Heat Transport System

Training Objectives

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

- Outline the functions or design features which the heat transport system must provide;
- Sketch one loop of a heat transport system including pressure tubes, feeders, headers, pumps and steam generators;
- Describe the difference between one and two loop systems and a reason for the latter;
- Explain how the two systems are designed to compensate for the fact that the inner channels are at higher power than the outer ones;
- Given a sketch of a steam generator, name the various parts or sections and explain their functions;
- Explain the advantage of the presence of vapour in the heavy water coolant as it enters the steam generator;
- Given a sketch of the heat transport system, name the various parts or sections and outline their functions;
- Given a diagram of the heat transport system, sketch in the following major, auxiliary systems. Outline the functions and operation of these systems:
  - the pressure and inventory control system,
  - the D_2O collection system,
  - the shutdown cooling system,
  - the purification system;

[NOTE: These systems are covered in more detail in other lessons therefore the coverage here is not comprehensive.]

- State the functions of the feedwater and steam system;
- Explain what control systems are used on the heat transport system and what provisions are made for protection against overpressure;
- Give a general description of cooldown and the reasons for the progression of events;
- Explain what 'solid' mode is and what are the problems which could occur while operating in this mode;

- Explain the capability of the system to cope with certain abnormal occurrences including the following:
  - failed-open, liquid, relief valve,
  - failed, variable, pressurizer heater,
  - failed pressurizer
  - steam generator tube leakage,
  - HT purification system isolated,
  - HT pump seals failed,
  - operation with one or two HT pumps blocked,
- Explain the capability of the system to cope with a loss of coolant accident. State what functions must be provided;
- State the radiation hazards, explain how they are created and describe generally the radiation protection program.

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1. Function

The heat transport system circulates pressurized coolant (D_2O) through the reactor fuel channels to remove the heat produced by fission and fission product decay heat in the uranium fuel. This heat is carried by the coolant to the steam generators. Figure 1 shows a schematic diagram of the system.

Functional Requirements

The major functional requirements are either safety related or process related. They are summarized below.

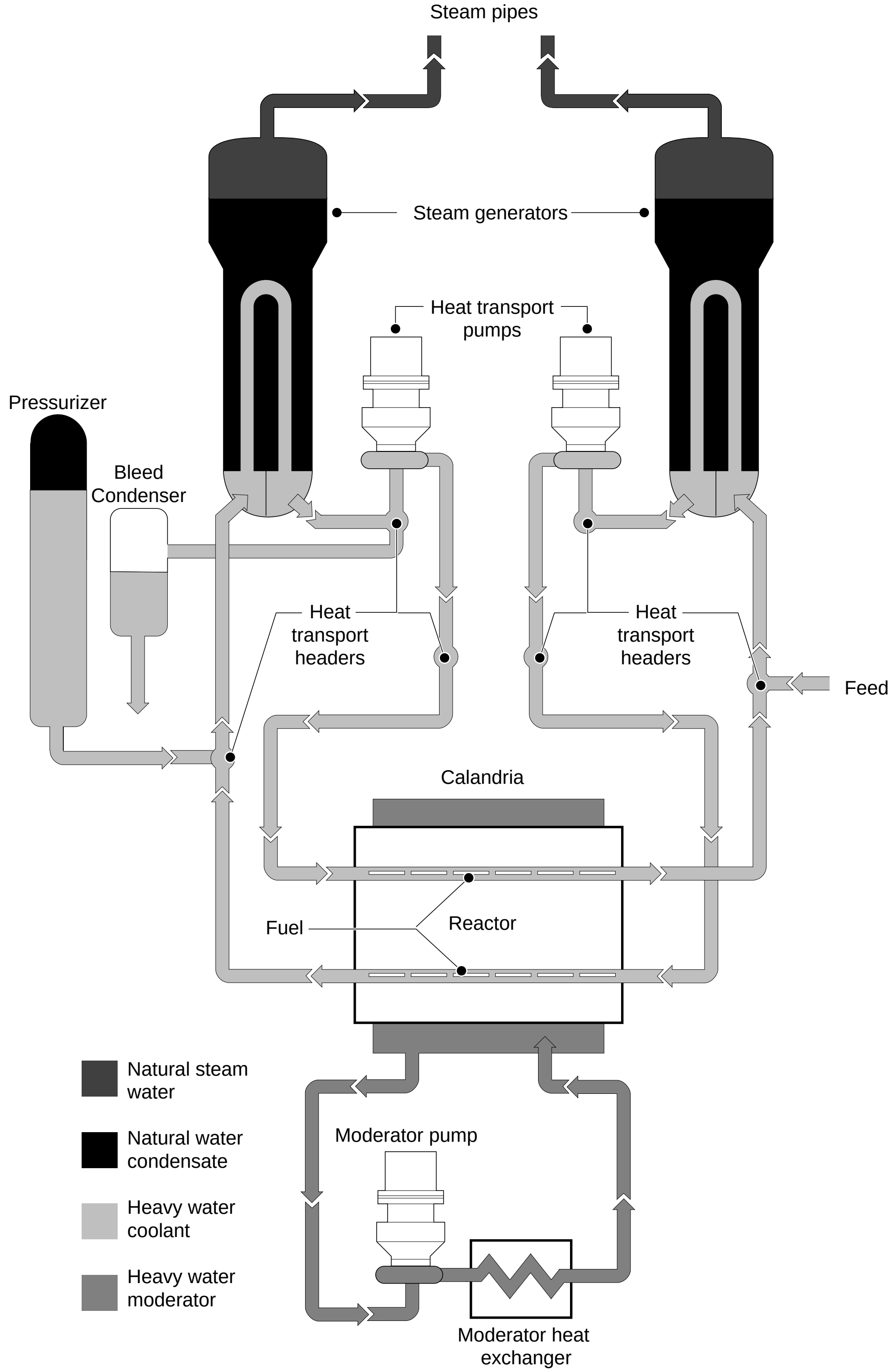
Safety Related

- Must be able to contain the effects of a postulated loss of coolant accident (LOCA) within the capability of the safety systems and to provide a path for emergency coolant flow to the fuel under the same circumstances.
- Provide process measurements which would be used to trip and shut down the reactor to maintain the system pressure within design limits.
- Maintain the integrity of the reactor coolant pressure boundary in the event of a design basis earthquake.
- Provide for the removal of decay heat by natural circulation during a total loss of pumping power condition.

