these seals is capable of sealing the pump gland should one of them fail. They are cooled by heavy water from the pump discharge. There is also a backup seal to prevent discharge of heavy water to the atmosphere should both of the mechanical seals fail.

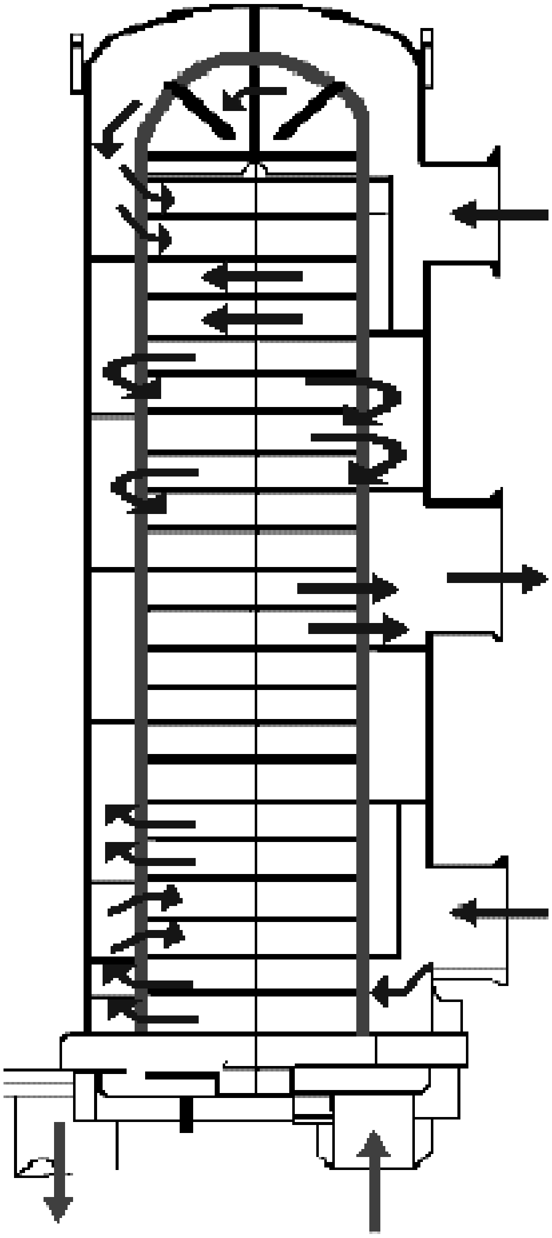
The main motors are directly coupled to the pump shaft so that they turn at the same speed and they are rated at 750 kw, 1190 rpm and 4.16 kV. They are three phase induction motors, totally enclosed and cooled by a water cooled heat exchanger. To prevent condensation of moisture inside the enclosure when they are not in operation the motors are provided with heaters. The motors are supplied by Class IV power.

The pony motors are three phase induction motors, fan cooled and totally enclosed and are rated at 15 kw, 245 rpm and 600 v. They are coupled to the main motor shaft and drive the pump through an integral speed reducer at 25% of the main motor speed. This provides a flow of 235 *l*/s at 34 m head sufficient to remove the decay heat during reactor shut down. The motors are supplied by Class III power.

4.2 Heat Exchangers

The moderator heat exchangers are U tube-in-shell design. The moderator D_2O, which is carried inside the tubes, makes two passes through the exchanger.The flow of the recirculated cooling water, on the shell side, makes four passes and is split. The split flow design for the shell side was chosen to minimize the pressure drop. Each heat exchanger is designed to remove 50% of the moderator system heat load which is 50 MW(th).

Figure 3:
Diagram of Flow Patterns in the Moderator Heat Exchanger



The temperature of the D_2O at the outlet from the calandria is controlled at 69 °C by controlling the flow of the cooling water through the heat exchanger. During normal operation, the heat exchangers cool the heavy water from 69°C to 46°C. The signal to the temperature control valves is provided from the calandria outlet D_2O temperature sensors by the station control computer.

To minimize the possibility of downgrading the moderator, the cooling water is maintained at a lower pressure than the heavy water. In addition, the possibility of heat exchanger tube leaks is minimized by using quality control methods in the manufacture of the tubes and the assembly of the tube to tubesheet joints and by strictly controlling the tube vibrations induced by the cooling water flow.

4.3 Head Tank

The moderator head tank accommodates the fluctuations in heavy water volume due to bulk temperature changes. During normal operation, moderator temperature is expected to be controlled at a fixed setpoint ±3 °C which will result in a volume change of ±0.57 m^3. The heavy water is free to flow between the calandria and the head tank through a connecting line. The head tank limits the level in the calandria to a maximum of 0.61 m above the top inside diameter of the calandria vessel.

5. Control, Monitoring and Diagnostics

5.1 Moderator Temperature

Temperatures measurements are required to;

- control the moderator temperature at the calandria outlet, to provide indication and high and low temperature alarms in the control room, and to initiate a reactor power setback on a very high temperature,
- provide calandria outlet temperature indication to the secondary control room,
- provide indication of the outlet temperature of the moderator heat exchangers and to provide high and low temperature alarms in the control room.

A control program in the station computers maintains a constant D_2O temperature at the calandria outlet.

The prime purpose of this control scheme is to maintain a constant calandria temperature using calandria outlet temperature as the controlled variable. The scheme has the following features;

- the response to the process transients
- calandria level control is not required,
- temperature cycling of all components is reduced
- calandria temperature is maintained near the operating level during limited shutdowns. This reduces the need to transfer inventory to accommodate the shrink and swell associated with large temperature changes. Maintaining a

relatively constant temperature also reduces the need to accommodate the reactivity swings associated with large changes in the moderator temperature.

The temperature control is accomplished by modulating the temperature control valves in the recirculated cooling water lines to the heat exchangers, thus varying the cooling water flow. A large and a small control valve are provided in each recirculated cooling water line. During full power operation the small valves are fully open and the large valves are used for control. At low power the large valves are closed and the small valves are used for control. The feedback signal is the median signal from three temperature sensors located in the calandria outlet piping. High and low temperatures are annunciated in the control room. High moderator temperature setback 79°C and trip on SDS1 87°C is provided to protect against loss of recirculated cooling water flows.

A duplicated control program in the standby computer takes over control in the event of failure of the operating computer. On failure of instrument air or control power, the recirculating cooling water valves fail open.

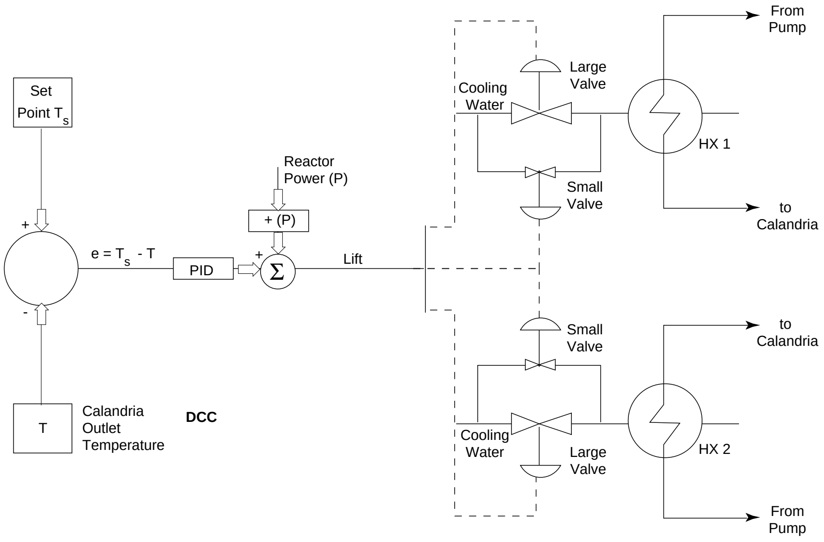
Temperature sensors located at the outlet of each heat exchanger and the pump suction header provide control room indication and alarm on high temperature.

The alarm setpoint is a function of the reactor power. Since the temperature measurement is used for monitoring only, it is not duplicated.

Figure 4 shows the control scheme for calandria outlet temperature control. During normal operation, the moderator temperature at the calandria outlet is controlled to a constant setpoint. This setpoint may be changed by operator command via the keyboard within upper and lower limits.

Fig 4

Moderator Outlet Temperature Control



A proportional-integral-derivative (PID) controller with a feed forward term is provided to control the moderator temperature at its setpoint. The feed forward term is a function of the reactor power. The integral valve lift component is limited to provide anti-reset windup.

During the first program pass and upon transfer from the main to standby computers, the integral term is initialized to zero.

High moderator temperature may result in:
- an increase in core reactivity due to the positive moderator temperature coefficient and localized boiling,
- an increase in thermal stresses between the calandria and end shields leading to component damage,
- an increase in deuterium levels leading to an explosion hazard in the cover gas system,
- the unavailability of the moderator system to act as a heat sink under certain accident conditions.

Under loss of coolant (LOCA) conditions, moderator crash cooldown is initiated by lowering the setpoint. This is initiated by low heat transport pressure coincident with either high reactor building pressure (signifying an out of core break), or by high moderator level (an in-core break).

In the event of low recirculated cooling water (RCW) pressure such as following a loss of Class IV ac power, RCW load shedding is initiated. For the moderator this would shut the large valves and partially close the small valves. If a LOCA is coincident with a loss of Class IV, the large valves are closed, even though crash cooling is initiated.

A reactor power setback at 1% FP/s to 25% of full power, then 4% of current power to a final value of 2% of full power is initiated by a 2 out of 3 measurement when the temperature exceeds 79°C. Full power may be restored when the temperature control of circulation problems are eliminated.

Other Temperature Measurements

Calandria Outlet Temperature Measurement (in Secondary Control Area)

A temperature detector in the pump suction header measures the calandria outlet temperature. Its ranges is 0 to 100°C, and the temperature is displayed on a panel in the secondary control area. It is only valid if there is D_20 circulation in the system.

Moderator Heat Exchanger Outlet Temperature Measurements

Alarms are provided on either low or high temperature. There is a temperature detector mounted downstream from each of the moderator heat exchangers. The measurement range is 0 to 100°C.

5.2 Moderator Level

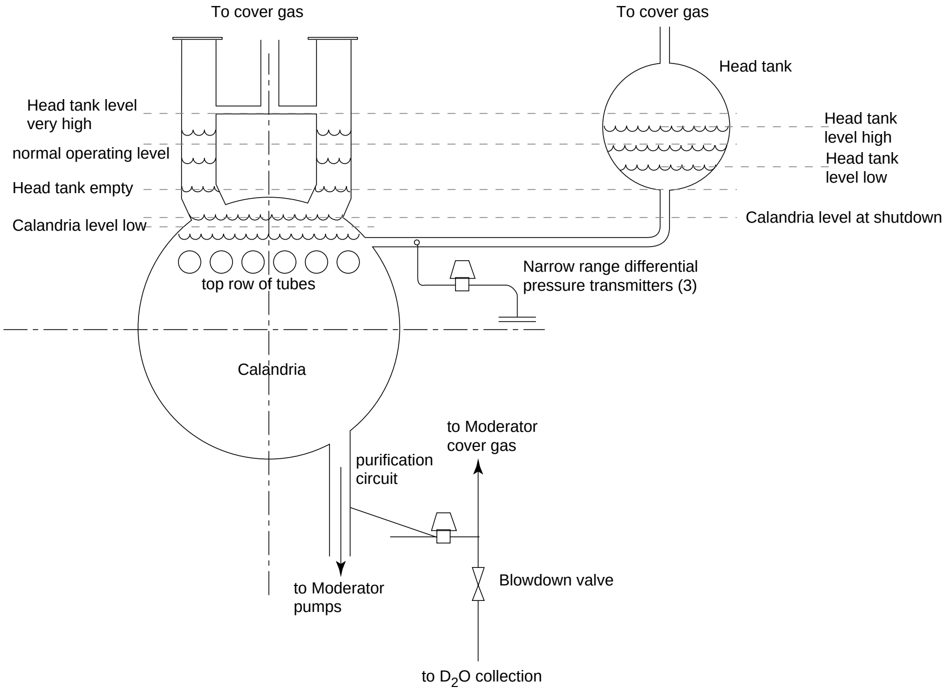
To simplify the instrumentation and controls, a head tank is provided to accommodate the $D_2$0 swell due to temperature fluctuations. During normal operation, the moderator level will be 30 cm. above the top inside diameter of the calandria when the moderator bulk temperature at the calandria outlet is 69°C.

The calandria is full under all modes of operation except during prolonged shutdown for maintenance purposes.

The maximum normal operating level is 61 cm. above the top inside diameter of the calandria. A level above 61 cm. reduces the gas space in the calandria pressure relief ducts and affects the peak calandria pressure following a burst pressure tube incident.

Fig 5

The relative position of the head tank and calandria.



Calandria Level Measurement - Narrow Range

Three differential pressure transmitters are used to measure the level in the top of the calandria and in the head tank. The range of each measurement is 6850 to 8600 mm. which covers the following:

• Top outside diameter of top row of calandria tubes	6864 mm.
• Top inside diameter of calandria	7595 mm.
• Centreline of expansion tank	8084 mm.

The normal level for the calandria at 69°C is 305 mm. above the top inside diameter of the calandria. This is at 7900 mm.

The output of each transmitter is used in a 2 out of 3 arrangement to provide annunciation on an alarm window of high or low moderator level.
The transmitters are also connected to the station computers for CRT display of the median of the three measurements and for very low alarm. The alarm setpoint is 7630 mm. The level measurements are used by the Moderator Temperature Control Program in the initiation of moderator crash cooling, during an in core LOCA.

The signal for the level measurement is taken from the line connected the calandria to the head tank. The reference pot is maintained full by a constant flow of water to it.

Calandria Level Measurement - Wide Range

Wide range level measurement includes both the calandria and the head tank. The range of the level measurement is 0 to 8600 mm. which is from the bottom of the calandria to 59.7 mm. above the top inside diameter of the head tank.

The level measurement uses a wet reference leg which is maintained full by a flow from the system with the overflow going to the head tank.

The output of the transmitter is connected to the station computers for CRT display. The transmitter output will not give an accurate reading of the calandria level when either one of the pumps is running due to the pressure drop in the pump suction lines.

Low moderator level may lead to:
- overheating of the calandria shell and upper calandria tubes,
- reduction in reactivity in the upper section of the core,
- increase in deuterium concentration due to an increase in surface area.
- S.T.B. (Stepback)

High moderator level may lead to:
- inadequate volume of gas space in the relief ducts to accommodate the poison following operation of SDS2,
- flooding of the SDS2 injection header, which will move the D_20/poison interface away from the calandria leading to a delay in poison injection should SDS2 operate,
- water hammer in the helium injection header should SDS2 operate.

5.3 Differential across Moderator Pumps

The differential pressure across the moderator pumps is measured to provide a display in the control room and alarms on low differential pressure. These triplicated differential pressure measurements are also used to initiate a reactor power setback on very low differential pressure, which happens when the running pump trips and the standby pump fails to start causing a loss of moderator heat sink. The setback is cleared when the circulation is re-established.

5.4 Instrumentation and Control for Moderator Pump and Motor

Each of the main and pony motors is controlled by a handswitch on a panel in the main control room. During normal operation one main pump is ON, one main pump is on STANDBY and two pony motors are on STANDBY. During shutdown the two main pumps are OFF, one pony motor is ON and the other pony motor on STANDBY. The logic is designed such that the pony motor cannot be started while the main motor is running and vice versa. If the main motor stops for any reason the differential pressure across the pump will fall and the STANDBY main motor will automatically start. Logic will prevent the failed main motor from re-starting.

On loss of Class IV power, the running main pump motors will stop and when Class III power is available both pony motors will start automatically on receipt of very low differential pressure signal across the pumps.

Instrumentation is provided to monitor the proper functioning of the pump and motor. These consist of pump seals pressure measurements, motor and pump bearing vibration monitoring and stator windings and bearings temperature monitoring. Alarms are provided when the monitored parameters exceed the setpoint values. Corrective maintenance or other actions can be initiated based on these alarms.

6. Protection Against Overpressure

6.1 System

Two sets of relief devices are provided primarily for the overpressure protection of the calandria: the bleed valves in the cover gas system and the rupture discs at the end of each of the four calandria pressure relief pipes. These devices also protect the moderator system against overpressure that may be caused by the calandria pressure transients. The bleed valves start to relieve at a gauge pressure of 28 kPa(g) and the rupture discs are rated at 138 kPa(g).

6.2 Moderator Pumps

Under normal operating conditions, the moderator pumps are protected against overpressure by the bleed valves in the cover gas system as the pump discharge is connected to the calandria through the moderator discharge header.

During maintenance on one of the pumps, the pump isolating valves are closed. The downstream isolation valve and the pump drain valve are interlocked so that both valves cannot be closed simultaneously, to prevent over-pressurization of the pump.

6.3 Moderator Heat Exchangers

Under normal operating conditions, the tube side is protected against overpressure by the bleed valves in the cover gas system since the tube side of the heat exchanger outlet is connected to the moderator discharge header.

During maintenance on one of the heat exchangers, the heat exchanger isolating valves are closed. The downstream isolation valve and the heat exchanger drain valves on the tube side are interlocked so that both valves cannot be closed simultaneously, to prevent over-pressurization of the tube side. The shell side is protected against overpressure by relief valves on the discharge header of the recirculated cooling water system.

6.4 Moderator Head Tank

The head tank is connected to the cover gas system and the calandria pressure relief ducts through the helium balance lines in the cover gas system. Overpressure protection of the head tank is provided primarily by the bleed valves in the cover gas system and, as a backup, by the rupture discs on the calandria relief ducts.

7. System Operation

7.1 Normal Operation

Main moderator system normal operation is defined as that operating condition when the reactor is at full power equilibrium fuel conditions. The moderator heavy water is drawn from the bottom of the calandria at the rate of 940 *l*/s by one of the two moderator pumps, while the other is on standby. The discharge flow from the pump is split approximately equally between the two heat exchangers to reject 100 MW(th) of the moderator heat load. The moderator bulk temperature at the calandria outlet is controlled at 69°C by regulating the recirculated cooling water flow to the heat exchangers. Normally, the RCW flow rate through each heat exchanger is up to 1244 *l*/s.