Process Related

- Must be able to transport the fission heat from the fuel to the steam generators.
- At all times during reactor operation the system must provide adequate cooling of the fuel and during shutdown provide a means for removing the decay heat.
- Protect the system and components from overpressure.
- Contain the D_2O inventory with the minimum possibility of either leaking or downgrading.
- Provide containment for failed fuel during normal operating conditions.

Figure 1
CANDU Reactor Simplified Flow Diagram



2. System Description

Referring to figure 1 it can be seen that the simple system is a continuous loop. Starting from the outlet of one pump the heavy water coolant passes to the inlet header and from there through the feeder pipes to the fuel channels (pressure tubes). It is heated by the fuel and passes from the fuel channels through feeder pipes to the outlet header and from there to a steam generator. After transferring heat to the steam generator it goes to the inlet of the heat transport pump and is pumped to another inlet header, through feeder pipes and the fuel channels to the outlet header on the other side. Finally it goes to another steam generator back to the inlet of the pump from which it started. In half of the fuel channels the coolant flows in one direction and in the other half in the reverse direction.

Some reactors, such as those in the Bruce Stations, use only one loop. Those at Pickering, Darlington and the CANDU 6 use two loops

The coolant used is heavy water because of its low neutron absorption characteristic. The coolant is pressurized to achieve high temperatures with a minimum of boiling. This in turn provides for higher steam side pressures and temperatures and increased unit efficiency. There is a down side to the pressurization since it requires pressure tubes with thicker walls creating a fuel burnup penalty.

Operating experience has led to changes in successive generations of CANDU reactors. Although the basic configuration has not altered, several significant changes have occurred. These include the elimination of HT pump redundancy, the increase in size of the pumps and steam generators, the minimization of non-welded joints, the reduction of the total number of valves and, as much as possible, the substitution of bellows sealed valves for packed gland valves. These latter changes to joints and valves have been made to reduce D_2O losses and radioactive releases. Table 1 gives a good picture of the evolution which has occurred.

Table 1

Evolution of Heat Transport Systems

	Pick A	Bruce B	Wolsong-1	CANDU 3/6	Darlington
Nominal Net Unit Output (MWe)	515	860	638	450	881
Heat Transport System Features					
Number of Loops	2	1	2	1	2
Number of Fuel Channels	390	480	380	380	480
Length of Fuel Channels (core region, m)	6	6	6	6	6
Dia.. of Fuel Channels (mm)	103	103	103	103	102.5
No. of Steam Generators	12	8	4	4	4
No. of HT Pumps					
Total	16	4	4	4	4
Operating	12	4	4	4	4
No. of Non-Welded Joints (approx.)	1000	250	200	4120	-
No. of Valves					
Packed Gland	170	75	90	70	-
Bellows Sealed	570	500	300	200	-

Note:
- indicates that the systems are undergoing engineering changes

3. CANDU 6 System

The schematic diagram of the CANDU 6 system is shown in Figure 2.

The CANDU 6 reactor has 380 fuel channels arranged in a square array within the calandria.

The heat transport system is arranged in two loops each of which is connected to a half of the pressure tubes, on one or other sides of the vertical centre line of the reactor. The two loops provide bi-directional flow so that the flow is in opposite directions in adjacent channels. Having two loops reduces the rate of positive reactivity insertion in the event of a void in the coolant caused by a loss of coolant accident. Each loop has inlet and outlet headers at each end of the reactor.

Because the neutron flux and therefore the fuel channel power output is not uniformly distributed across the reactor core some channels require more cooling than others. If the coolant temperature and state is to be the same when it arrives at the outlet header then the flow through the channels will need to be greater in the higher power ones. To set the flow in each channel the feeder pipes vary in diameter from 3.8 cm to 8.9 cm. This is to achieve at the outlet headers at 100% full power coolant at a temperature of 310 °C with 4% steam. The reactor regulating system also controls the flux and power distribution to produce the same result.

It can be seen from Figure 2 that there are two pipes connecting the outlet header to the steam generator and a single pipe connects it to the pump suction. The pump delivers the coolant to the inlet header through two pipes. From there it is distributed to the channels through the feeder pipes.

As the HT coolant passes through the steam generator tubes it remains at the same temperature until the small quantity (4%) of steam is condensed and then it is slowly cooled. The condensation of the vapour provides a higher temperature differential along a greater length of the steam generator tubes than if the coolant entered as a saturated liquid only. Figure 3 shows this. On the steam side the conditions at exit from the steam generator 4.69 MPa and 260 °C. Table 2 sets out the heat transport parameters for normal operation at 100% of rated power.