The moderator level in the calandria, during reactor normal operation, is about 0.30 m above the top inside diameter of the calandria with the normal maximum level set at 0.61 m above the top inside diameter.

7.2 Shutdown

Shutdown State with Shutdown Cooling as Heat Sink

In a long term shutdown, the shutdown cooling system can act as the heat sink for the moderator system to remove decay heat produced in the calandria.

In this state, the following conditions apply:
a. The HT system is running on shutdown cooling with bulk HT temperature of < 55°C.
b. All four moderator pump motors are selected OFF.

Guaranteed Shutdown State
For the reactor shutdown guarantee the moderator is either over-poisoned or drained below cold critical level. The purification system must be completely isolated. These states will be used during maintenance of reactivity mechanisms or long shutdowns.

7.3 Operation Under Abnormal Operating Conditions

The abnormal operating conditions include moderator system `upset' operating conditions and emergency conditions.

Total Loss of Class IV Power
Following a total loss of Class IV power, a reactor shutdown occurs and flow from the moderator pumps stops due to a loss of power to the main motors. Both pony motors start automatically on Class III power, 15 seconds after the initiation of the pump very low differential pressure alarm.

Recirculated cooling water (RCW) flow through the moderator heat exchangers also stops due to the loss of Class IV power. The RCW flow is restarted within three minutes of the loss of Class IV power as Class III power is re-established. Since it is possible that only one RCW pump will restart, RCW load shedding is initiated. This limits the total RCW flow available to each of the moderator heat exchangers to 140 *l*/s.

This time delay and increase in D_2O temperature will cause an increase in calandria end shield and tubesheet temperatures, however within 100 seconds after the reactor trip, nuclear heating of the end shield and tubesheets would drop to about 7% of full power heating value and the stored heat from the tubesheets would start to flow out into the end shield cooling water. Even though Class IV power failure causes the end shield system pumps to fail until Class III power is re-established, the increase in the calandria tubesheet temperature (and resultant thermal stresses) is within acceptable limits.

Loss of Moderator Pump Main Motors
In the event of failure of the operating pump main motor and unavailability of the standby pump's main motor, the operating motor will run down. Since the reactor is at full power, the heat load of 100 MW(th) will increase the temperature of the moderator D_2O rapidly. Power is available to the pony motors throughout this transient and they will be switched on, automatically,

15 seconds after the main motor runs down. When the measured calandria outlet temperature exceeds the setback temperature setpoint of 79˚C, the reactor is automatically set back at a rate of 1% full power/s to 25% of full power and then at 4% of current power to a final value of 2% of full power. If the temperature increases to the reactor trip set point of 87°C, a reactor trip will occur via SDS1.

Loss of Recirculated Cooling Water to Moderator Heat Exchangers
If recirculated cooling water is lost to one or both heat exchangers, then the temperature will increase in the moderator. The reactor will remain at full power until the setback temperature of 79˚C is reached. If the recirculated cooling water is completely lost to both heat exchangers, then the temperature will increase to the reactor trip setpoint of 87˚C and the reactor will be tripped.

The reactor trip prevents significant release of heavy water through bursting of the rupture discs. Fuel cooling is ensured by the heat transport system which is still intact. The Heat Transport System will be cooled down to < 55°C.

Loss of Coolant Accident (LOCA)
Following a loss of coolant accident, a reactor shutdown will occur. The emergency core cooling system will act as the reactor heat sink and the heat load on the moderator system .is reduced by the reactor shutdown.

Upon receipt of the LOCA signal, the moderator maximum cooling is initiated by the moderator temperature control program. The temperature control valves on the recirculated cooling water lines are opened to the mechanical travel stops in order to cool the moderator as rapidly as possible. The maximum cooling water flow through the temperature control valves to each heat exchanger is the normal flow rate of 1244 *l*/s. The flow rate is limited to prevent damage to the heat exchangers caused by flow induced vibration on the shell side. However, the flow rate is adequate to remove the anticipated heat loads during this transient.

LOCA Coincident with Loss of Emergency Core Cooling (LOECC)
If the reactor loses its primary coolant (due to a heat transport system pipe break) coincident with the unavailability of the emergency core cooling system, the moderator system acts as the reactor heat sink. The anticipated heat loads during this transient will be removed by the moderator heat exchangers.

Upon receipt of the LOCA signal, the moderator maximum cooling is initiated as described above

LOCA Coincident with Loss of Class IV Power
Following a loss of coolant accident coincident with loss of Class IV power, a reactor shutdown occurs, the ECC is activated and the flow from the moderator pumps stops due to a loss of power to the main motors. Both pony motors start automatically when Class III power is available after the main motor runs down.

Recirculated cooling water flow through the heat exchangers also stops due to the loss of Class IV power. The cooling water flow is restarted within three minutes after power is re-established from the Class III diesel generators. Upon receipt of the LOCA + loss of Class IV power signal, the moderator temperature control program will limit the opening of the temperature control valves such that the RCW flow through each heat exchanger is limited to 442 *l*/s. The flow is limited due to the RCW load shedding requirements.

Out-of-Core LOCA Coincident with Loss of Emergency Core Cooling and Loss of Class IV Power

In the event of an out-of-core loss of coolant accident, coincident with the unavailability of emergency core cooling system and loss of Class IV power, the moderator system acts as the reactor heat sink. The flow from the moderator pumps stops due to a loss of power to the main motors. Both pony motors start automatically on Class III power, when available, after the main motor runs down.

Recirculated cooling water flow through the heat exchangers also stops due to the loss of Class IV power. The RCW water flow is restarted within three minutes after power is re-established from the Class III diesel generators.

Upon receipt of the LOCA + loss of Class IV power signal, the moderator temperature control program will limit the opening of the temperature control valves such that the RCW flow through each heat exchanger is limited to 442 *l*/s. The flow is limited due to the RCW load shedding requirements.

Transients Resulting in Burst Rupture Discs

In the event that the rupture discs burst, due to an in-core LOCA for example, the liquid effluent released from the rupture discs will be collected by the reactor building active drainage system and then transferred to the service building active drainage system receiver. Depending on the isotopic content, the effluent will be transferred from the service building active drainage system receiver to either the D_2O cleanup system or to the liquid waste management system for further treatment. The effluent transferred to the D_2O cleanup system will be upgraded as necessary and returned to the main moderator system. The effluent transferred to the liquid waste management system will be treated, diluted and then released from the plant through controlled release.

The gaseous effluent released from the rupture discs will be collected by the D_2O vapour recovery system and the reactor building ventilation system. The ventilation system will pass the gaseous effluent through charcoal and HEPA filters prior to controlled release to the atmosphere.

Deuterium Excursion

Deuterium excursions can be a result of high moderator temperature, low moderator level, increased radiolysis of water or organic impurities, recombiner failure or low oxygen level.

The upper explosive limit of 8% must be avoided. Operator actions are required to ensure the D_2 concentration never exceeds the operational limit of 6%.

8. Interdependencies with other Systems

The main moderator system interfaces with a number of auxiliary systems, each of which provide supporting functions to the system for stable and continuous operation.

Moderator Cover Gas System

Helium is used as the cover gas for the moderator system because it is chemically inert and is not activated by neutron irradiation.

Radiolysis of the heavy water moderator in the calandria results in production of deuterium and oxygen gases. The cover gas system prevents accumulation of these gases by catalytically recombining them to form heavy water. Thus deuterium and oxygen concentrations are maintained well below levels at which an explosion hazard would exist.

The system essentially consists of two compressors and two recombination units which form a circuit for the circulation of cover gas through the calandria relief ducts. Normally one compressor and both recombination units operate and the other compressor is on standby.

Another function of the cover system is to prevent the normal and transient pressures at the lowest row of calandria tubes from exceeding the design limit.

Moderator Liquid Poison System

The liquid poison system is provides a facility for adding soluble neutron poisons with large neutron capture cross-sections to the moderator D_2O. The liquid poisons employed are boron, as boric anhydride (B_2O_3), and gadolinium, as gadolinium nitrate ($Gd(NO_3)_3$., $6H_2O$) dissolved in D_2O. This is discussed in detail in the manual " Moderator Purification System".

The moderator liquid poison system adds negative reactivity to the moderator when required and also provides adequate neutron poison in the moderator to prevent criticality during reactor shutdown.

Moderator Purification System

The moderator purification system uses ion exchange columns to keep the heavy water within the chemistry control requirements of the moderator system. It also removes liquid poisons from the moderator heavy water.

D_2O Sampling System

There are several sampling points on the moderator system which permit the operator to take samples for laboratory analysis. Samples can be taken from the main moderator system, the D_2O collection system, the moderator purification system and the D_2O cleanup system.

D_2O Collection System

The heavy water collection system is designed to collect the leakage from the pump seals, the valve packings and the flange gaskets of the heat exchangers. It is also used to collect drained and vented heavy water from the pumps and heat exchangers during maintenance. Depending on the results of the tests done on samples of the collected heavy water, it may be sent for cleanup, upgrading or disposal.

D_2O Vapour Recovery System

An independent D_2O vapour recovery system is provided for the main moderator system area within the vapour barrier because of the downgraded isotopic purity of the D_2O recovered from the area and the high tritium activity in the moderator D_2O, i.e., approximately 3.0 Ci/*l* after one year compared to 0.15 Ci/*l* of the heat transport system D_2O. Recovered D_2O is collected in the recovery system moderator D_2O collection tank and returned to the D_2O clean up system.

D_2O Supply System

The moderator D_2O is transferred to and from the main moderator system through a line connected to the pump common suction header. This line is connected to the high tritium D_2O transfer section of the D_2O supply system.

Recirculated Cooling Water System

The recirculated cooling water system supplies cooling water to the shell side of the heat exchangers. The temperature of the moderator heavy water is controlled by controlling the flow of cooling water through the heat exchangers.

9 Management of Radiation Hazards

The moderator system is a major source of radioactive materials, however, because it operates at low temperature (about 70°C) and pressure (about 0.6 MPa gauge), it has a much lower heavy water escape rate than the HT system. The moderator contains activated corrosion products and activated heavy water products, such as N-16 and O-19, and also tritium.

Because the principal materials used in the construction of the moderator system are austenitic stainless steels and zirconium alloys, the concentration of corrosion products is generally lower than in the HT system. On the other hand, the activity of tritium in the moderator is higher than that of the HT heavy water by a factor of about 30.

The principles used to limit exposure to the plant personnel are the same as those used in the heat transport system. Specifically for the moderator system, the equipment is located in areas which have limited, controlled access. The area where the pumps and heat exchangers are contained has a vapour barrier and is equipped with liquid and vapour recovery systems separate from the HT systems. All of these measures follow the overall tritium control philosophy which includes the following elements:

- design components for minimum leakage,
- eliminate unnecessary mechanical joints and components where it is practical and cost effective,
- provide a liquid heavy water recovery system,
- provide a heavy water vapour recovery system,
- provide separate liquid and vapour recovery, cleanup and upgrading systems for HT and moderator heavy water.

In addition to these barriers external and internal exposure to plant personnel are controlled by providing training in radiation protection theory and procedures. External dose is controlled by making appropriate measurements of radiation conditions and application of time, distance and shielding. Internal exposure control is achieved by contamination control practices, good hygiene procedures and the use of protective clothing. Doses are measured using external dosimeters and appropriate bioassay techniques.

Although the moderator heat exchangers have been built to rigid specifications in CANDU stations they have on occasions leaked. As the moderator is maintained at higher pressure than the cooling water at some stations the leakage is from the heavy water to the light water side. This creates concern because of the loss of D_2O and the release of tritium into the cooling water. At Pickering where the cooling water is pumped directly from and to the lake this has on occasions caused considerable public alarm because of the possible impact on drinking water for the region. The releases have been within limits established for the stations but nevertheless the events are troublesome and strengthen the need for reliable heat exchangers not susceptible to leakage.

10 Comparison with Main Moderator Systems in other Candu Stations

The CANDU 6, Bruce B and Darlington A main moderator circuits are similar in concept with slight differences. Each has 2 x 100% main pumps and 2 x 50% heat exchangers in a parallel-series arrangement such that either or both pumps can be utilized with either or both heat exchangers. The CANDU 6 uses the main pumps operated at 1/4 speed on Class III power to provide moderator circulation on loss of Class IV power and during reactor shutdown. The Bruce and Darlington units use smaller separate pump-sets in parallel with the main pumps to perform this function. Other differences in the main pump inlet and outlet piping exist but these differences do not result in different operating modes.

The Pickering main moderator circuit consists of 5 x 25% pumps and 2 x 50% heat exchangers. The pumps are arranged in two banks of two and three pumps respectively. There are cross ties downstream of the pumps in order to permit the operation of any four pumps and either of the two heat exchangers. The pump motors are supplied with Class III power in order to improve availability of the pumps and thus minimize the possibility of boiling in the calandria. In the event of loss of Class IV power the running pumps will remain running supplied by Class III power. It will be operator's decision to switch off all but one of the pumps to minimize the load on the Class III power system and still maintain the moderator at an acceptable temperature.

The Pickering "A" units are provided with a unique gravity dump of moderator into the dump tank to augment the shutoff rods. This system allows rapid draining of the moderator water from the calandria to the dump tank assuring reactor shutdown.

There are differences in operating flows and temperatures due to size differences and hence different fission heat loads. In addition the moderator heat exchangers in Bruce, Darlington and Pickering units are cooled using fresh water from a lake, whereas in the CANDU 6 plants the moderator is cooled by recirculated cooling water which is in turn cooled by raw service water. This difference also results in a difference in the moderator temperatures in the various units.

Moderator Purification System

Training Objectives

On completion of this lesson, the participant will be able to:

- Briefly outline the safety related function of the moderator purification system.
- List three process related functions of the moderator purification system.
- Briefly explain why it is necessary to maintain the moderator water as pure as can be achieved.
- Explain why removal of Boric Acid by ion exchange is more difficult than the removal of Gadolinium Nitrate.
- Explain why the moderator purification system must be taken out of service when using gadolinium for reactivity control.
- List three reasons why gadolinium may be added to the moderator.
- Briefly describe when and why gadolinium must be added to a moderator following a shutdown.
- Given an unlabeled diagram of a typical moderator system with the purification shown, mark on it the following:
    - Purification inlet and outlet
    - Filter, filter inlet and outlet, filter bypass valve
    - Ion exchange columns
    - Purification heat exchanger.
- Outline the typical uses of the ion exchange columns for a system with five columns available.
- Define the term "purification half life".
- Explain why the moderator purification system is taken out of service while the reactor is shut down.

- Explain why temperature control of water entering the moderator purification system is important and give the approximate value of the upper limit for temperature.
- State two ways in which flow may be increased through the moderator purification system and state the potential problem if flows are too high through an individual IX column.
- Briefly describe the interlock of the moderator purification system with the Shut Down systems.
- Briefly describe how moderator Boron concentration can be maintained with the purification system in service.
- Briefly describe the two step removal process to remove a "high" concentration of Boron from the moderator system.
- Briefly describe the process of removing a "high" concentration of Gadolinium from the moderator system.
- Briefly describe the addition and control of gadolinium to compensate for the xenon transient after a reactor shutdown, and during a subsequent start-up. (for a start-up after three days delay)
- Briefly state the effect of each of the following on moderator purification system operation:
  Loss of Class IV Power
  Loss of Service Water to One Moderator Heat Exchanger
  Loss of Instrument Air.
- Given any of the interdependent systems from the text, briefly state the interdependence.
- List and briefly explain three radiological hazards of the moderator purification system.
- For each of the radiological hazards, briefly state how they are controlled.

Table of Contents

1. Introduction

The moderator purification system is provided for chemistry control.

The primary objective of chemical control of the moderator is to minimize the radiolysis of heavy water which leads to an accumulation of deuterium and oxygen gasses in the cover gas, presenting an explosion hazard. To reduce the effects of radiolysis, it is necessary to maintain the system as free of impurities as possible. This has three side benefits, keeping parasitic neutron absorption to a minimum, keeping activated species in the moderator system to a minimum, and keeping the system pH (pD actually) in line with the requirements for system materials of construction.

The next objective is to provide appropriate reactivity control. Neutron absorbing chemicals (poisons) are added to the moderator to provide negative reactivity to shut down the reactor or to compensate for excess reactivity caused by fresh fuel or absence of fission products such as Xenon-135. The concentration of these chemicals is controlled by removal in the purification system. Conversely, the purification system must be isolated to prevent removal of poisons during shutdown or upon SDS2 action.

Lastly, the system provides control of corrosive, erosive, and depositing impurities. Depositing impurities are notably bothersome from a maintenance point of view; as many of them have been activated in the neutron flux, resulting in high radiation fields.

The moderator purification system accomplishes these tasks by using filters for particulate removal and ion exchange columns for removal of ionic impurities and neutron poisons which are ionic.

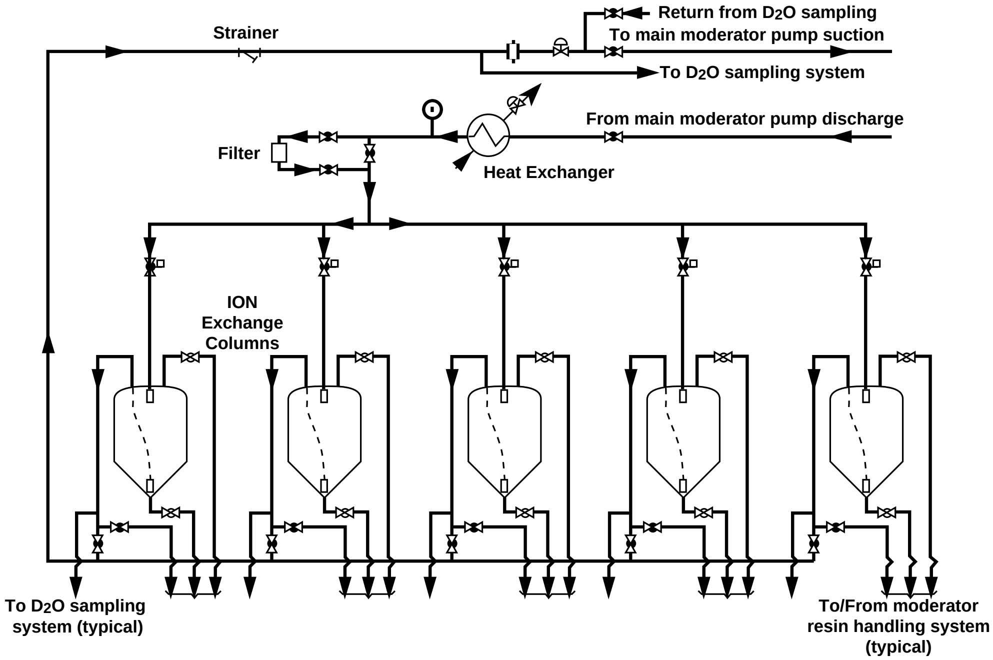
2. Functional Requirements

2.1 Safety Related

The safety related functional requirement for the moderator purification system is as follows:

To prevent inadvertent criticality (by removal of negative reactivity due to poisons) while the reactor is shutdown and either or both shutdowns systems are not poised, the moderator purification system must be automatically isolated from the moderator system. The purification system can be returned to service only when both shutdown systems are restored to the poised position.

Figure 1
Moderator Purification System



2.2 Process Related

The process related functional requirements for the moderator purification system are as follows:

a. Normal Purification

Maintain the purity of the moderator D_2O which, in turn, prevents excessive accumulation of D_2 through radiolysis and minimizes corrosion of components. Normal purification is achieved by passing a portion of the main moderator flow through the purification loop.

The accumulation of deuterium gas in the cover gas is strongly influenced by the presence of certain ionic impurities in the moderator. The process of radiolysis is often thought of as a simple process wherein heavy water, under the influence of gamma rays and fast neutrons, decomposes to deuterium, oxygen and a small amount of deuterium peroxide. The process is actually a complex sequence of intermediate reactions involving molecules, atoms, ions, electrons and free radicals. Unlike ions, free radicals are uncharged. They are also highly reactive. During the radiolysis process, radicals of D and OD are formed. These will combine with each other as follows:

$D + D \text{ ----> } D_2$
$D + OD \text{ ----> } D_2O$
$OD + OD \text{ ----> } D_2O_2$

The deuterium peroxide will break down to heavy water and oxygen gas with the peroxide approaching an equilibrium concentration. The presence of impurities which can react with the OD radicals (eg Cl^- + OD----> Cl[chlorine atom] + OD^-) will remove OD radicals, reducing the likelihood of the middle reaction above which regenerates heavy water. Thus, the top reaction will proceed to generate deuterium gas, possibly to the extent of a cover gas excursion if impurity concentrations are high.

It should be noted that the purity is so high that a pH measurement will not give a very meaningful result. This is why (for Ontario Hydro at least) there is no specification for pH. The target of 5-7 for pH is left below neutral as the residual gadolinium nitrate and any nitrates from radiolysis of air in the cover gas will tend to depress the pH.

b. Reactivity Control

Remove the soluble poisons, boron, as D_3BO_3 and gadolinium, as $Gd(N0_3)_3$, in response to reactivity demands for positive reactivity. Gadolinium is used for short term purposes such as Xenon transients during start up. Boron is used for long term shim control to compensate for excess reactivity in the initial core. Shim control may also be used when a fuelling machine outage has been anticipated and the reactor has been deliberately overfuelled to allow extra time before fuelling is required. (Pickering NGS-A use Boron in place of Gadolinium to cater to their very large cover gas systems. ie to prevent accumulation of radiolytic deuterium caused by the highly ionic gadolinium nitrate.)

Gadolinium nitrate is so highly ionic that the purification system must be taken out of service when using gadolinium for reactivity control. If the purification were in service, the ion exchange resins would rapidly remove the poison from the moderator, drastically reducing its negative reactivity worth. Use of the purification circuit for normal clean-up and control of corrosion and radiolysis is therefore not available. Boric acid is only slightly ionized. Not all of it is removed on a pass through an ion exchange column (typically 1/7 remains). In fact, when using boron with purification in service, the IX columns equilibrate with the boron in the moderator and the purification system can be left in service, maintaining moderator boron concentration while performing its normal duties. Some adjustment of moderator boron concentration can be made by changing the IX column in service to one of lower boron concentration to lower the boron in the moderator or by placing one of higher boron concentration in service to elute boron into the moderator to raise the poison concentration.

Remove the soluble poison, gadolinium, under three conditions:

i. Gadolinium Removal After Poison Injection Shutdown
   Gadolinium poison is injected by shutdown system, SDS2, for a fast shutdown of the reactor. In order to provide for a reactor restart without

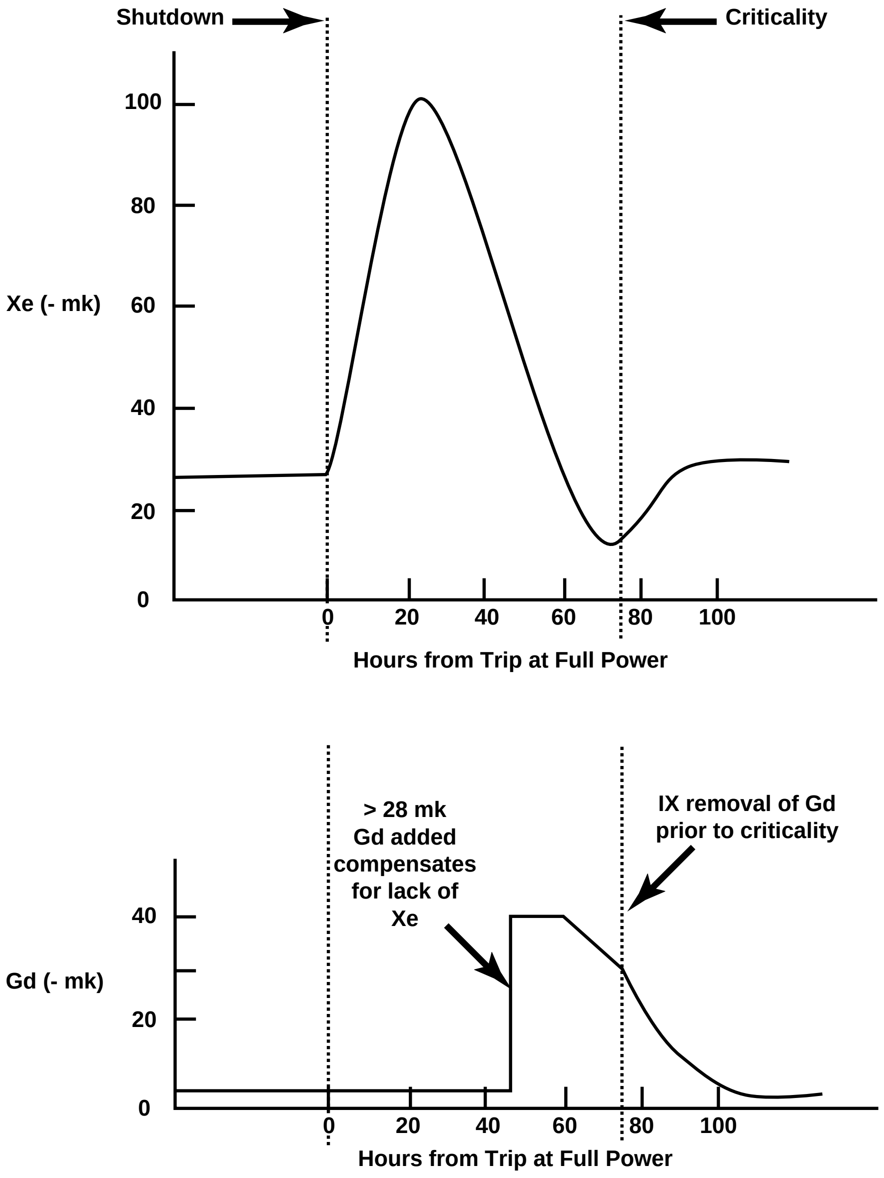
excessive delay, the gadolinium must be removed during the reactor poison out period i.e., before about 30 hours after shutdown.

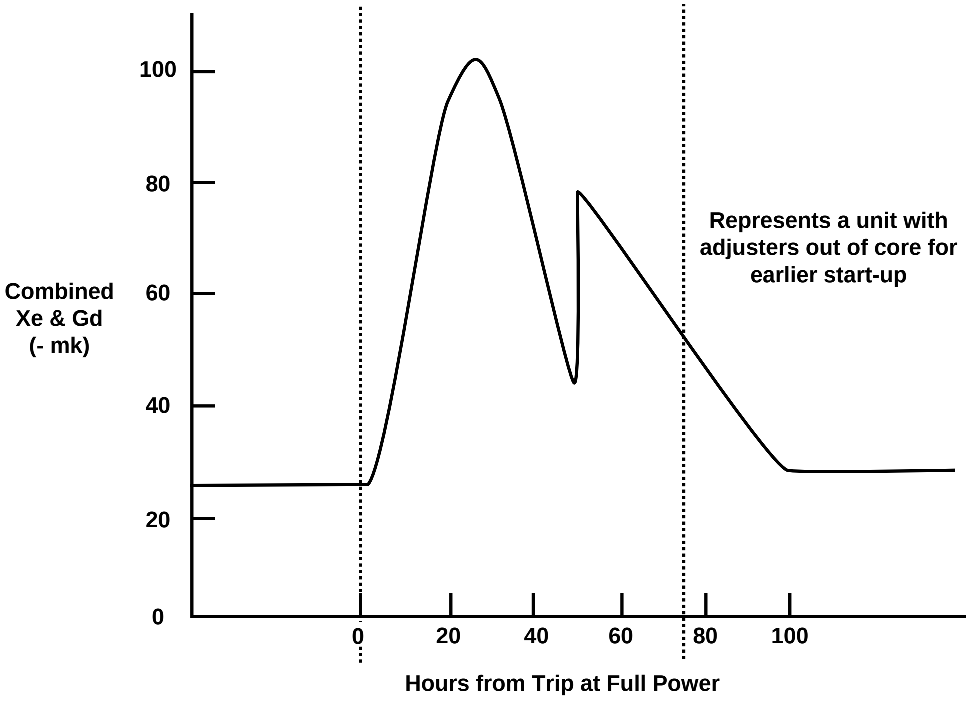
ii. Gadolinium Removal After Guaranteed Shutdown
If maintenance is to be performed, gadolinium is added to the moderator to provide enough negative reactivity that the reactor cannot go critical under any credible circumstance.

iii. Gadolinium Addition and Removal After a Reactor Shutdown - (Shutdown Not Caused by Action of SDS2)
When the reactor has been shut down for a period exceeding 35 hours gadolinium poison is added to the moderator to compensate for the lack of xenon, Xe-135, in the fuel. This ensures that the reactor stays sub critical. During the 35 hour poison-out period, Xe-135 goes through a transient and decays below its normal equilibrium level while at power. Following the return to full power, Xe-135 builds up to its equilibrium value (-25 to -30 mk). During this period, two of the factors affecting reactivity are the gradual build up of Xe-135 to its equilibrium value along with the burn out by the neutron flux of the gadolinium which was added to compensate for the lack of Xe-135 during the shutdown (Xenon shim). The gadolinium concentration must be adjusted by removal in the Purification System. This is in addition to removal by burnup. Once Xe-135 reaches its equilibrium value there shall be no gadolinium remaining in the moderator. The reason for this is that any remaining gadolinium nitrate will represent a high ionic system load, causing problems with respect to deuterium formation from radiolysis, even though the gadolinium isotopes may now all be non-neutron absorbing varieties.

Figure 2

Typical curves for a Xenon transient after shutdown along with Gadolinium addition when the negative reactivity of a poison is required to account for Xenon decay.





3. System Description

Note: For the balance of this document, numerical data and equipment specifications are quoted for a CANDU-6 but are typical of other plants. Participants who need accurate data for other plants are cautioned to consult individual plant manuals.

The CANDU-6 moderator purification system forms a closed circuit and recirculates a portion of the moderator D_2O (see Figure 1 for a typical installation). The supply comes from the discharge header of the main moderator pumps and the return connects back to the suction lines of the two main moderator pumps. The purification flow is taken from the pump outlet rather than from the main moderator heat exchanger outlet to maximize available pressure drop and to minimize variation of purification temperature. (Nevertheless, some plants do take the purification inlet flow from after the main moderator heat exchangers.)

The system consists of the following major components:

- Five ion exchange columns. Each ion exchange column contains 200 litres mixed anion/cation resins with a maximum flow rate of 8.3 kg/s at 49°C through each column.
- One filter upstream of the ion exchange columns for removal of particulate matter.
- One purification heat exchanger for cooling the flow of moderator D_2O leaving the main moderator circuit before entering the filter and ion exchange columns.
- One strainer downstream of the ion exchange columns for removal of resin

beads which may escape due to failure of the resin retaining elements.

- All associated piping, valves in the ion exchanger columns and sampling connections.

The purification heat exchanger cools the purification flow which originates upstream of the moderator heat exchangers, from 69˚C to 49˚C, before the D_2O enters the ion exchange columns. The purification flow then passes through the filter to the ion exchange column(s). One ion exchange column is utilized for normal purification. Up to two columns are used for gadolinium removal, depending on whether moderator purification follows short term reactivity excursions (one column required) or liquid injection shutdown (two columns). For the latter case, two columns are used in parallel since a cleanup within the poison-out period is desirable. Boron removal utilizes a two-stage sequence, one column for high concentration range and the other for low concentration range, with one column used at one time.

The purification half-life is the time required to reduce the original concentration of impurities by a factor of two. The half -life is a function of purification flow and main system size. Using one IX column, the normal purification half-life for flow of 8.3 kg/s at 49˚C is approximately 6.2 hours. Using two IX columns in parallel, the purification half-life based on gadolinium removal for flow of 16.7 kg/s at 49˚C following liquid injection shutdown SDS2 is 3.1 hours. The total time for the removal of gadolinium following LISS (SDS2) is about 24 hours.

The material of construction for both the ion exchange columns and the filter, as well as all the valves, strainers and piping, is generally stainless steel, 304 or 304L.

The inlet temperature to the filter and ion exchange columns are controlled at a constant purification temperature of 49˚C by modulating the heat exchanger cooling water valve.

In order to safeguard against inadvertent critically while the reactor is shut down over a long term, the moderator purification system is automatically isolated from the main moderator system by the closure of ion exchange column inlet valves. These valves remain closed until shutdown system SDS1 and SDS2 have been repoised. The isolation of the moderator purification system will also occur in the event of unavailability of either SDS1 or SDS2. The valves are designed to fail closed on loss of instrument air or control signal.

The moderator purification system is seismically qualified to the same standards as the main moderator system, namely DBEA, except for containment extensions and the inlet pneumatic valves to the ion exchangers which are seismically qualified to Design Basis Earthquake B (DBEB).

The purification system is environmentally qualified to enable isolation from the main moderator system after a loss of coolant accident (LOCA), i.e., the inlet pneumatic isolating valves for column are qualified for the steam environment and radiation.

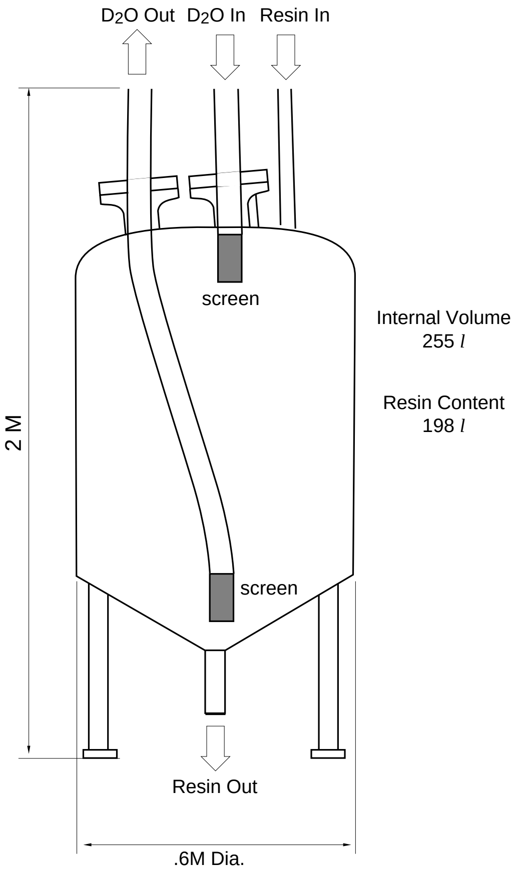
4. Major Equipment/Components

4.1 Ion Exchange Columns

The five ion exchange columns each hold 200*l* of ion exchange resin. They are designed to prevent the resin from entering the main moderator system, and to be loaded and unloaded with resin using slurry techniques. For this purpose, the column incorporates a conical bottom to facilitate resin removal by resin slurry. Resin addition is via a penetration at the top of the column. Materials of construction are stainless steel (type 304*l*) wherever there is contact with D_2O. The type of ion exchange resins used in each column is selected on the basis that boron is only removed by a strong anion resin mixture whereas, gadolinium is removed by a mixed bed of cation and anion resins in equal equivalent ratio.

Figure 3

Typical Ion Exchange Column



4.2 Purification Cooler

This heat exchanger is rated at 1.42 MW and sized to maintain an outlet temperature of 49°C with a purification flow rate of 16.7 kg/s based on two column operating simultaneously and a moderator temperature up to 69°C. The cooling water comes from the recirculated cooling water system at a flow rate of 41 l/s.

The tube material for this heat exchanger is Incoloy 800 chosen for its resistance to general and chloride stress corrosion (from the cooling water side). The shell side is carbon steel.

4.3 Filter

The filter, consisting of a replaceable basket in a permanent stainless steel vessel, removes any insoluble debris from the incoming flow before it passes through the ion exchangers. The filter may be by-passed during removal of gadolinium following operation of the liquid injection shutdown system. This should only be done if the desired flows cannot be achieved.