Figure 2

CANDU 6, Heat Transport System

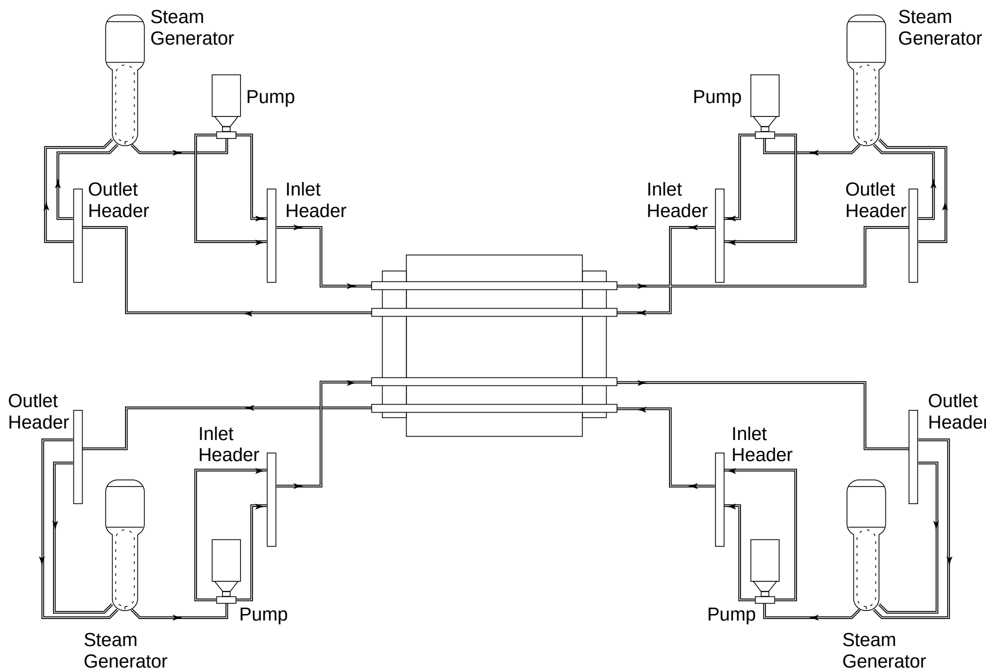


Figure 3
CANDU Steam Generator Heat Load Diagram

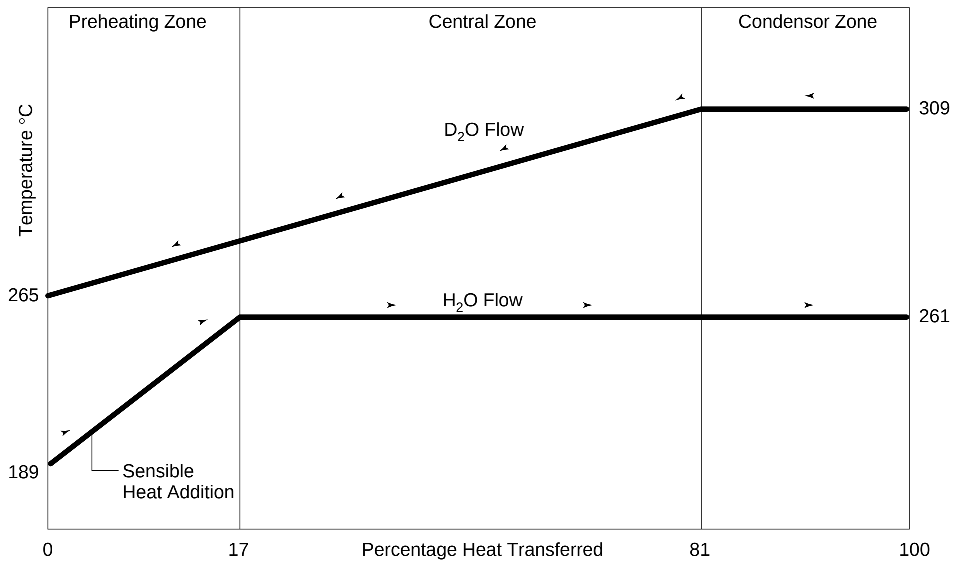


Table 2
CANDU 6 Heat Transport System Parameters, 100% Full Power

Coolant	**D_2O**
Maximum Channel Flow	24 kg/s
Total Core Flow	7.7 Mg/s
Maximum Channel Power	6.5 Mw
Heat Balance	
Heat from fuel to coolant	2056 Mw
Heat from HT pumps	17 Mw
Circuit Losses	9 Mw
Heat to SGs	2064 Mw
Reactor Inlet Header	
Pressure	11.25 MPa(a)
Temperature	266 °C
Reactor Outlet Header	
Pressure	10 MPa(a)
Temperature	310 °C
Quality	4%

4. Bruce NGS 'A' System

The Bruce reactor has a single loop heat transport system which is shown in a line diagram in Figure 4. It has four circulating pumps, eight steam generators and four external pre-heaters unlike the CANDU 6, Darlington and Pickering B stations.

The 280 channels of the inner zone produce more power than the 200 outer zone channels. The higher power is removed by sub-cooling the D_2O which enters this inner zone by passing it through the steam generator pre-heaters. These pre-heaters have relatively cool feed-water on the shell side. As can be seen in Figure 4, the effective Δt for the heat transport water is 39°C in the outer zone channels and 53°C in the inner zone channels. All channels operate with approximately the same outlet temperature of 304°C and are connected to a single outlet header on each side of the reactor. Coolant to the outer channels proceeds directly from the pumps through the inlet header and feeders. The outlet header saturated pressure is 9.18 MPa(a).

Major Components

This section gives a brief description of the major components of the heat transport system. All of the examples are from the CANDU 6 reactor. Included are the steam generator, the circulating pumps, feeders, headers and valves.

Steam Generators

Steam generators (SG) transfer heat from the heavy water HT coolant on the SG primary side to light water on the secondary side, causing it to boil and produce steam which drives the turbine. Each SG is identical and consists of an inverted vertical U-tube bundle in a cylindrical shell. Steam separator equipment is provided in the upper end of the shell. A typical CANDU 6 SG is shown in Figure 5.

The primary side of the SGs consist of the SG head or channel cover, the primary side of the tube-sheet and the tube bundle. A divider plate separates the inlet half of the SG head from the outlet half. The Incoloy U-tubes are welded to the primary side of the Inconel clad carbon steel tube-sheet and rolled into the tube-sheet. The carbon steel SG head is provided with two man-ways for entrance for maintenance, e.g. plugging of leaking tubes.

D_2O HT coolant enters the SG head at a quality ranging up to 4.4% steam by weight at full power. As the flow passes through the tubes, the D_2O steam is condensed and cooled.