5. Layout

For a CANDU-6 the moderator ion exchange columns are located below the basement floor of the service building. Access to the columns is through removable shielding slabs in the basement floor. Valves over 2.5 cm., piping and the purification cooler are located in a shielded but accessible gallery above and to one side of the ion exchange columns. The walls of the gallery are 0.6 m thick and extend to the ceiling. The purification cooler is located at the top of the gallery and above a 0.5 m thick concrete shielding floor. The manually operated valves have extensions into the accessible areas. Other plants have similar shielding and access arrangements, not necessarily in the same location.

The sampling facilities and conductivity cells are located on the ground floor of the service building. Pressure and temperature read-out is available on demand from the control computer in the main control room.

6. Control, Monitoring & Diagnostics

6.1 Temperature Control

The inlet temperature to the filter and the ion exchange columns is controlled by modulating the amount of cooling water passing through the shell of the purification heat exchanger.

On loss of Class IV power, recirculated cooling water to the purification heat exchanger is interrupted in order to divert the limited water available to meet more immediate needs.
Annunciation of high temperature is provided in the control room.

Excessive temperatures (>65°C) lead to resin damage and release of contaminants along with gadolinium and/or boron to the moderator bulk fluid.

6.2 Purification Flow Control

The total purification flow through the ion exchange columns can be adjusted remotely by means of a hand controller located in the control room.

Normal purification flow is 8.3 kg/sec. However, for quick removal of gadolinium poison the flow may be increased up to double its normal value by using two gadolinium removal columns in parallel, and by bypassing the filter.

A low flow alarm is provided which is operative only when at least one of the five ion exchange column inlet valves is fully open.

Depending on the station and number of IX columns in service, flows from 5-25 kg/sec are available. In some stations, high flows in individual columns have resulted in resin attrition leading to release of resin fines to the moderator and plugging of column outlet strainers.

6.3 Interlock to Prevent Inadvertent Criticality

To guard against inadvertent criticality during a long shutdown, the ion exchange column inlet valves are interlocked with the shutdown systems SDS1 and SDS2 in such a way that none of these valves can be opened if either shutdown system is unavailable. Examples of unavailability are:

a) SDS1 is considered unavailable if more than two shutoff rods are not fully withdrawn.
b) SDS2 is considered unavailable when helium tank pressure is less than a specified value or when more than one injection tank is not available.

6.4 Miscellaneous Instrumentation

Instrumentation is provided for CRT readout of differential pressure across the filter and the ion exchange columns. High differential pressures are annunciated.

Local pressure gauges measure differential pressure across the two y-type strainers to indicate their condition.

7. System Operation

The system operation consists of the following modes:

7.1 Normal Startup

When the reactor has returned to full power following a shutdown, the moderator purification system is then restarted. One of the ion exchange columns is returned to service to handle the normal purification and, if required, gadolinium adjustments. Two other columns are available for boron removal, and the remaining two for removal of gadolinium in the event of a liquid injection shutdown.

In the shutdown state, the inlet valves to each column are shut. To restart any column, the inlet valve is opened by a handswitch in the control room. Flow adjustments can then be made by the adjustment of a control valve.

Note that purification may have to be put in service to approach criticality from shutdown if the concentration of gadolinium is high in the moderator from use of SDS-2 or if a guaranteed shutdown had been placed in effect.

7.2 Purification System Shutdown

The purification system is normally in continuous operation as long as the reactor is on power and the moderator circuit is operating. If a reactor trip is initiated by either of the shutdown systems, SDS1 or SDS2, then the inlet valve to each ion exchange column will close automatically. The columns are then left in this isolated state until the reactor is returned to full power or gadolinium cleanup is required.

7.3 Normal Purification

The flow through the system for normal purification is from the discharge header of the moderator pumps, through the moderator purification system heat exchanger, through the filter, through the ion exchange column and then back to the moderator pump suction header.

Normal operation should be with the filter bypass valve closed and the two filter valves open.

For normal periods, when poison adjustment is not required, the flow is 8.3 kg/s through one ion exchange column. Each of the ion exchange columns is loaded with 200 *l* of a mixed bed deuterated ion exchange resin having equal capacities for anions and cations. If a strong base anion resin is used, it will become saturated with boron at the boron level currently in the moderator. This is done by maintaining boron concentration with the poison addition system while removing it with the ion exchange resin until the resin is saturated and there is no effect on the boron concentration in the main moderator system. That is, the ion exchanger inlet and outlet boron concentrations are the same at the desired main moderator concentration.

7.4 Boron Adjustments

Recall that boric acid is not highly ionic and will not be completely removed on a pass through an IX column. If you start with some boron in the moderator and a fresh column, the column will stop removing boron at one point after it has been "borated" and you will have to switch to a new fresh column to continue removal of boron to achieve a new lower moderator boron concentration.
Two ion exchange columns are available for boron removal: one dedicated to

high level removal while the other is for low level removal.

If boron removal is desired, the following steps are taken:

a) determine current boron concentration in the moderator,
b) determine which of the ion exchange column has previously been used for high level boron removal and which column for low level removal,
c) if boron concentration is "high" (ie. > 0.6 mg B/kg D_2O), select the column that is for high level removal,
d) if boron concentration is "low" (ie. < 0.6 mg B/kg D_2O), select the ion exchange column that is for low level removal.

When the desired boron level is reached, the flow to the column is isolated. When the columns are no longer effective, the following steps should be taken:

a. for the high level removal column, discharge resin to the dedeuteration tank of the moderator deuteration/dedeuteration system and add a batch of deuterated fresh resin from the deuteration tank of the same system.
b. for the low level removal column, use for subsequent high level removal. If low level removal is still required, the resin in the other column will have to be discharged and fresh deuterated resin added making it the new low level removal column.

If some boron addition is required at a slower rate than can be conveniently controlled by the boron addition system, the purification system may be used. This requires use of a column that had previously been saturated with boron at a higher concentration than presently in the moderator. Putting this column in service, with the "lower" boron concentration moderator water passing through it, will cause boron to leave the resin in the column and join the process stream. Thus boron is removed from the column and added to the moderator. It is also possible to operate the purification system for a short time at a higher temperature which may cause the anion resin to release boron. This practice would have to be commensurate with temperature limitations for IX resin.

7.5 Gadolinium

Gadolinium is used for three purposes:

(i) Short term reactivity excursions (eg., compensation for the absence of xenon after a prolonged shutdown).
(ii) Liquid injection shutdown.
(iii) Guaranteed Shutdown via the liquid poison addition system.

In the first case when a small concentration of gadolinium is required to be added to the bulk moderator to compensate for absence of xenon after a prolonged shutdown, normal purification must be isolated to prevent removal of gadolinium. As reactor power is raised to full power, the effective gadolinium level is lowered by the neutron flux at almost the same rate as the xenon is increased.

In the second and third cases following large gadolinium input, the following steps must be taken:

a. From the control room, isolate any ion exchange column that was in service prior to the operation of SDS2 or placement of the guaranteed shutdown.
b. Set the manual control of the flow control valve to maximum and allow the flow to pass through the two gadolinium removal columns. If necessary to achieve adequately high flow rates, the filter may be by-passed.
c. When gadolinium removal is almost complete (~24 h) and the xenon-135 has decayed (~30 h), the reactor is ready to restart. The remaining Gd can compensate for lack of xenon-135 during start-up. The burn out of gadolinium during start-up will closely match the build-up of Xe-135; little poison concentration adjustment will be required. Normal purification should not be restarted until full power at equilibrium Xe-135 load has been established.

The resin used in these two ion exchange columns should be the same cation/anion mixture as used in the normal "purification" column.

7.6 Operation Under Upset Conditions

The moderator purification system can operate only if the moderator circuit is operating. Consequently upset conditions affecting the moderator circuit operation will also affect the purification system.

7.6.1 Loss of Class IV Power

On loss of Class IV power, the reactor will trip and the moderator circuit will be supplied with Class III power after about a 90sec. delay. The tripping of the reactor by the shutdown systems will automatically close the inlet valve to each ion exchange column. The isolation of the columns ensures that moderator poisons are not removed during the reactor shutdown period.

When Class III power is available, one moderator pump driven by the pony motor will start D_2O circulation to cool down the D_2O in the calandria. The purification flow will be between about 3.5 to 8.5 kg/s. The lower number is for one column only with dirty column and dirty strainers; the higher number is with two column operation and clean columns and strainers. Both SDS1 and SDS2 must be poised before purification can continue

7.6.2 Loss of Cooling Water to One Moderator Heat Exchanger

If RCW is lost to one main heat exchanger, the moderator D_2O temperature will rise, the heavy water will swell, and a high moderator level signal will be sent to the reactor regulating system for a reactor power setback. The purification system will therefore remain in operation. Note that the resin will release impurities to the moderator at temperatures above about 65°C. The resin will likely suffer structural damage at these temperatures as well.

7.6.3 Loss of instrument Air

On loss of instrument air supply the pneumatic valves at the inlet to the ion exchange columns and the flow control valve will fail closed.

8. Protection Against Overpressure

Overpressure protection is provided for the isolable pressure vessels of this system. Each of the ion exchange columns, the filter and the cooler tube side are provided with a rupture disc.

9 Interdependencies with Other Systems

The moderator purification system interfaces with a number of other systems, each of which provides supporting functions as described below:

a. **Main Moderator System**
Depending on the station, the flow to the purification system leaves the moderator system either before or after the moderator heat exchangers and the purified flow returns to the circuit at the pump suction interconnect.

b. **Resin Deuteration and Dedeuteration System**
The purification system receives fresh resin that has had the light water replaced by reactor grade heavy water (deuteration) and transfers spent resin to the dedeuteration system where heavy water is replaced by light water (dedeuteration).

c. **Moderator D_2O Collection System**
The leakage collected from the purification system is transferred to the moderator D_2O collection system.

d. **D_2O Sampling System**
This system provides for the on-line conductivity measurements and a means for taking intermittent `grab' and continuous D_2O samples downstream of each ion exchange column. Measurement of a specific isotope in the moderator compared with that from a grab sample from a particular IX column indicate when that resin should be replaced.

e. **Vapour Recovery System**
The piping gallery is purged continuously with ventilation flow. Ventilation flow is also provided continuously through each ion exchanger column enclosure. The negative pressure in these spaces ensures no out leakage of tritium in the event of leaks or spills.

f. **Breathing Air System**
Breathing quality compressed air is required to enable persons wearing plastic suits and respirators to enter all contaminated or potentially contaminated areas of the system.

g. **Auxiliary Services**
Instrument air, 120 VAC, 48 VDC power are provided for pneumatic valves, instruments and control valves as required. Instrument air may also be used to dry filters prior to change-out.

h. **Tritium in Air Monitoring**
Radiation monitoring for tritium in the vicinity of the equipment provides an early warning of a release of tritiated water to plant personnel.

i. **D_2O Supply System (at DNGS)**
To supply connections for make-up, on line up-grading, or tritium removal where applicable.

j. **Liquid Poison Addition System**
To supply heavy water for the poison mixing tanks.

k. **Resin Storage Tank**
A shielded spent resin storage tank is located underground to accept the spent resin after it has been dedeuterated. A slurry system transfers the spent resin to the tank. The tank is sized to accept all the spent resin anticipated over the life of the plant.

10. Potential for Radioactive Release and Radiation Hazard to Operator

The major radiological hazards are:

1. When the reactor is at power and the purification system is in service, there is significant gamma (N-16 decay) and neutron (photo-neutrons via N-16 gamma) hazard from any unshielded equipment and piping carrying D_2O directly from the reactor.
2. Crud concentrators such as strainers, filters, tanks and equipment cavities pose a gamma hazard due to the accumulation of activated corrosion products.A beta hazard from surface contamination is also possible and will exist during maintenance when the system is open.
3. There is a tritium hazard from moderator D_2O leakage or whenever the purification system is open to the atmosphere.

The principal hazard with this system is the tritium-in-air hazard. At the design stage, a review of the design is made to incorporate measures to reduce the source of the hazard.

Filters and ion exchangers are shielded; slurrying of resin and filter exchange is done remotely. Valves are fitted with stem connections, thus, the source of hazard is reduced.

The effect of the internal hazard is reduced by confining the moderator valve gallery into an enclosure and connecting this area to the moderator system enclosure (confinement area) which is served by a dedicated drier system.

11. Comparison of Moderator Purification Systems in Different CANDU Stations

The CANDU 6, Bruce B and Darlington A moderator purification systems are identical in concept. Each has a heat exchanger and filter in series with the ion exchange columns which are in parallel. The purification flow rates vary between the reactor types to give similar time constants accounting for the differences in moderator volume. The filters and ion exchangers are then sized for the flows. The heat exchangers differ because of the flows; differences in the cooling water temperatures; and moderator temperatures.

Annulus Gas System

Training Objectives

At the end of this lesson the participant will have acquired the knowledge to:

- State the purpose of the Annulus Gas System.
- List eight safety and five process related functional requirements of the system.
- Given a sketch of the Annulus Gas System, identify the major components.
- Describe the evolution of the system.
- Describe the system operation during:
  - Normal Operation
  - Abnormal Operation
- Describe how overpressure protection is provided in the system.
- State two safety hazards associated with the system and briefly describe how the hazards are created.
- State the concern for operating the Annulus Gas System under the following abnormal states:
  - air in the annulus gas
  - high AGS pressure
  - high moisture content

Table of Contents

1.0 Purpose of the Annulus Gas System

The Annulus Gas System (AGS) provides a monitored, continuous flow of controlled, low pressure, carbon dioxide gas to the annular space between the pressure tube (P/T) and the calandria tube (C/T). The gas acts as a thermal barrier, restricting heat transfer from the H.T. coolant to the moderator. Carbon dioxide gas is used for prevention of corrosion to fuel channel components. The system is designed to detect leakage into the annuli from pressure tubes or calandria tubes. Carbon dioxide has a low thermal conductivity and produces limited activation products.

The system has been designed to detect very small leaks, to enable location of a leak quickly, and to enable location of a leak with the reactor at power.

2.0 Functional Requirements

2.1 Safety Related

- Provide a means for detection of a leak from a pressure tube/calandria tube into the annulus, for assessing the leak rate, for identifying the leak source, and for locating the leak with a minimum of radiation exposure to personnel.
- Minimize the hazards to personnel from carbon dioxide (which is an asphyxiant) and from radioactive isotopes C-14 and Ar-41 which will be normally produced in the channel annuli.
- Maintain a hydrogen/deuterium level below 0.1 percent by volume in the annulus gas to minimize hydrogen pickup by the pressure tubes.
- Maintain a pressure in the annuli at or above atmospheric pressure under normal operating conditions. This is to protect the calandria tubes from collapse and prevents ingress of air into the system.
- Provide over-pressure protection to protect the system if there is a pressure regulation failure and to limit damage if a pressure tube fails.
- **Seismic Requirement**
  This system, with the exception of the containment extension, which is DBE qualified, is not required to be seismically qualified.
- **Environmental Qualification Requirement**
  This system is not required to function in a harsh environment and hence is not required to be environmentally qualified.
- **Grouping and Separation**
  This system is designated as a Group 1 system because it performs a power production function during normal plant operation in addition to performing a required safety function.

2.2 Process Related

- Provide a thermal barrier between the pressure tubes and the calandria tubes, restricting heat transfer from the H.T. coolant to the moderator.
- Provide a means for draining the system.
- Provide a means for detecting and identifying a plugged pigtail.
- Provide a means for continuously circulating the annulus gas to maintain dry conditions in the annulus to minimize pressure tube deterioration.
- Provide a means of purging the system

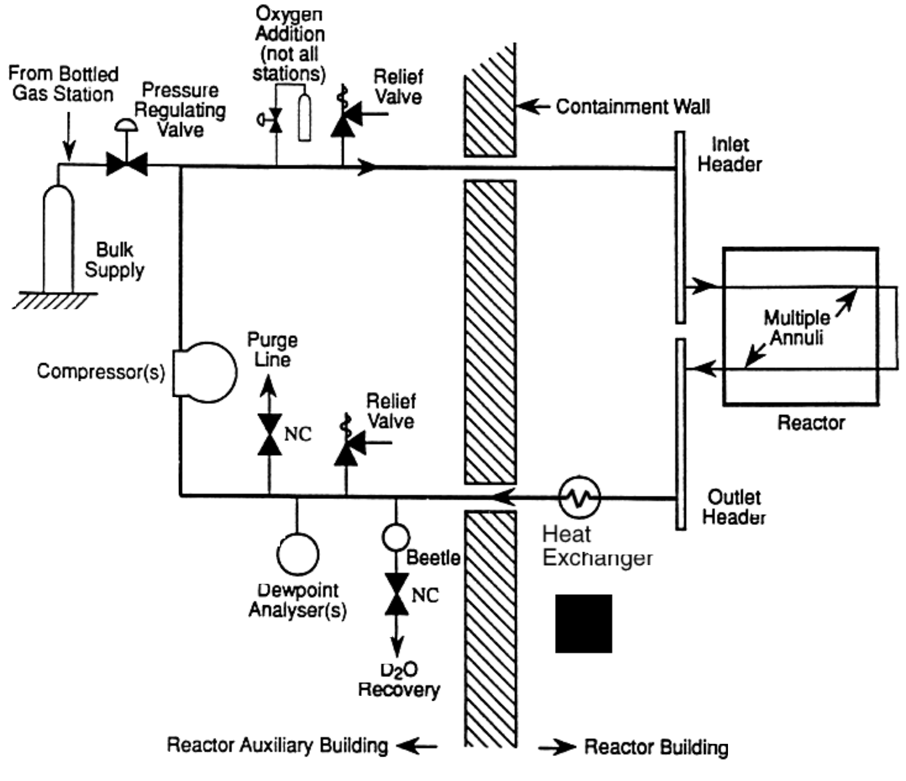
3.0 Description of the System

3.1 General Layout

A simplified version of the system is shown in Figure 1. The system normally operates in the recirculation mode with a nominal recirculation flow rate of 6 *l*/s provided by two of three metal bellows compressors.