Figure 4
Heat Transport System (Bruce)

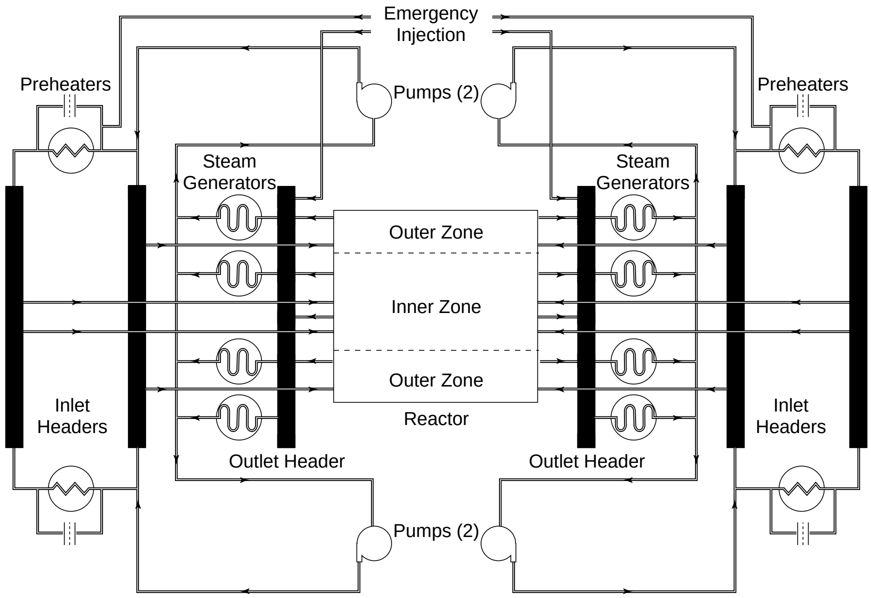
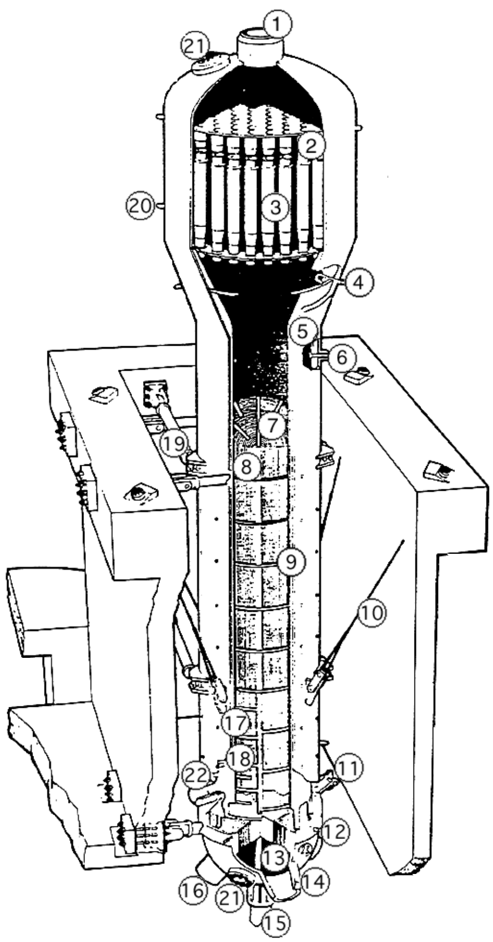


Figure 5
CANDU 6 Steam Generator



1 Steam Outlet Nozzle
2 Secondary Steam Cyclones
3 Primary Steam Cyclones
4 Chemical Feed Nozzle & Headers
5 Downcomer Annulus
6 Reheater Drains Return & Emergency Water Supply Nozzle
7 U-Bend Supports
8 Tube Bundle
9 Tube Support Plate
10 Back-up Supports
11 Observation Port
12 Blowdown Nozzle
13 Divider Plate
14 D_2O Inlet Nozzle
15 Base Support
16 D_2O Outlet Nozzle
17 Baffle Plate
18 Preheater
19 Lateral Supports
20 Water Level Control Taps
21 Manway
22 Feedwater Nozzle

The secondary side of the SGs consist of the shell, the steam separating equipment, the tube bundle shroud, the secondary side of the tube sheet, the secondary side of the tube bundle, the preheater baffles and the tube support plates. Carbon steel is the principal material of construction. The shell is provided with a man-way and connections for addition of feedwater and for SG water blowdown.

Feedwater enters the preheater section and the baffles force it to flow over the D_2O outlet end of the U-tube bundle. Figure 6 shows a cross-section of a Pickering steam generator and illustrates the preheater section well. The pre-heaters at Pickering are an integral part of the steam generators unlike the Bruce stations which have separate pre-heaters. Feedwater from the preheater section at saturation temperature mixes with the recirculating saturated water flowing over the other section of the U-tube bundle.

The steam-water mixture rising from the upper end of the U-tube bundle passes through the steam separators. The steam with less than 0.25% moisture by weight leaves the boiler through the outlet nozzle. The saturated water from the steam separating equipment recirculates to the bottom of the tube bundle through an annulus (downcomers) between the shroud and the shell. The water passes through holes in the bottom of the shroud and penetrates and flows over the D_2O inlet half of the U-tube bundle.

The water level in the SG is controlled by a combination of level measurement, steam flow measurement and feedwater flow measurement. The level setpoint is programmed to increase with increasing power.

The secondary side thermal/hydraulics performance of the tube bundle is such that tubing survival is ensured. The general objective is a design as tolerant to the realities of plant chemistry as possible. A major problem with steam generators has been chemical attack of the steam generator nickel alloy tubing. To minimize the risk of chemical attack, the steam generator internals meet the following performance requirements:

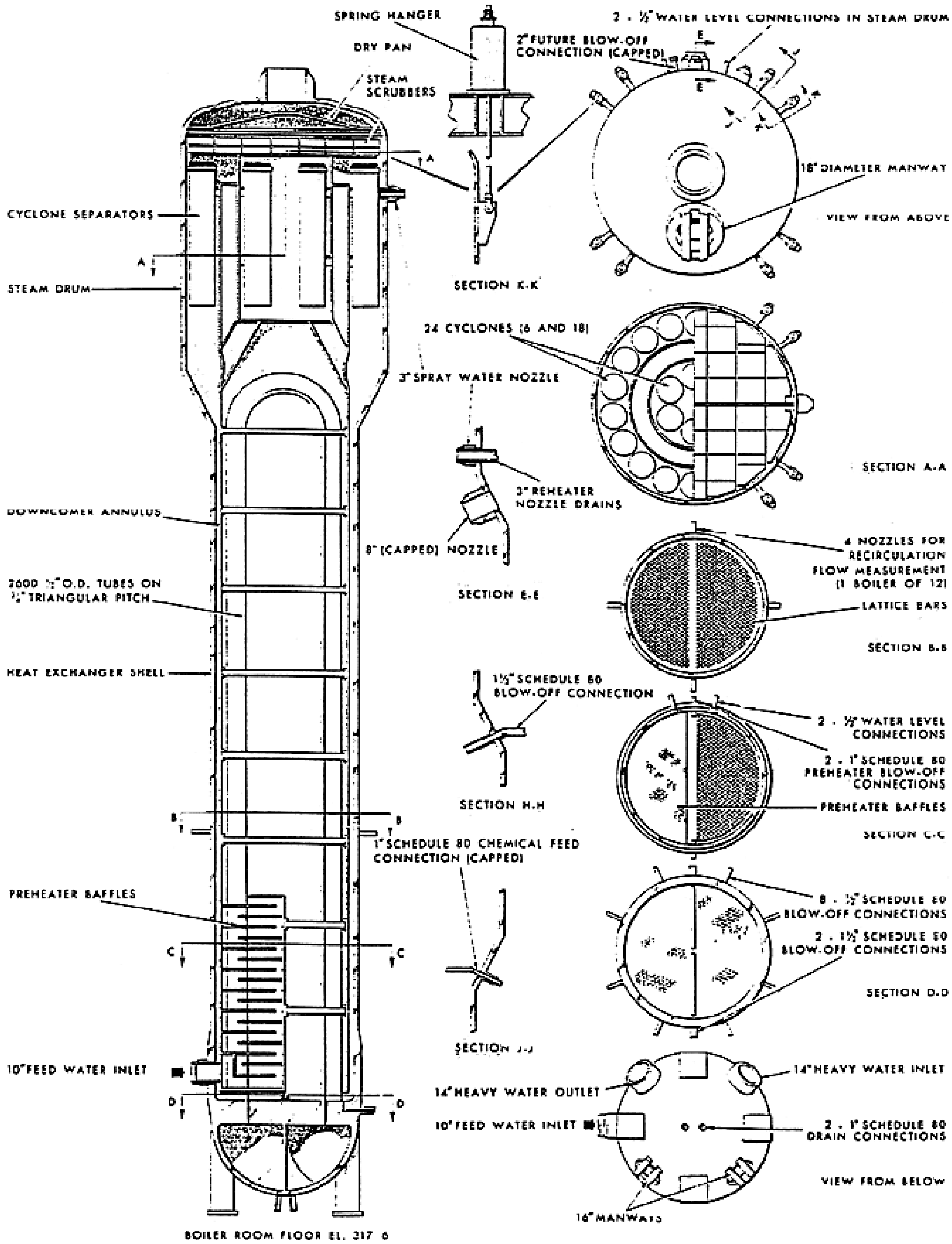
a) The steam generator is designed to obtain a recirculation ratio, at 100% full power and with the water level in the drum at the Normal High Water Level, of not less than 5:1. The recirculation ratio in the steam generator is defined as the total mass flow rate of downcomer flow divided by the steam mass flow rate.
b) The steam generator is designed to minimize the accumulation of solids in the steam generator and in particular in the tube-sheet region. In this regard, the steam generators incorporate features to minimize flow stagnation above the tube-sheet and maximize the sweeping velocity of downcomer water across the tube-sheet.
c) Tube supports are designed to minimize the possibility of concentration of any chemicals or solids present in the secondary side water and to avoid the

potential of tube denting. In this regard, the tube support design provides the greatest possible open flow while still maintaining its function of preventing tube vibration and fretting.

The steam generator internals in the preheater region is designed to withstand low temperature feedwater conditions during the various transient operating conditions.

Corrosion of the steam generator secondary side is controlled through feedwater chemistry by adding chemicals to the secondary water prior to entry into the steam generator and by continuous blowdown of water from the steam generator secondary side. The feedwater chemistry is controlled to limit dissolved solids and the feedwater is treated with hydrazine to reduce the oxygen concentration and with amine to raise the pH. Continuous blowdown is used to remove the precipitated solids from the secondary side, and to control the concentration of dissolved solids in the water in the steam generator secondary side.

Figure 6
Pickering 'A' Steam Generator



The blowdown intake header is provided close to the tube-sheet. Suction from these headers is, if possible, taken from the regions of lowest velocity, where solid particles are most likely to deposit. Blowdown from the preheater has also to be provided. The steam generator is presently required to have a continuous blowdown capability of 0.25 kg/s. However, the maximum blowdown capacity of the steam generator has been increased to 1.5 kg/s.