Figure 1

Simplified Annulus Gas System



Carbon dioxide flows through the channel annuli in the pressure tube assemblies (Figures 2,3 and 5) via 44 parallel flow paths. The 44 parallel paths contain from 3 to 11 channel annuli in series. Each inlet tube contains a rotameter with an integral outlet needle valve which is adjusted to allow equal flow through each of the 44 lines. When there is a blockage in an annulus the corresponding rotameter indicates lower than normal flow.

The reactor has 380 channels arranged in a lattice with 22 vertical rows containing from 6 to 22 channels per row. The channel annuli are connected in a series-parallel arrangement (Figure 4). In each vertical row, alternate channel annuli are connected in series.

Each row has two inlet connections in parallel, running to the top two channel annuli and two outlet connections in parallel, running from the bottom two channel annuli.

All 44 outlet tubes from the channels are connected to two leakage indicators. A drain connection is provided on the bottom and a vent connection is provided on one end. The indicators are mounted with a slope to facilitate drainage.

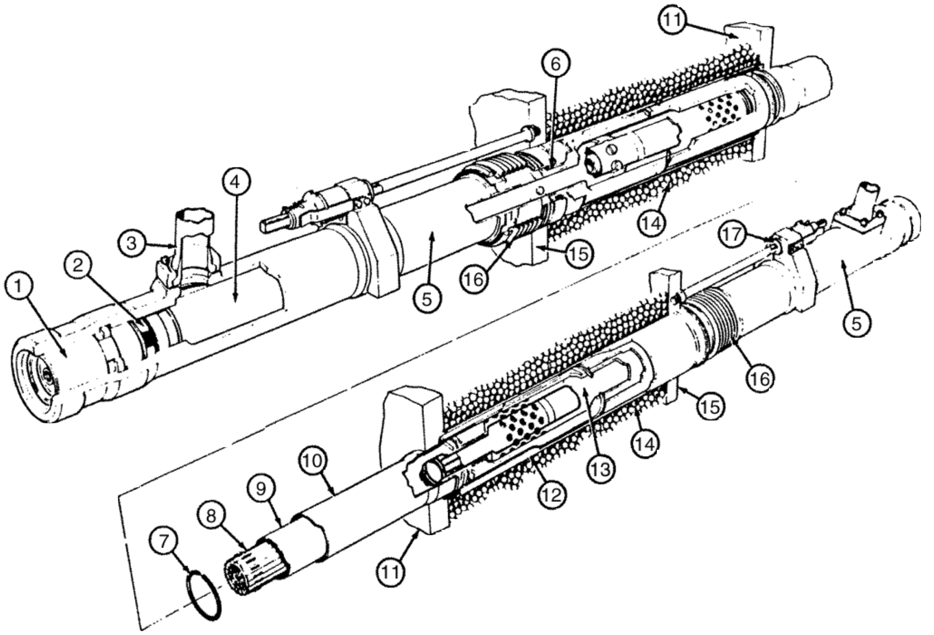
All liquid is drained by gravity to a drain header where two moisture beetles alarm on the presence of moisture. A normally closed valve separates the drain header from the drain tank. The drain tank consists of 6L stainless steel tank mounted vertically and provided with a level transmitter. The tank is used to measure the rate of collection of heavy water (D_2O) with time as an indication of the leak rate.

The vent lines from the leakage flow indicators are connected to a common vent line which leads to a gas-to-air heat exchanger. The heat exchanger cools the annulus gas flow to prevent overheating of the concrete at the vault penetration.

The gas return piping contains a filter to remove debris in the system that may cause reduced compressor performance. Downstream of the filter are the flow, pressure and temperature monitoring instrumentation. A sampling station, with cold finger, is provided to obtain moisture and gas samples for chemical analysis. The return piping also contains two on-line dew point hygrometers in parallel for monitoring the dew point of the recirculating gas.

Figure 2

Fuel Channel Assembly



1	Channel Closure	10	Calandria Tube
2	Closure Seal Insert	11	Calandria Side Tube Sheet
3	Feeder Coupling	12	End Shield Lattice Tube
4	Liner Tube	13	Shield Plug
5	End Fitting Body	14	End Shield Shielding Balls
6	End Fitting Bearing	15	Fuelling Machine Side Tube Sheets
7	Tube Spacer	16	Channel Annulus Bellows
8	Fuel Bundle	17	Channel Positioning Assembly
9	Pressure		

Figure 3

Fuel Channel Detail

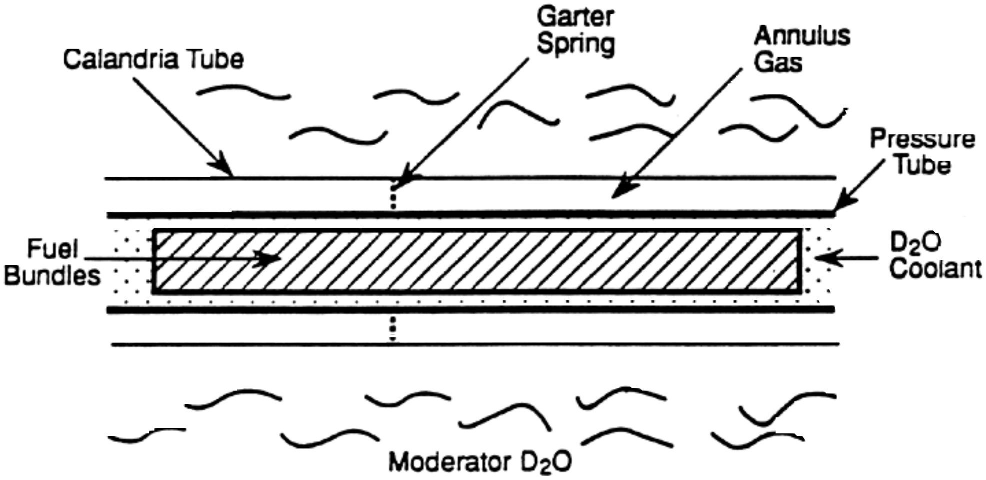
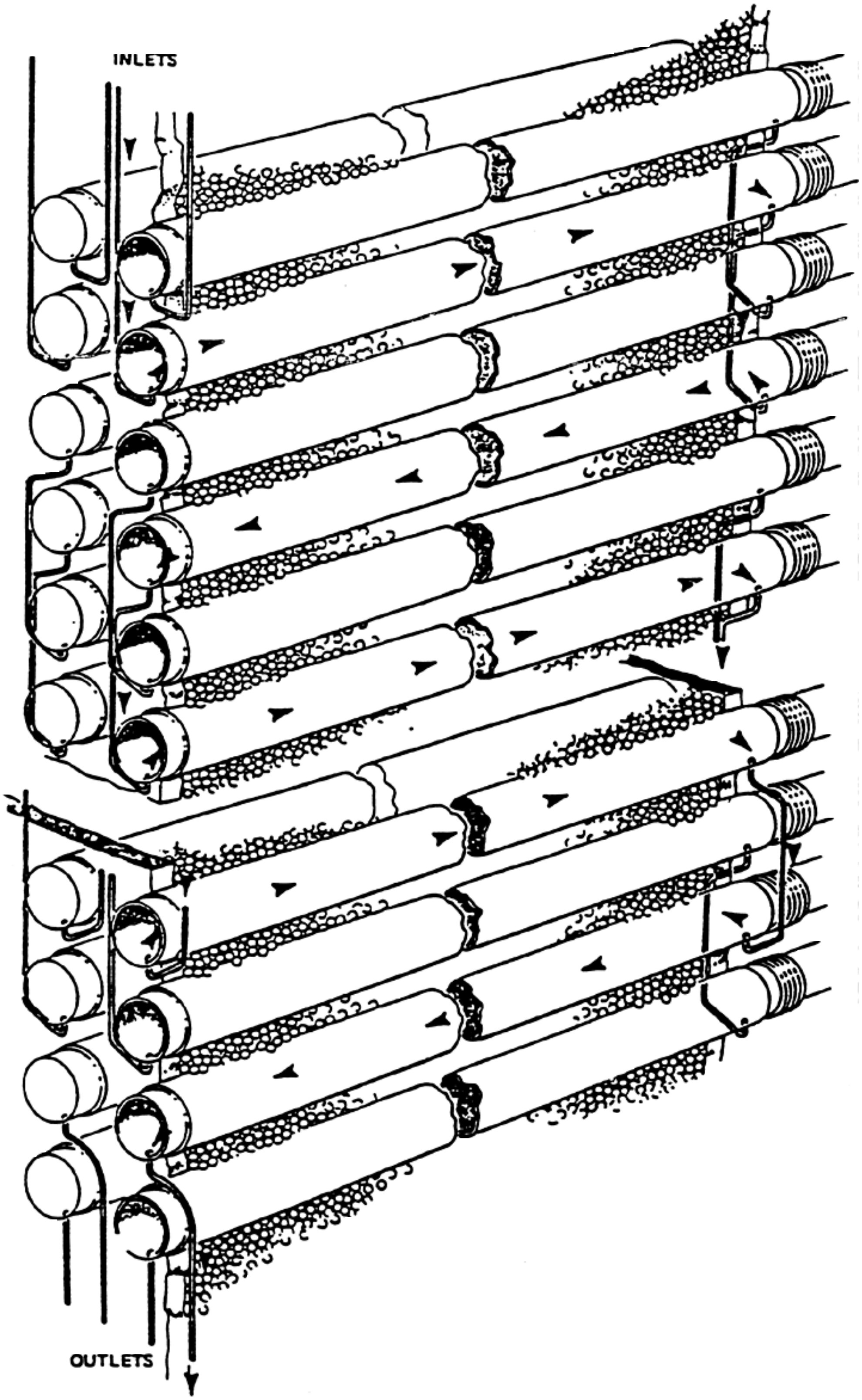


Figure 4
Annulus Gas Channel Flow System



Provisions are made to purge the system to maintain the desired dew point level from 0˚C through -30˚C. It is expected that the purging, to achieve the above dew point level, will reduce the hydrogen/deuterium level.

Dry carbon dioxide with a purity of about 99.9% and a dew point of about -40˚C is supplied from cylinders located in the service building. Each cylinder contains approximately 14.9 m^3 of gas.

The cylinders are arranged in two manifolds of five cylinders each.

Two single-stage low pressure line regulators are located in the supply line outside containment. One pressure regulating valve is connected to the discharge side of the compressor and the other to the suction side. Make-up CO_2 flow occurs automatically through these two pressure regulating valves should the system pressure drop below the set delivery pressure.

A bypass line containing a normally closed diaphragm valve is provided around the line regulator to pass the higher flow required for purging.

Relief valves protect the gas supply and recirculation piping from over pressure.

3.2 Major Components

Compressors

Three x 50% compressors are provided to recirculate the CO_2 gas through the system. A filter is installed upstream of the compressor to remove any debris in the gas. Each compressor is designed to provide a flow of 3 *l*/s CO_2 gas. The recirculation is required to monitor the dewpoint of the CO_2 continuously and annunciate if a leak develops. The compressors normally operate at 30 kPa(g) pressure and a temperature of 65˚C and are located in the basement of the reactor building. The compressors are operated from the main control room.

Heat Exchanger

An air-cooled natural-circulation heat exchanger is provided to cool the gas during purging and to prevent damage to the concrete from overheating at the vault penetration. It is designed to cool 7 L/s of carbon dioxide gas from 232˚C to 65˚C.

The heat exchanger is usually located in the fuelling machine vault and is not accessible during operation.

Filter

The cartridge-type filter is installed to eliminate debris in the recirculating system. The filter medium is a resin-impregnated glass fibre, with a rating of 0.45 micron.

Relief Valves

Relief valves in the system are spring-loaded relief valves with set points of 100 kPa(g). The valves have resilient seats to minimize seat leakage.

Piping

All system piping is stainless steel. The interconnecting tubing between annuli are 6mm diameter. This configuration will limit the pressure rise in adjacent channel annuli in case of major pressure tube failure.

CO_2 Supply Station

The station consists of two banks of 5 cylinders each. The gas should be 99.9% pure with a dew-point of -40°C.

4.0 Evolution of the System

The AGS has evolved from a simple concept with passive features, to a complex system with stringent performance requirements. At NPD and Douglas Point, each of the annuli between the pressure tubes (P/T) and calandria tubes (C/T) were open to the reactor/fuelling machine vault. For NPD, by use of suitable air dampers, a portion of the vault cooling system air flow was directed through the annuli. At Douglas Point, there was no forced air flow through the annuli. Air flow was by natural convection into the fuelling machine vaults. The air gap between the P/T and C/T was required to separate the high temperature HT fluid from the moderator. As the vault atmosphere was found to be too wet, substances which formed in the radiation fields (e.g. nitric acid) were present at levels which were deemed to be deleterious to components in the vault.

Therefore, to protect these components, an enclosed system was proposed for PNGS "A". With the addition of the bellows assembly on the fuel channel assembly, and the interconnecting of the annuli, a closed AGS was created. Its original requirements were to provide a thermal barrier between the P/T and C/T, to maintain a dry and non-corrosive atmosphere for the fuel channel assembly, to provide a rudimentary form of leak detection, and to provide drainage in case of a P/T leak. The heat transferred from the P/T's was removed by the moderator system. The AGS was a low pressure system. The PNGS A system was designed as a recirculating system, but because of compressor vane problems, it was normally operated in the stagnant mode i.e., no forced circulation of the gas in the AGS. Subsequent AGS systems were designed to operate in the stagnant mode.

Following the discovery of pressure tube cracks (due to delayed hydride cracking) at Pickering A Units 3 and 4 in 1975/1976 and pressure tube leaks at Bruce A in 1982, a requirement for improved leak detection by the AGS was identified. Conversion from a stagnant system to a recirculating system was regarded as an improvement to enhance the leak detection capability of the AGS.

The subsequent improved understanding of DHC, and the resulting emphasis on leak detection has required the AGS system to become a sophisticated, high performance, highly reliable system. Small quantities of water must be detected and quantified reliably, in a short time period. This detected water must be related back to the size of the pressure tube leaks. The rapid determination of the leak location is a very important requirement.

5.0 System Operation

5.1 Normal Operation

A flow chart of a typical CANDU 6 annulus gas system is shown in Figure 5. The gas supply piping operates at ambient temperature. The temperature in the annuli during operation is between the heat transport D_2O temperature and the bulk moderator temperature.

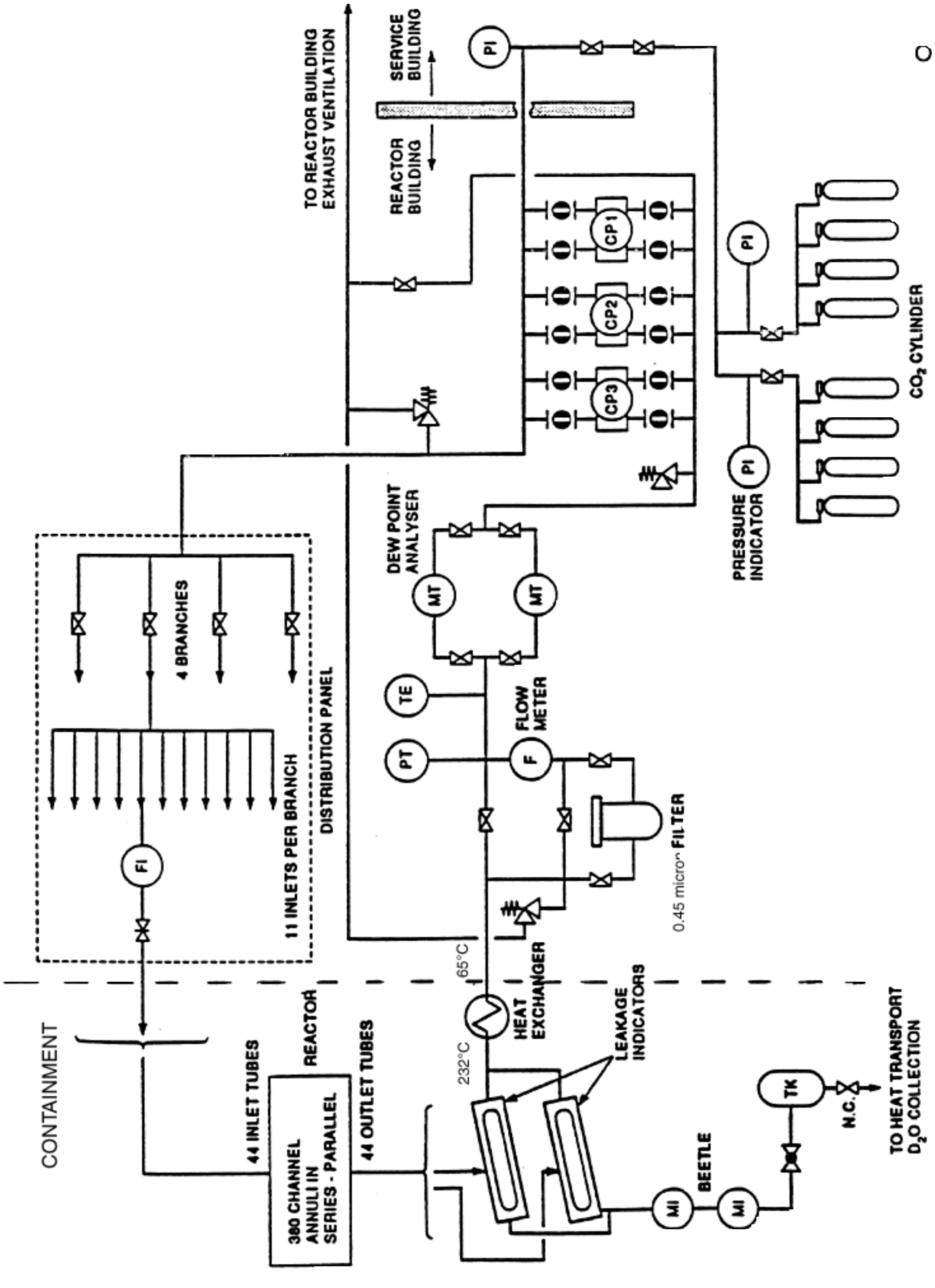
From the cold state, the compressors are running and the pressure is maintained normally by the two pressure regulating valves. The temperature in the annuli will increase as the heat transport and moderator system temperatures rise. As the temperature rises, the gas pressure increases up to the operating pressure. This pressure will be maintained during operation, unless there is leakage from the system. Carbon dioxide make-up is supplied automatically if the pressure in the system drops.

During normal operation, the annulus gas is continuously recirculated through the system for dewpoint measurement. The moisture level under normal operating conditions should not exceed -18˚C dewpoint. If the dewpoint exceeds this value, the compressors are stopped and the system is operated in the stagnant mode. This helps to prevent redistribution of the moisture. If the moisture level shows a continuing tendency to increase, a leak into the system is indicated.

5.2 Draining and Purging

The 44 outlets from the channels are tubed, in parallel, to two visual leakage flow indicators. The drain lines from the flow indicators are piped to a common line to the drain tank. This line contains beetles and a normally closed valve. The drain tank is further connected to the heat transport D_2O leakage collection tank.

Figure 5
Annulus Gas System



The carbon dioxide gas line from the flow indicators is used for purging the system. The system should be completely purged prior to initial start-up and to remove moisture when the dewpoint increases (slowly) during normal operation.

To purge, the system is vented and gas is added. The outlet humidity is continuously monitored, during purging, by a dewpoint analyzer.

5.3 Abnormal Operation

Increasing Moisture Level

A moisture level showing a continuing tendency to increase, indicates a leak into the system from a calandria tube, pressure tube, or end shield leak.

To locate the leak source, the purge flow is passed through each of the four banks of 11 inlets in turn. Recirculation mode is changed to purge mode. Three out of the four header valves are closed at one time and moisture level is observed on a dewpoint analyzer. This will narrow the location down to one of the four banks.

In the bank giving the high moisture reading, the purge flow is passed through each of the 11 inlets in turn by closing the rotameter outlet needle valves in the other ten lines. The line giving the high moisture reading contains the leaking channel. This will narrow the location down to one line which feeds from three to eleven channels depending on the number of channels in series.

If the leak rate is small, it may be possible to control the moisture level by more frequent or continuous purging. If the leak rate is significant as indicated by annunciation of one of the beetles, the leakage indicators can be examined during a shutdown to verify the leaking path.

The methods for final leak source identification are as follows:

a) If the leak source is a calandria tube, the leaking channel can be identified by lowering the moderator level. With the reactor in a cold, shutdown state, the moderator level is progressively lowered while monitoring the carbon dioxide (annulus gas) for helium (moderator cover gas) and the helium for carbon dioxide at each step. In this way, the leaking channel can be identified.

b) If the leak is a pressure tube, the leaking channel can be identified using the channel outlet temperature scanning.

Channel outlet temperature scanning is used to monitor temperature differences in the heat transport coolant leaving a channel, thus, indicating a leak. This technique can be used with the reactor on-power.

6.0 Operational Hazards

6.1 Conventional Hazards

Carbon Dioxide leaks from the Annulus Gas System could result in an asphyxiation hazard in low-lying areas of the Reactor Building as CO_2 is denser than air. Fixed and Portable CO_2 monitors are used for work in affected areas for automatic identification of CO_2 leakage.

Handling of high pressure CO_2 cylinders requires normal safety precautions.

6.2 Radiological Hazards

Carbon-14 is produced in the AGS by the following nuclear reactions:

a) Activation of ${}^{13}_{6}C$
$${}^{13}_{6}C + {}^{1}_{0}n \rightarrow {}^{14}_{6}C + \gamma$$
b) Transmutation of N14 (from trapped air or air ingression)
$${}^{14}_{7}N + {}^{1}_{0}n \rightarrow {}^{14}_{6}C + {}^{1}_{1}p$$

Most ${}^{14}C$ will be produced by the transmutation of ${}^{14}N$.

Carbon-14 hazard requires special radiological protection whenever the system is opened for maintenance which will include as a minimum:
1) a personal air sampler for assessment of dose
2) an air supplied plastic suit

Argon-41 can become a hazard in the system if air in-leakage or low purity CO_2 is used in the AGS. Fixed area gamma monitors are employed to warn personnel of high or increasing radiation fields.

7.0 Operational Problems

7.1 Air in the AGS

Leaks or maintenance work may lead to air ingress to the AGS. Radioactive Hazards (Ar-41) and the formation of nitric acid are a direct result.

7.2 High AGS Pressure

High pressure in the AGS can cause damage to the calandria tubes and bellows seals which join the AGS to the pressure tube and fitting. Pressure relief valves protect the system from high operating pressures.

7.3 High Moisture Content

High moisture content leads to radiological concerns due to the increase in tritium levels in the AGS and the potential for a Loss of Coolant Accident (LOCA) and subsequent fuel and calandria tube damage.

Service Water Systems

Training Objectives

At the end of this lesson the participant will have acquired the knowledge to:

- State the purpose of the Raw Service Water Systems (RSW), the Recirculated Cooling Water System (RCW) and the Emergency Cooling Water System (EWS).
- List eight safety related functional requirements of the RSW and RCW.
- List three safety related functional requirements of the EWS.
- List three process related functional requirements of the RCW and EWS.
- Given a sketch of the Service Water Systems, identify the major components.
- Describe the operation of the Service Water Systems during normal operation.
- State the method of overpressure protection of the Service Water Systems.
- List three accident scenarios that would require the EWS system to be placed in service.
- List the automatic operations that occur in the Service Water Systems during a failure of the class IV electrical system.

Table of Contents

1.0 Introduction and Purpose

The water systems typically required for a CANDU reactor located on a fresh/seawater site include:

- the Condenser Circulating Water system and Raw Service Water system, each using raw fresh/sea water,
- the Emergency Water Supply (EWS) system, and Fire Protection system, each using untreated fresh water from the EWS pond,
- the Domestic Water system, and Demineralized Water system, each using treated fresh water,
- the Recirculated Cooling Water system and Chilled Water system, each using water from the Demineralized Water system for make-up.

The service water systems include the Raw Service Water system, the Recirculated Cooling Water system and the Emergency Water Supply system. These perform the safety function of transferring heat from other safety related systems to the ultimate heat sink. We will limit our discussion to these three systems.

The Raw Service Water (RSW) System is an open loop system supplying cooling water to the RCW heat exchangers and all other components suitable for fresh/sea water cooling.

The Recirculated Cooling Water (RCW) System is a closed loop demineralized water cooling system which supplies water to coolers and components unsuitable for cooling with the RSW due to water quality or requiring higher pressure service.

The Emergency Water Supply (EWS) System is a separate back up system capable of providing an alternate source of cooling water, should the normal heat removal systems be lost.

A simplified flow diagram of these systems and the Condenser Circulating Water system is provided in Figure 1.

Service water system designs differ from station to station, due to specific site features, station requirements and the availability of fresh or sea water. The reference design for this description is a CANDU 6 reactor located on a seawater site. Detailed information refers to the Wolsong 3 & 4 reactors in Korea, the most recent design. A comparison with some Ontario Hydro stations is presented in Table 1,2 and 3

Figure 1
Cooling Water Systems Simplified

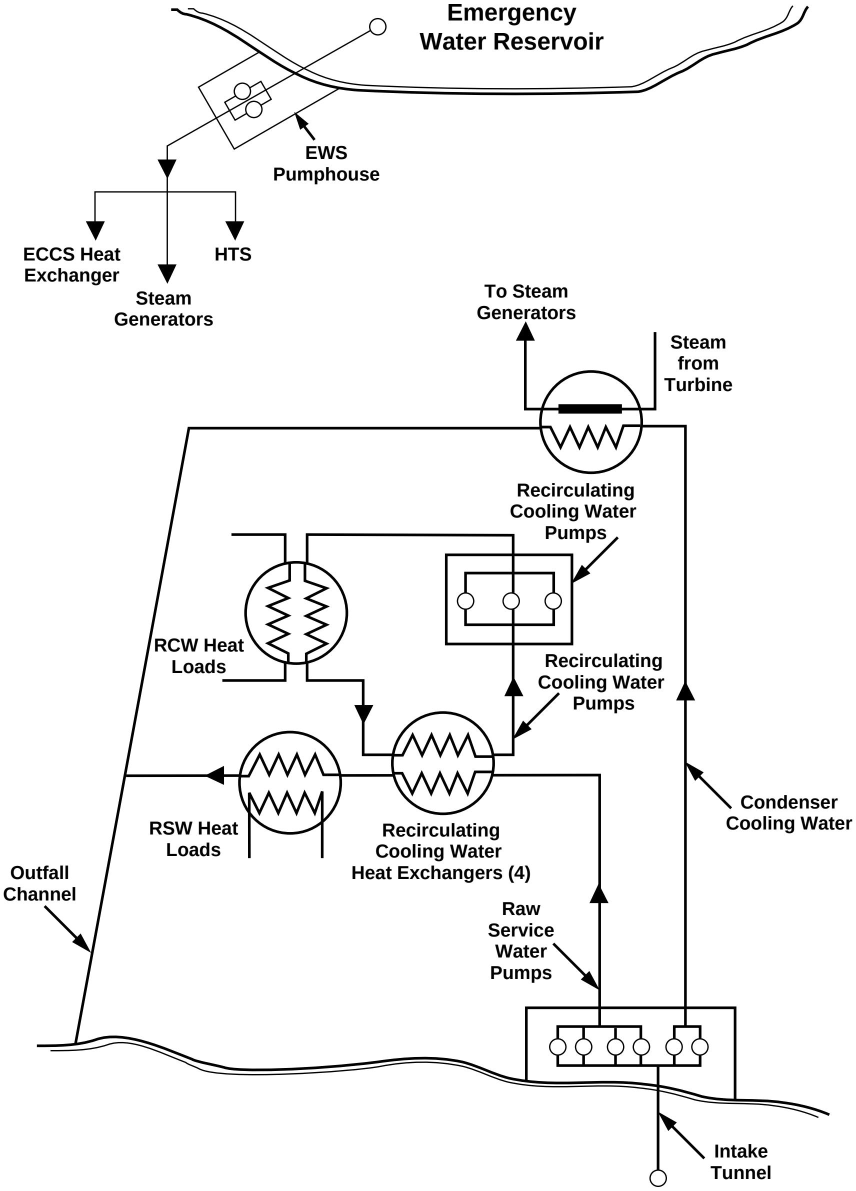
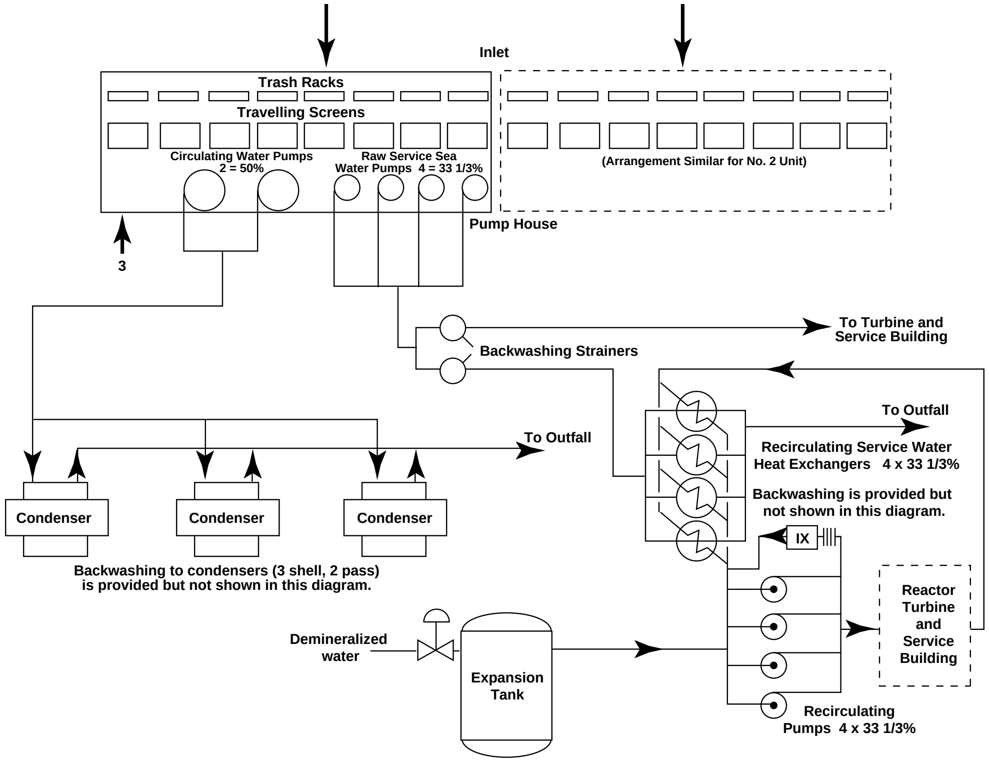


Figure 1a

Circulating Water, Raw Service Water and Recirculated Water



2.0 Functional Requirements

Functional requirements can be grouped into two categories: safety related and process related.

2.1 Safety Related Requirements

2.1.1 Raw Service Water (RSW) and Recirculated Cooling Water (RCW) Systems

The safety requirements for the RSW and the RCW systems are as follows:

- The pumps shall normally be supplied with Class IV power.
- The systems shall be designed such that on loss of Class IV power one pump, of each system, on each Class III bus would be available.
- The systems shall be capable of supplying water to safety systems during the loss of coolant accident with or without the loss of Class IV power.
- A sufficient portion of the service water piping shall be seismically qualified to the Design Basis Earthquake (DBE) to ensure piping integrity to prevent the possibility of flooding of the Service Building basement and to protect Emergency Core Cooling Pumps.
- The systems shall have suitable physical separation from the EWS system to ensure that a common mode occurrence does not incapacitate both systems.
- All reactor building penetrations shall be seismically qualified to DBE to ensure containment integrity.
- Provision shall be made to supply water to the RSW pumps using all intake

screens, including any which are normally dedicated for condenser cooling pump supply. The condenser cooling pumps shall be tripped automatically on low water level to ensure that a sufficient supply of water is maintained to the RSW pumps in the event of screen or intake blockage.

- Services associated with the operation of the service water systems, such as instrument air, lube oil, cooling water, filters, and screen backwash systems, shall be sufficiently redundant to avoid loss of entire service water system through the failure of one of these systems.
- The starting of the service water pumps after loss of Class IV power must not be delayed by any system characteristic or delay requirement, such as delays associated with valve closures. This may involve consideration of the effects of the RSW system emptying, water hammer, loss of air to pneumatic valves, etc.
- Provision shall be made in the RCW and RSW systems to isolate cooling loads not related for safety.

2.1.2 Emergency Water Supply (EWS) System

- **Seismic Qualification**
  The EWS system is seismically qualified to DBE Category B in accordance with the Safety Design Guide (SDG), to supply water to the secondary side of the steam generators following a loss of normal feedwater supply and to the heat transport system for makeup following a DBE. The system vent and drain valves are seismically qualified to DBE Category A because these valves are not required to be operated during a DBE.
  However, the piping and valves from the motorized EWS inlet valves to the ECC heat exchanger connections and the piping and the valves of the return line are seismically qualified to the Site Design Earthquake (SDE) Category B because the earthquake is assumed not to occur within 24 hours after LOCA.

- **Environmental Qualification**
  The EWS system is environmentally qualified in accordance with the SDG.

  The check valves are environmentally qualified to supply water to the steam generators in the event of loss of feedwater under the harsh environmental conditions such as LOCA, small LOCA, or Main Steam Line Break (MSLB).

  Other valves and components are not required for environmental qualification because they are located outside containment.

- **Grouping and Separation**
  In accordance with the SDG, the pumping system and source of cooling water must be sufficiently independent of the cooling water systems (either Service Water or Steam Generator Feedwater systems) so that any postulated common mode events that cause failure of the cooling water systems will not prevent the ability to remove decay heat.

  Since the EWS system must not use sea water, the source of water for the

EWS system must be an on-site fresh water reservoir. Such a reservoir must be physically separated from the main plant service water source.

The controls for the EWS system must not be located in the main control room.

- **Containment Extensions**
  In accordance with the SDG, the EWS system is designed to meet the Atomic Energy Control Board (AECB) Regulatory Document R-7 requirements.

2.2 Process Related Requirements

2.2.1 Raw Service Water (RSW) System

The RSW system is required to feed both essential and non-essential loads. However, the non-essential loads are to be shed in an accident scenario where Class IV power is lost and the reactor is shut down. The reduced flow requirements on Class III power will only require one pump to be operational.

The RSW major loads are presented in Table 4.

2.2.2 Recirculated Cooling Water (RCW) System

- The RCW system is required to feed both essential and non-essential loads. The latter shall be shed on loss of Class IV power. The reduced flow requirements on Class III power require one of the four RCW pumps to operate.

The RCW major loads are presented in Table 5.

- The RCW System shall maintain the temperature to the users at 35°C or less.
- The RCW system shall maintain sufficient pressure to prevent boiling in the system due to heat rejection from the loads. Also the pressure shall be low enough to ensure that in the unlikely event of a heat exchanger tube failure, flow shall be from the D_2O side to the H_2O side, so that the leak will be detected by the D_2O in H_2O detection system.
- The RCW system shall have provision for maintaining its water chemistry within specifications.

2.2.3 Emergency Water Supply (EWS) System

The EWS system is classified as a safety support system. It provides a backup heat sink when the main heat removal system is lost for the following components or systems:

- Steam Generators when feedwater is lost,
- Heat Transport System make-up on loss of Pressure and Inventory Control system (in conjunction with the ECC system).
- RCW lost to the ECC heat exchanger secondary side.

3.0 System Description

3.1 Raw Service Water (RSW) System

There is one RSW system for each unit, hence a two-unit station has two identical RSW systems and so on for multi-unit stations..

The RSW System is an open loop cooling water system which supplies raw cooling water to the recirculated cooling water heat exchangers, and other components listed in Table 4.

The raw service water is supplied by four (4) 33-⅓ percent capacity vertical column pumps, located in each unit pumphouse. Each pump is isolated with a stop log on the suction side, and a check valve and motorized butterfly valve on the discharge side, permitting pump maintenance work at any time as shown in Figure 2.