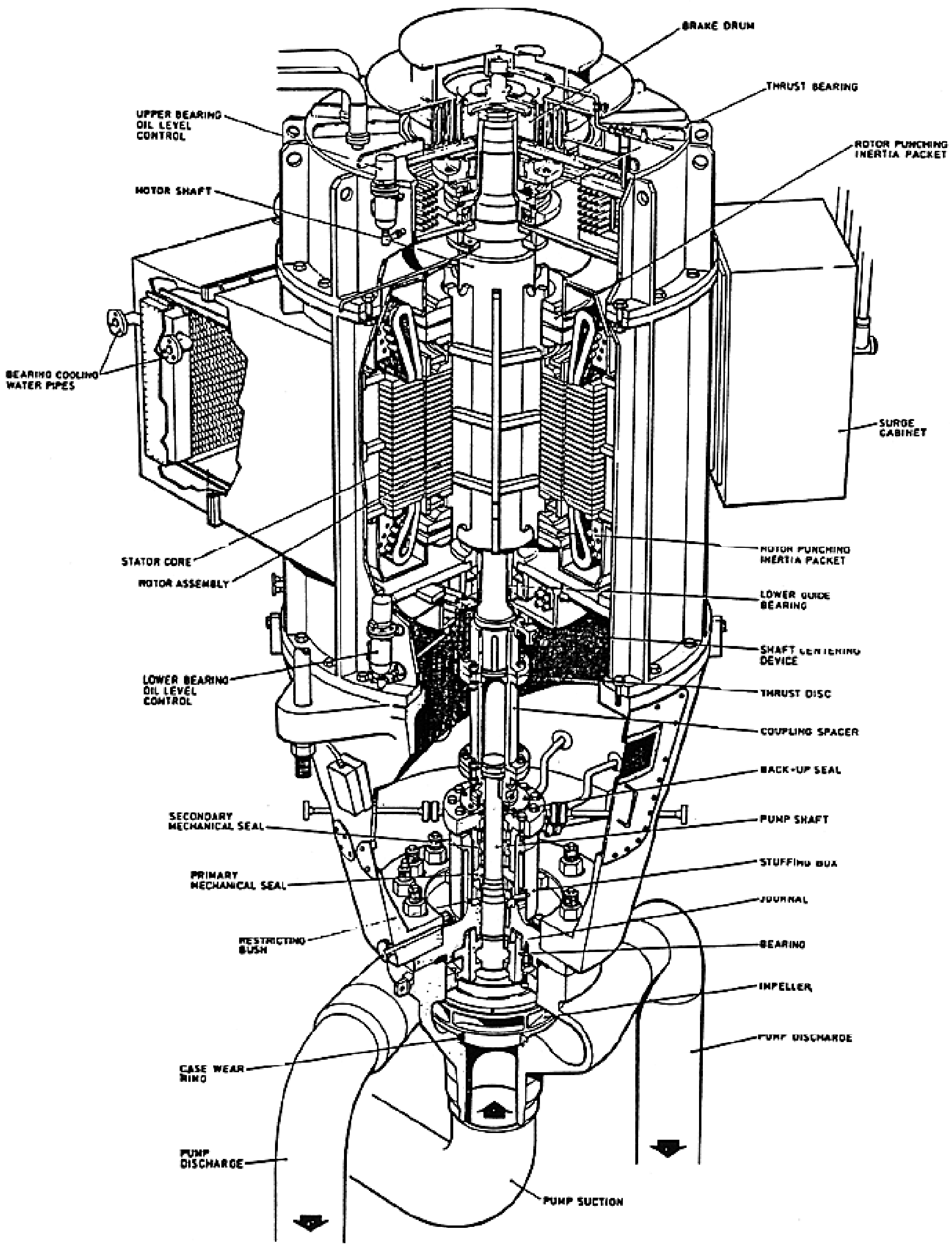
The blowdown system is also capable of providing a means of (a) draining the secondary side of the steam generator for maintenance, and (b) introducing or withdrawing chemicals in the event that chemical cleaning of the secondary side of the steam generator is required.

Heat Transport Pumps

The heat transport pumps in CANDU plants are typically vertical, single stage, single suction, single or double discharge centrifugal pumps. A typical CANDU 6 pump is shown in Figure 7. There are no suction, discharge or check valves in the circuit for these pumps.

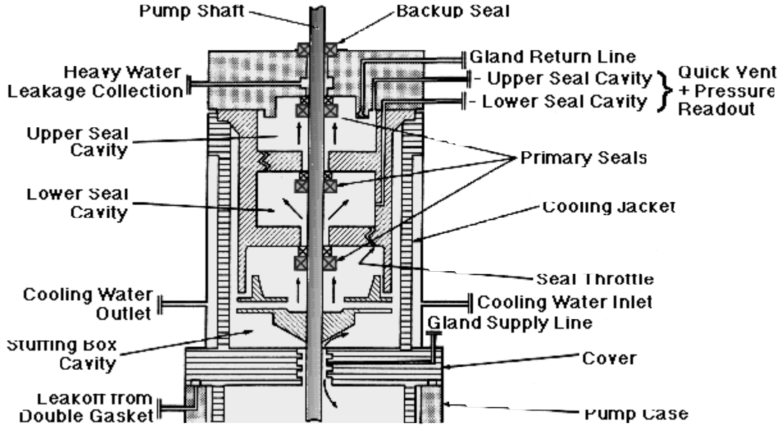
Each pump is driven by a vertical, totally enclosed, air-water cooled, squirrel cage induction motor. The pump is supported by the motor mount, the bottom flange of which is bolted to the top of the pump case. The pump gland is retained between the pump case and the bottom flange of the motor stand. A removable shaft coupling connects the motor to the pump. Removal of the coupling allows sufficient space for the pump seals and bearing to be removed without removing the motor.

Figure 7
Point Lepreau Primary Heat Transport Pump



The pressure boundary includes the pump case, the gland and the bottom flange of the motor mount. The interface between the gland and the pump case is sealed by a double gasket with leak-off provision. The case consists of a vertical bottom suction nozzle, the main bowl and two horizontal discharge nozzles.

Within the gland are a heavy water lubricated hydrodynamic bearing, an auxiliary impeller, shaft seals and the shaft. The main impeller is located at the end of the shaft below the bearing.



The shaft sealing arrangement consists of three mechanical seals and one back-up seal in series. Each mechanical seal is capable of holding full differential pressure with low leakage rates. A leakage collection cavity located above the third mechanical seal drains seal leakage to the heat transport D2O leakage collection system. The back-up seal, located above the leakage collection cavity, is provided to limit leakage to the motor mount in the event all three mechanical seals fail.

The pump gland cooling circuit consists of a pump gland cooler and a recirculation facility, using the auxiliary impeller to ensure a flow of cool high pressure heavy water to the seals even during loss of gland injection.

In order to eliminate the possibility of missile generation, the heat transfer pump motor does not have a flywheel. Instead, it incorporates two sets of inertia packets on the motor rotor, one set at the thrust bearing end and the other set at the pump end. Each inertia packet is an assembly of circular non-magnetic stainless steel punchings (about 1/8″ to 1/4″ thick) shrunk into spider arm of the motor rotor.

This inertia packet arrangement, instead of a flywheel, in the heat transport pumps eliminates the possibility of missile generation due to the following "built-in" design features:

- Distributed inertial mass at the two ends of the motor rotor presents a well balanced rotor. Thus the fatigue life of the rotor itself will not be degraded due to higher vibrations and bending moments associated with a flywheel.
- There is no stress concentration at any single location of the motor rotor. Hence stress related failures at any of the locations of the inertial packet are eliminated.
- Centrifugal forces due to inertia packets are more uniformly distributed. Thus the axial restraints necessary to keep the flywheel in place are not required for inertial packet arrangement.