The pumps discharge into a manifold, and from there into a set of two self-cleaning strainers. The strainers remove all particles larger than 0.25 mm and are automatically back washed as pressure drop increases. Strainers are provided with isolation and bypass valves.

The service water from downstream of the heat exchangers is combined with discharge to the condenser circulating water discharge duct.

For a station with fresh water cooling, the RSW system also supplies water to the water treatment plant which in turn supplies water to the Domestic Water system and the Demineralized Water system. For a station with salt water cooling, a separate fresh water supply is provided to the water treatment plant.

3.2 Raw Service Water (RSW) System Major Components

3.2.1 The RSW Pumps

The raw service water is supplied by four (4) 33-⅓% capacity vertical column pumps, located in the unit pumphouse. They each have a capacity of 2273 *l*/s at 27.6 m head, and are driven by 800 kW motors mounted vertically above each pump. The pump housings are cast iron with stainless steel impellers, and stainless steel shafts are fitted with bronze sleeves. The bearings are pressure lubricated with treated water.

Before entering intake of the pumps, the water is screened through trash racks and travelling screens to prevent debris from entering pump intake.

3.2.2 Backwash Strainers

The pumps discharge into manifolds, and from there into 2 x 50% automatic backwash strainers. Backwashing may be performed intermittently based on pressure drop measurement or continuously if required. Retention size is 0.25 mm, flow rate is typically 3400 *l*/s for a CANDU 6 station. Strainer internals are of stainless steel, while the shell is ceramic metal coated carbon steel.

3.3 RSW System Controls

The RSW pumps are controlled from the main control room and can be started manually by the operator or automatically if one should fail.

The strainers are automatically cleaned, by initiating a timer or an increase in differential pressure.

The trash racks and travelling screens are automatically cleaned by initiating a timer or an increase in differential pressure across the travelling screen.

> Note: The Condenser Cooling Water (CCW) pumps and RSW pumps share a common intake. On high differential pressure across the travelling screens the CCW pumps are shut down to ensure RSW remains available for essential cooling.

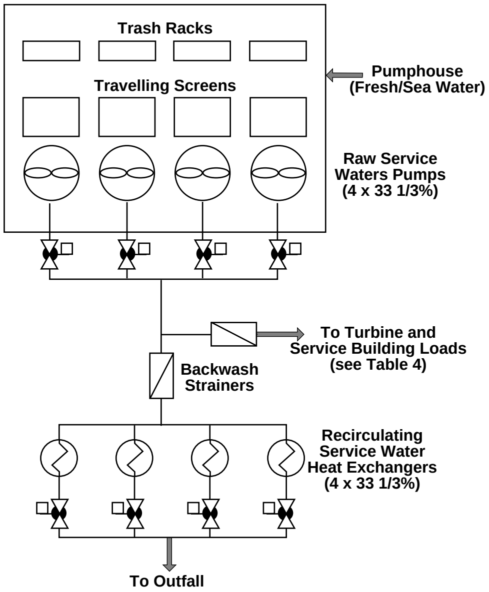
3.4 Raw Service Water System Operation

Three pumps provide the water requirement for all loads during normal operation, the remaining pump provides standby capability in the event of the loss of one pump.

All raw water pumps are connected to the Class III Power Bus. Two pumps in each sub system are connected to the "odd" bus and the others to the "even" bus, with relays arranged to permit only one pump from each bus to run on loss of Class IV power.
In the event of a Class IV failure, non-essential equipment is isolated from the systems, and one pump only is required to supply the requirements of the essential equipment.

Figure 2

Simple Raw Service Water System



3.5 Raw Service Water Overpressure Protection

The piping, fittings, valves are designed to withstand the pump shutoff pressure.

The only components that are protected from overpressure are the heat exchangers since they can be isolated and thermal expansion caused by any heat source on the shell side either internally or externally can overpressurize the RSW tube side.

Pressure relief valves are normally located on a small drain or vent line. Relief capacity is small since it is designed to cater only for thermal expansion of an incompressible fluid.

4.0 Recirculated Cooling Water (RCW) System

4.1 System Description

There is one RCW system for each unit, hence a two-unit station has two identical RCW systems and so on for multi-unit stations.

Each unit RCW system is a closed loop of treated water supplying cooling to all equipment for which salt water is unsuitable. For details of the major loads refer to Table 5. A flow diagram of a CANDU 6 RCW system is given in Figure 3. Flow is provided by four 33-⅓% pumps. The pumps draw water from a common header, and discharge into a common manifold from which the flow is divided into two headers, one supplying the Service Building and the Reactor Building, the other one supplying the Turbine Building. Following distribution to the various users, the cooling water is returned and combined in the turbine building. The water then enters the four 33-⅓% RCW heat exchangers where it is cooled by transferring heat to the RSW. The cooled water then returns to the suction side of the RCW pumps.

The water in the RCW system is treated. A filter is provided and connected across the recirculated cooling water pumps for continuous filtration of water. Also, chemical injection equipment, consisting of a mixing tank and an injection pump, is provided for pH and corrosion control. The chemicals are injected at the suction side of the recirculated cooling water pumps. Make up water is drawn from the demineralized water system.

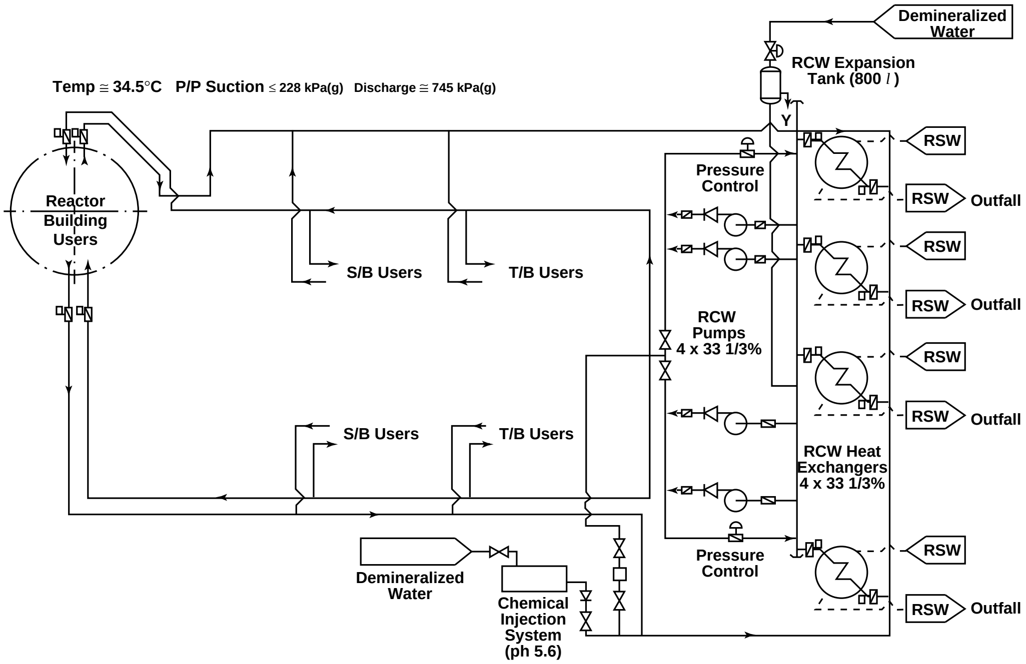
The system operates at 745 kPa(g). An expansion tank, which is connected to the suction side of the recirculated cooling water pumps maintains the system pressure at a constant value.

The pumps and heat exchangers are located in the Turbine Building.

All pumps have their power supplies from the Class III bus - two from one bus and two from the other, with relays arranged to permit only one pump on each bus to run on loss of Class IV power. In the event of a Class IV power failure, non-essential equipment is isolated from the system; two pumps (one pump per bus) start and deliver flow to essential equipment when Class III power is available. Flow is adequate even if one of the two pumps fails to start.

Figure 3

Recirculated Cooling Water System, CANDU 6



4.2 Recirculated Cooling Water (RCW) System Major Components

4.2.1 RCW Pumps

Flow in the Recirculated Service Water System is provided by four (4) double volute, double suction, horizontal pumps each rated at 33-⅓% of total required flow. Each pump delivers 1,535 *l*/s at 49 m. The pumps have carbon steel casings and stainless steel impellers.

4.2.2 RCW Heat Exchangers

There are 4 x 33 ⅓% RCW Heat Exchangers. They are of the tube and shell type, horizontally mounted, located in the Turbine Building. They exchange heat with raw service water flowing in the tube side, while the recirculated demineralized water, flows through the shell side. The heat exchanger tubes are titanium, the shell carbon steel, and the water boxes are epoxy lined carbon steel.

4.2.3 Expansion Tank

The service water expansion tank is large enough to compensate for level variations at start-up and shut-off of the RCW pumps. It is of a vertical straight cylindrical shape, made of fabricated mild steel. The tank has a protective coating against corrosion.The expansion tank has a capacity of 8000*l*.

4.2.4 Chemical Station

A preparation and storage tank for chemical mixing and a dosing pump are employed to add chemicals to the RCW

4.2.5 Valves

Temperature Control Valves
Pneumatically operated ball and globe valves are used for temperature control.

Power Operated Valves
Pneumatically and electrically operated gate valves are used for isolation when valves are required to be operated immediately (i.e. loss of class IV power)

Manual Isolating Valves
Manual isolating gate valves are provided to isolate equipment for maintenance.

4.2.6 D_2O and H_2O Monitoring

All branches of the RCW system which pass through heat exchangers containing D_2O at a higher pressure than the service water are monitored for D_2O.

All the main discharge lines going out of the Reactor Building are monitored for D_2O. A connection is provided outside the containment isolating valves.

4.2.7 Containment Isolation Valves

One butterfly valve is provided on each containment penetration. These valves are seismically qualified to the DBE Category B. They are controlled remotely from the MCR and SCA, and must be shut by the operator in the event of the D_2O leak into the RCW from the moderator heat exchangers or in the event of a RCW pipe break inside containment.

Note that some of these valves serve for equipment isolation purposes as well. This is consistent with containment isolation requirements, and economizes on the total number of isolation valves.

4.2.8 Relief Valves

Relief Valves are provided for all heat exchangers and pumps to protect them against overpressure.

4.2.9 Restriction Orifices

When the flow to the equipment such as the heat exchanger, pump, condenser, and local air cooler is constant, restricting orifices are provided on the discharge line of this equipment to establish the required RCW flow rate through the equipment in an economical way. Restricting orifices is particularly important to the Local Air Coolers inside the reactor building in order to maintain a high back-pressure and avoid RCW local boiling during LOCA events.

4.3 RCW System Controls

The RCW pumps are controlled from the main control room panel and can be started manually or automatically should one fail. One RCW pump per bus is permitted to operate on Class III power.

A pressure transmitter controls the operation of the pumps by-pass valve to maintain the system supply pressure at a preset constant value. When the limit switch on the valve indicates in the control room that the valve is more than 90% open, the operator may re-select the number of operating pumps. A pressure switch signals the STANDBY pump to start via a time delay on low system pressure.

4.3.1 Flow Measurement

The flow going to the Reactor Building is divided into two identical loops. Each loop consists of a moderator heat exchanger, a moderator pump cooler and two local air coolers.

The normal flow rate to each moderator heat exchanger is expected to be 1125 *l*/sec and a maximum flow is 1492 *l*/sec.

4.3.2 Pressure Measurements

The pressure at the inlet and the outlet of the heat transport purification cooler, shutdown coolers, moderator coolers and their associated control valves is measured.

For each moderator heat exchanger there is a pressure gauge installed upstream of the control valve, one downstream on the inlet side to the heat exchanger, and one on the outlet of the heat exchanger.

For every other heat exchanger, there is a pressure gauge installed on the inlet, one on the outlet line upstream of the control valve, and a third downstream of the control valve.

The value of the differential pressure across the heat exchangers or the control valves could be obtained by calculating the difference in any of the two relevant inlet and outlet indicators.

4.3.3 Temperature Measurement

Temperatures on the inlet and the outlet of the heat exchangers are measured by temperature indicators installed in the respective lines. The following equipment is monitored:

- Moderator heat exchangers
- Purification heat exchangers
- Shutdown Cooling heat exchangers
- F/M Room Cooler South
- F/M Room Cooler North
- Boiler Room Cooler

4.3.4 Control of Load Shedding Valves

Load shedding valves are air operated and fail open, on-off type and each is controlled by a 3-position handswitch, close/auto/open. With the valve handswitch in the auto position, on class IV power failure, logic from load shedding causes motorized valves to close automatically. Each motorized valve is equipped with two limit switches which provides remote indication of whether the valve is open or closed.

4.4 Recirculated Cooling Water System Operation

4.4.1 Normal Operation

During normal operation, Class IV and Class III Power and Instrument Air must be available. Raw Service Water System pumps must be operating. All isolating valves must be fully opened. Three RCW pumps will be operating and one RCW pump will be on STANDBY.

- **Loads Requiring Continuous Control**
  The Moderator HX's, Shutdown HX's, HT Purification HX and F/M Auxiliaries HX's require continuous control of the cooling water flow rate. A control valve in the RCW line adjusts the flow rate automatically, based on a measurement of the D_2O Outlet Temperature.

  RCW to the F/M Oil heat exchanger (where used) is controlled by a self-regulating control valve.

- **Other Loads**
  The Liquid Zone Control Heat Exchanger requires full RCW flow rates at all times, this flow rate is established by a manual globe valve with an adjustable mechanical stop.

  The Boiler Room Coolers, the F/M Room Coolers, the Shutdown Cooling Pumps, the Moderator Pumps' Motors, the HT D_2O Collection Tank heat exchanger and Vent Condenser, the Delayed Neutron Monitor heat exchangers, the Moderator Cover Gas Condenser, the Pressure and Inventory Control heat exchanger and Vent Condenser, the HT Pumps and Gland heat exchanger and the D_2O and H_2O Leaking Detection heat exchanger have full RCW flow rates at all times. This flow rate is established by restricting orifices in the return line.

4.5 Recirculated Cooling Water Overpressure Protection

Protection against overpressurization of all heat exchangers and pumps is provided by means of pressure relief valves. Set pressure is 860 kPa.

5.0 Emergency Water Supply System Description

5.1 Description

A schematic diagram of the Emergency Water Supply (EWS) system is shown in Figure 4. Note that one system provides emergency water to all units of a multi-unit station.

5.1.1 Water Supply

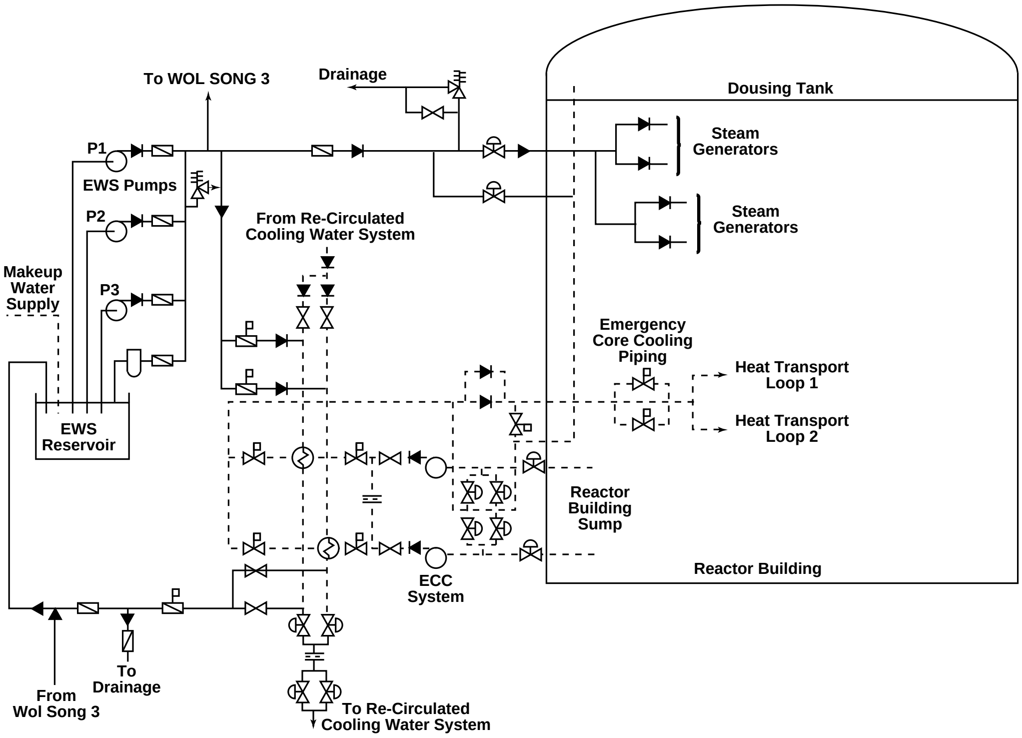
Water from a seismically qualified (DBE) reservoir is used in the operational mode and for testing the rest of the system. The amount of water available for EWS operations in the reservoir is 6.8×10^6 litres.

Three 100% motor driven pumps provide water for distribution to EWS sub-systems inside the reactor building and the service building. The pumps, supplied from the emergency power supply (EPS) system, are located in a separate pumphouse. The pump discharge line is buried and runs to the reactor building wall going through two penetrations to the steam generators and to the heat transport system. A third line from the pump discharge runs to the service building and connects to the recirculated cooling water system supply pipings of the ECC heat exchanger. Isolation valves are provided on each line penetrating containment and on the line to the service building to allow selection of the required EWS sub-system.

For a multi-unit station, two of the three EWS pumps are required to operate to provide adequate EWS flow for the worst case scenario.

Figure 4

Emergency Water Supply System for Multi-Unit CANDU 6



5.1.2 Emergency Water Supply to Steam Generators

The steam generators serve as the long term heat sink for decay heat removal as long as circulation in the heat transport system can be maintained. The main steam generator feedwater pumps are on Class IV power with the standby feedpump (4% of nominal flow) on Class III power. If the power system fails or if a break in the feedwater train occurs, the EWS system provides an independent source of water for the steam generators of up to 30 *l*/s . The normal feedwater system is designed to supply water at the normal steam generator operating pressure. This normal feedwater supply is backed up by two low pressure sources of water via the EWS system. One is the gravity feed system taking water from the dousing tank via EWS valves to the steam generators. The EWS piping connects to the dousing tank via the ECC downcomer. So both the dousing water and ECC water stored in the dousing tank are available for makeup to the steam generators. The other supply is from the EWS reservoir via the EWS pumps to the steam generators. The EWS pumps are started manually by the operator. The EWS valves are automatically opened when the steam generator pressure falls below 345 kPa(g).

After a loss of the normal feedwater to the steam generators, due to a total loss of Class IV and Class III power disabling the three main feedwater pumps and the auxiliary feedwater pump, the EWS reservoir water and the dousing tank water are the remaining sources of feedwater to maintain the steam generator as a heat sink. However, the EWS system is a low pressure system. Therefore, the steam generator auto-depressurization is initiated automatically by opening the main steam relief valves on an abnormally low steam generator level, which indicates an unavailability of normal feedwater. Sufficient number of main steam relief valves operate for depressurization to reduce the steam generator pressure from normal operating pressure to 345 kPa(g) and to maintain the steam generator at a sufficiently low pressure to permit continuous makeup equivalent to the steam generation by a decay power at 30 minutes after reactor shutdown.

5.1.3 Emergency Water Supply to Heat Transport System

In addition to its normal function to provide cooling of the heat transport system after a LOCA, the ECC system is also used to provide make-up to the heat transport for leaks after a seismic event, as follows: At first, the water in the high pressure ECC (HPECC) tanks provides the initial pressurized make-up to the heat transport system following the bursting of the rupture discs. Upon depletion of the HPECC water tanks, the water in the dousing tank provides the make-up to HT system by gravity. Ten of the ECC valves on Class II power or Class I power can be powered by the EPS system to ensure that the ECC piping path is available. Since the entire low pressure ECC (LPECC) circuit is seismically qualified to DBE Category B, when the dousing tank becomes depleted one of ECC recovery pumps (606 *l*/s,) is used to recirculate the water accumulated in the reactor building back into the dousing tank via LPECC circuit until the dousing tank is refilled. The water in the dousing tank can then provide make-up to the heat transport circuit by gravity.

Because of these reasons, the water from the EWS pumps is only required for heat transport system make-up if makeup is required and the HP ECC tanks are depleted, and:

- The dousing water was depleted for other purposes, such as a much larger than expected flow to the steam generators immediately following a DBE (prior to start-up EWS pumps), or
- The LPECC circuit is incapacitated, preventing recycling of the dousing water.

5.1.4 Emergency Water Supply to ECC Heat Exchanger

The EWS system backs up the RCW system to ensure the reliability of a supply of cooling water to the ECC heat exchanger during long term ECC operation. Long term ECC is required for a period of up to three months after a LOCA. A flow of more than 104.6 *l*/s from the EWS is supplied to the ECC heat exchanger for decay heat removal.

The EWS flow is recirculated back to the reservoir while the reservoir water is cooled by heat transfer from surface if no external make-up is available since the make-up line is not seismically qualified. Cold water can also be brought in from off-site for reservoir make-up to reduce the reservoir temperature.

5.2 Emergency Water Supply (EWS) System Major Components

Pumps

Three 100% vertical type sump pumps rated at 114 *l*/s, 79 m head, are provided to supply water to the EWS sub-systems in the reactor building and the service building. They are motor-driven and supplied from the EPS system. Temperature detectors are incorporated to provide thermal protection for mechanical components. For operation at 30 *l*/s the pump discharge is manually throttled to run back on the head flow curve, which has sufficient slope to provide reasonable control. Flow indication is by an elbow tap flow element located outside the reactor building.

5.3 EWS Systems Controls

Flow is measured in the main stream generator line and in the ECC heat exchanger line by elbow taps. These indicate the presence of flow and process demands. Pressure indication and flow measurement by an elbow tap are provided for testing the pumps.

5.4 Emergency Water Supply System Operation

With the exception of the steam generator makeup water sub-system, the emergency water supply system is not required immediately following an event. As a result, most operations are manual and are performed by the operator from the secondary control area or the emergency water supply building.

Emergency water supply system operations consist of manually starting the pumps and then operating the handswitches to open the appropriate motorized or pneumatic isolating valves to supply water to the required loads. These isolating valves also have hand wheel actuators so that they can be operated locally. The steam generator makeup water sub-system operates automatically when the steam generator pressure falls below 345 kPa(g). The steam generators are thus kept supplied with water for the first 30 minutes until the emergency water supply system becomes available.

Following a design basis earthquake or loss of Class III and IV electrical power, the emergency water system will supply water to the heat transport loops and the steam generators for long-term operation. This will ensure that thermosyphoning can continue and that decay heat can be transferred to the steam generators and then discharged.

Following a LOCA and failure of the RCW system (eg. DBE or loss of Class III and IV power) EWS water is supplied to the ECC heat exchangers by remote manual operation of the appropriate EWS and RCW valves.

5.5 EWS System Overpressure Protection

The overpressure protection for the EWS system is provided by the pressure relief valve and the normally opened vent valve.

The section of the piping between isolation valves is normally filled with light water and depressurized. This piping is protected by a spring actuated relief valve which is set at a pressure of 0.93 MPa(g). The relief capacity is 0.13 *l*/s.

A normally open vent valve prevents pressure buildup in the piping due to leakage from the steam generators via the check valves. Leakage via the check valves will pressurize the normally liquid filled EWS piping from the steam generators up to the normally closed valve. If the vent valve is closed (for testing or for EWS operation), spurious operation will not lead to an upstream pressure buildup since the pressure is relieved via a spring actuated relief valve which is set at a pressure of 1.03 MPa. The relief capacity is 3175 *l*/s .

6.0 System Configuration in Various CANDU Stations

The water systems concepts of Bruce B, Darlington, and CANDU 6 are necessarily different owing to the differences in the geographical nature of their site locations. Bruce B and Darlington are situated on lakeside sites and subjected to a relatively cool climate, whereas the CANDU 6 reference stations are situated on seashore sites in a relatively warm climate. Tables 1, 2 and 3 show the major similarities and differences between the stations being compared for the RSW, RCW, and EWS systems respectively.

Table 1

Raw Service Water Systems Comparison

	Bruce B (4x850) 4 unit station	Darlington (4x850) 4 unit station	CANDU 6 1 or 2 unit station
System	Low Pressure Service Water This system is an open loop lake water system provided for each unit.		Raw Service Water (RSW) This system is an open loop seawater system.
Major Users	- Moderator Heat exchangers - Shield Cooling - Vault coolers		- RCW heat exchangers - Chiller condensers - Irradiated fuel bay coolers
	- Emergency Coolant Injection Heat exchangers - Shutdown Cooling heat exchangers - High pressure service water pumps - miscellaneous	- Powerhouse upper level service water pumps - Turbine oil coolant - Hydrogen coolers - RCW heat exchangers	
No. of pumps/unit	4 x 33 $^{1}/_{3}$%	4 x 33 $^{1}/_{3}$%	4 x 33 $^{1}/_{3}$%
flow rate and head/unit	5.68 m^3/s @ 44.8 m (3 pumps)	6.33 m^3/s @ 48.15 m (3 pumps)	6.82 m^3/s @ 27.5 m (3 pumps)
System	High Pressure Recirculation System (open loop) The system supplies loads at elevations too high to be supplied by the Low Pressure open system and where potential freezing of heavy water would be a problem. The system is open loop and draws water from the Low Pressure Open System when RCW temperature 16°C.	Powerhouse Upper Level Service Water (open loop)	
Major Users	- Heat Transport main circuit pumps - Liquid Zone control heat exchangers - Moderator purification heat exchangers - Vapour Recovery heat exchangers		
	- Demineralized service water intercoolers	- Shutdown cooling heat exchangers	
No. of pumps/unit	3 x 50%		
flow rate per pump	258.7 *l*/s	500 *l*/s	

Table 2

Recirculated Cooling Water System Comparison

	Bruce B (4x850) 4 unit station	Darlington (4x850) 4 unit station	CANDU 6 1 or 2 unit station
System	<u>Closed Loop Demineralized Service Water System</u> A closed loop system supplying demineralized water to users listed below. The heat load is removed via two heat exchangers by lake water taken from the High Pressure Recirculation System.	<u>Recirculated Cooling Water System (RCW)</u> Heat loads not directly cooled by raw service water are cooled indirectly by raw service water by means of a closed loop Recirculated Demineralized Cooling Water system. RCW system supplies water to equipment which may be subject to corrosion and protects vital D_2O equipment from untreated water or raw (salt) water.	
Major Users	- Bleed cooler - Gland seal cooler - Delayed neutron monitoring system	- Heat Transport Pumps - D_2O collection tank cooler and vent condenser - D_2O sampling coolers	
	- heat transport pump jacket - heat transport pump cavity concrete cooler - LSD Speed Sensors	- Heat Transport pump and shutdown cooling pumps gland supply coolers - D_2O in H_2O sample cooling - Fuelling service area rehearsal facility equipment and vapour recovery driers - Filter and ion exchange columns recirculation - Heat Transport and shutdown cooling pumps glands supply emergency cooler - Bleed coolers	- HTS purification cooler - Pressure and inventory control cooler - Pressure and inventory control vent condenser - Fuelling machine auxiliaries and heat exchangers - Moderator System heat exchangers - Heat Transport pump gland cooler - Moderator purification cooler - End Shield cooling heat exchangers - D_2O Upgrading - Shutdown cooling system heat exchangers - Class III diesel generators - T/G Auxiliaries
No. of pumps/unit	x 2 100%	3 x 50%	4 x 33 $^1/_3$%
flow and head per pump	158 *l*/s @ 45.7 m	170 *l*/s @ 45.7 m	1535 *l*/s @ 48 m

Table 3

Emergency Water Supply System Comparison

	Bruce B (4 x 850) 4 unit station	Darlington (4 x 850) 4 unit station	CANDU 6 1 or 2 unit station
System	Emergency Water System Emergency water is supplied by three pumps (504.7 *l*/s at 80.8 m of water) and powered by the emergency power system both located in the emergency water and power supply building. The system can supply all four units, each unit having a manually operated valve station located next to the secondary control area. The system is seismically qualified.	Emergency Service Water System The emergency service water (ESW) system is independent and physically separated from the normal water systems and provides cooling water to essential safety related loads in the event of loss of normal service water. One system supplies all four units. The ESW system has four pumps (900 *l*/s at a head of 82 m of water) which are capable of supplying all four units in the most severe load case. The system is seismically qualified to withstand a design basis earthquake and supplied with seismically qualified on-site standby power, i.e., the emergency power system.	Emergency Water Supply system (EWS) This system provides an alternative supply of water to essential systems in the event of loss of normal water supplies. The system is operated by manually starting the pumps (2x100% for a 1 unit station or 3 x 100% for a 2 unit station) which supply water from the EWS reservoir through a common supply header to the emergency water supply subsystems in the unit service and reactor buildings.
Major Users	The emergency water system supplies the following: - Emergency water to the steam generators for decay heat removal. - Emergency makeup to the heat transport system in the event of minor leakage in the heat transport system. - Cooling water to the emergency coolant injection recirculation heat exchangers to provide seismically qualified long-term emergency core cooling capability. - Cooling water to containment area coolers to condense any steam in containment which could result from heat transport system breaks. - Cooling water to primary and secondary irradiated fuel bays. - Cooling water to the air conditioning unit in the secondary control area.		- Emergency water to steam generators - Emergency water to ECC heat exchangers - Emergency makeup to heat transport system

7.0 Abnormal Operation

7.1 Failure of Class IV Power and Control of Class III Loads

All raw water pumps are connected to the Class III Power Bus. Two pumps in each sub system are connected to the “odd” bus and the others to the “even” bus, with relays arranged to permit only one pump from each bus to run on loss of Class IV power. In the event of a Class IV failure, non-essential equipment is isolated from the system, and two pumps start and deliver flow to essential equipment. Flow to this essential equipment is adequate even if one of the two pumps fails.

On loss of Class IV Power, the process water supplies are interrupted. With Class III Power, only a limited amount of cooling water is available thus necessitating a load reduction to ensure sufficient supplies for essential loads.

Flow to the moderator heat exchangers is reduced by a computer controlled signal to the main control valves to close linearly in two minutes. The by-pass line control valves are left on control from the computer.

The Heat Transport Purification heat exchanger control valve and the Pressure and Inventory heat exchanger motorized valve close completely. A by-pass line around each of these valves is sized to pass the minimum required flow, once pumping power has been restored by the Class III standby generators.

The motorized valve in the Pressure & Inventory Control Vent Condenser line closes completely.

The motorized valves in the Heat Transport Pump lines close completely. This eliminates coolant flow to the pump bearings and air coolers but maintains cooling of the gland seal coolers.

All other loads require equal Class IV and Class III flow rates.

7.2 Reactor Trip

During the first half-hour of the shutdown conditions, all the flows are identical to those during normal operation with Class IV Power. After this period, the flow to the Heat Transport heat exchanger is reduced in order to meet the Shutdown Cooling demand.

7.3 Loss of Coolant Accident

During the first 15 minutes after a Loss of Coolant Accident, the control valves on the lines to the Moderator HX's will open fully. After this period, the Control Valves on the HT Purification heat exchanger line, closes completely. The control valves to the Moderator HX's stay fully open.

7.4 Loss of Instrument Air

All control valves in the system will open fully if the supply of instrument air fails.

8.0 Interdependencies With Other Systems

8.1 RSW System Interfaces

8.1.1 The Pumphouse Building Structure

Adequate discharge tunnels are provided to dispose of large leaks originating from a pipe break inside the structure. These tunnels lead to the outside and are generally fitted at the end with a non-return flap closure to maintain the building flood proof from an external environmental condition.

8.1.2 Instrument Air System

Instrument air is supplied to all pneumatically operated valves in the system.

8.1.3 Recirculated Service Water System

The RSW System interface with the RCW system at the heat exchangers located in the turbine building.

8.1.4 Electrical Power Supply System

Motorized butterfly valves are powered from Class III power. During failure of CL IV power only one of the main pump motors is connected to Class III power on each bus. The remaining motors are supplied from Class IV power.

8.2 RCW System Interfaces

8.2.1 Building Structure

Adequate tunnels are provided to accommodate leaks from the RCW or discharge due to pipe failure.

8.2.2 Process Systems and Equipment

All users provide adequate piping connections (nozzles, flanges, and gaskets) at their equipment to receive the required amount of recirculating cooling water.

Each user is supplied with a pair of isolation valves on the RCW side for maintenance purposes.

8.2.3 Instrument Air System

Instrument air is supplied to all pneumatically operated valves in the system at the required pressure.

8.2.4 Raw Service Water RSW

The RCW System interfaces with the RSW System upstream and downstream of the ECC system heat exchanger.

The two systems also interface outside containment, upstream and downstream of the Shutdown Cooling system heat exchangers.

Isolation valves are normally open and both ECC and SDC heat exchangers are normally filled with RCW (when ECC or SDC are operational).

8.2.5 D_2O in H_2O Leak Detection System

The D_2O in H_2O Leak Detection System uses an infra-red spectrometer to monitor certain heat exchangers on the RCW side for heavy water leaks. These heat exchangers are:

- Main Moderator Heat Exchangers
- Shutdown Cooling Heat Exchangers
- Heat Transport Pump Seal Cooling Heat Exchangers

The normal concentration of D_2O in H_2O in the RCW System is 148 ppm.

8.2.6 Electrical Power Supply System

All control valves are pneumatically operated. Solenoid-operated valves and motorized butterfly valves are powered from Class III power. The supply pump motors are also connected to Class III power.

8.3 EWS System Interfaces

8.3.1 Building Structure

In the pumphouse adequate facilities are provided to dispose of leaks due to pipe breaks and vice versa to prevent flood water from entering the building.

8.3.2 Process Systems

The system interfaces with the steam generators. Alternatively, the S/G's can be connected to the ECC injection line while preparing the EWS pumps to start.

The EWS is also connected to the Heat Transport System via the ECC system. This path avoids the need for an additional D_2O/H_2O interface which remains at the rupture discs, one on each end of the reactor in the ECC system where the separation between H_2O and D_2O piping is located.

The dousing water is also connected to the EWS piping via the ECC downcomer.

The EWS system is connected to the ECC heat exchangers. Segregation is achieved during normal operation by means of a check valve in the connection line to the RCW system. It should be noted that the EWS system interfaces with the Recirculating Cooling Water System by means of a motorized valve which, when open, directs flow from the EWS pump into the RCW side of the ECC heat exchanger.

8.3.3 Other Interfaces of the EWS

The EWS system also interfaces with the instrument air system, the emergency power supply system to power its motorized valves, motor pumps and air operated equipment.

9.0 Potential for Radioactive Release and Radiation Hazard to Operator

No appreciable activity from corrosion and fission products or tritium expected from these systems.

Table 4

Raw Service Water Major Load Requirements

	Class IV ***l*/s**	**Class III** ***l*/s**
Recirculated water cooler heat exchangers	5227.0	1734.0
Standby generator coolers	176.5	169.0
Irradiated fuel bay cooling	125.0	0.0
Chillers	273.0	136.0
Strainer backflushing	98.5	98.5
Turbine generator lube oil coolers	169.5	0.0
Vacuum pumps	134.0	0.0

Table 5

Recirculated Cooling Water Major Load Requirements

	Class IV ***l*/s**	**Class III** ***l*/s**
Moderator heat exchangers	2488.00	886.0
Moderator purification heat exchangers	41.0	0.0
Heat transport pumps	49.3	12.1
Pressure and inventory control degasser cooler	121.0	5.3
Heat transport purification cooler	417.0	15.2
Shutdown coolers	0.0	606.0
Shield cooling heat exchangers	174.0	98.5
Liquid zone control heat exchangers	75.8	75.8
Emergency core cooling heat exchanger	758	656.0
Turbine generator hydrogen coolers	158.3	0.0
Turbine generator stator coolant coolers	126.0	0.0
Reactor building local air coolers (containment cooling)	222.0	222.0