The incorporation of the two inertia packets on the motor rotor provides the heat transport pump motor sufficient rotational inertia to prolong pump rundown after loss of power. The decrease in the rate of flow approximately matches the power rundown following a reactor trip.

The motor is equipped with two removable radial bearings and a double acting thrust bearing. The motor bearings are lubricated by cooled and pressurized oil. An oil lift system supplies high pressure oil to both sides of the thrust bearing simultaneously during start-up.

Each pump-motor set is provided with a brake capable of stopping an unpowered pump subjected to 80% forward flow.

The pumps and the pump suction and discharge lines are thermally insulated by the feeder cabinet and by insulating material applied to the pump casing.

Some pump motors are supported by spring hangers and are seismically restrained. Pumps are also supported by the headers through the connecting pipes system.

Shielding is installed around the pump, at the level of the top of the volute case, to reduce radiation fields for maintenance.

In Ontario Hydro Bruce and Darlington stations the containment boundary is at the HT pumps and the pump motors and upper part of the steam generator vessels are outside of containment.

To protect the heat transport pump motor from damage which would be caused by operating under conditions of insufficient cooling to the bearings, a trip circuit has been incorporated, which is initiated by a high temperature of the upper thrust bearings. A similar trip also protects the pump from unacceptable vibration following a LOCA. The latter trip utilizes pressure sensors (in the header) to switch off the pump.

Feeders

Figure 8 shows a typical arrangement of feeders and major pieces of equipment located inside containment. They run vertically upwards from the fuel channels across the face of the reactor and horizontally over the fuelling machine to the headers.

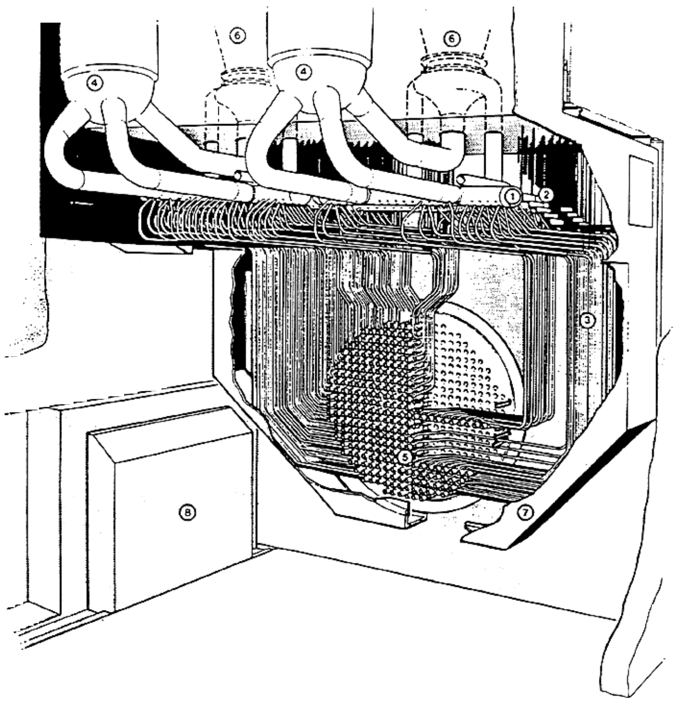
The flow in each fuel channel is based on the channel power and is designed to give steam/water of 4% quality (mass). The maximum inlet to outlet header pressure drop is determined and together with the required channel design flow rate is used to size the feeders.

Reactor Headers

A CANDU 6 station has four reactor outlet headers and four inlet headers. There are two of each at each end of the reactor. This can be seen schematically in figure 2 and in perspective in figure 8. One inlet and outlet header at each end of the reactor handles the flow for 95 feeders.Other reactors have different numbers and arrangements of headers.

Figure 8

Feeder and Header Arrangement



1	*Outlet Header*	*2*	*Inlet Header*
3	*Feeders*	*4*	*Steam Generators*
5	*End Fittings*	*6*	*Heat Transport Pumps*
7	*Insulation Cabinet*	*8*	*F/M Maintenance shielding door*

Valves

There are no isolation valves in the main circuit of the heat transport system because the benefit of having these valves is outweighed by the disadvantage of the radiation dose incurred in the maintenance of such valves and their capital cost. There are valves at the interfaces with the auxiliary systems.

Valves are not required for the prevention of reverse flow because the flow through a heat transport pump and steam generator continues in the same direction when the pump is shut down.

Protection against overpressure for the heat transport system is provided by the combined effects of the pressurizer, liquid relief valves and the reactor safety shutdown systems.

Piping

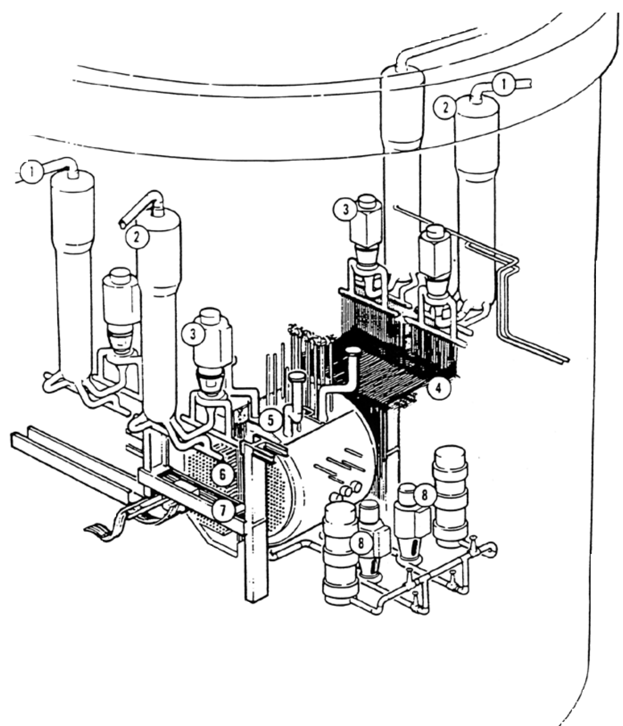
Carbon steel piping is used to minimize the possibility of chemically induced stress corrosion cracking. The procurement specifications for the carbon steel ensures low cobalt content and helps keep the heat transport system low in cobalt-60 activity.

Layout

In CANDU 6 reactors, all portions of the Heat Transport System are located within the Reactor Building. The system components (SGs and HT pumps) are symmetrically distributed at either end of the calandria, and are all situated at an elevation above the reactor (Figures 9 & 10). This permits the system coolant to be drained to the header elevation for maintenance of the HT pumps and SGs, and this also facilitates thermosyphoning (natural circulation) should the HT pumps be unavailable.

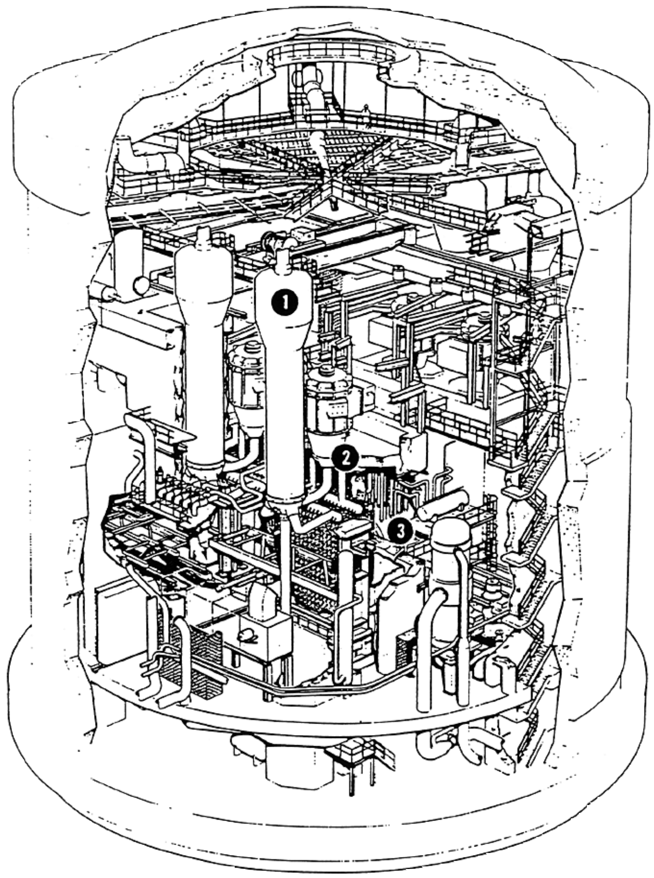
Shielding is provided around all the heavy water portions of the HT system components. The steam generators are shielded up to the level of the steam drums (i.e., above the level of the tube bundle). The circulating pump is shielded from the remainder of the pump set to facilitate pump seal replacement. In Ontario Hydro Bruce and Darlington stations the HT pumps and upper portions of the steam generator vessels are outside containment.

Figure 9
CANDU Nuclear Steam Supply Arrangement



1	*Main Steam Supply Piping*	*2*	*Steam Generators*
3	*Main Primary System Pump*	*4*	*Feeders*
5	*Calandria Assembly*	*6*	*Fuel Channel Assembly*
7	*Fueling Machine Bridge*	*8*	*Moderator Circulation System*

Figure 10
Location of Heat Transport System Equipment



1 *Steam Generators*
2 *HTS Pumps*
3 *Reactor*

5. Interconnections with Other Systems

The HT system interconnects with several auxiliary systems which are integral to its overall performance. Four of the principal systems are illustrated in Figure 11. They are the Pressure and Inventory Control System, the D_2O Collection System, the Shutdown Cooling System and the Purification System. Two of these systems will be discussed in later lessons, the Pressure and Inventory Control System and the Shutdown Cooling System.

5.1 Pressure and Inventory Control System

The components of the HT pressure and inventory control system are the pressurizer, a feed pump, a storage tank and feed and bleed valves. The function of the system is to provide:

- pressure and inventory control for each heat transport circuit;
- overpressure protection;

- a controlled degassing flow.

The pressurizer is designed to absorb the pressure transients to control excessively high or low pressures. The heavy water in the pressurizer is heated to form a vapour space above the liquid. It is this vapour space which cushions the pressure variations. In addition the pressurizer accommodates the change in the volume of the reactor coolant from zero to full power. This permits rapid changes in reactor power and restricts the demands on the components of the feed and bleed circuit.

Under normal operation, when the reactor is at power, pressure is controlled by the pressurizer and the inventory of coolant is maintained by the feed and bleed circuit. At low reactor power, < 5%, and at shut down, the pressurizer is isolated and the pressure is controlled by the feed and bleed circuit, the system is said to be in 'solid mode'. This circuit is also designed to accommodate the changes in coolant volume that take place during heat-up and cool-down.

The coolant for the HT pump glands is provided from the pressure and inventory control system.

5.2 D_2O Collection System

The functions of the D_2O collection system are:

- to collect drainage from equipment prior to maintenance;
- to recover leakage from mechanical components;
- to receive D_2O sampling flow.

Providing the isotopic purity of the collected D_2O is high enough it is pumped to the storage tanks for re-use in the HT system. Otherwise it can be pumped into the D_2O clean up system, which will be discussed in a separate lesson.

5.3 Shutdown Cooling System

The major functions of the shutdown cooling system are:

- to cool the heat transport system from 177°C to 54°C and hold it at that temperature indefinitely;
- to provide core cooling during maintenance work on the steam generators and heat transport pumps and maintain the level of the headers during maintenance.

The shutdown cooling system consists of a pump and heat exchanger at each end of the reactor connected between the outlet and inlet headers of each loop of the heat transport system. The system is normally full of D_2O and is isolated from the HT system by duplicated isolation valves. The shutdown cooling pumps are sized so that no boiling occurs in the fuel channels.

5.4 Purification System

The functions of the purification system are;

- to limit the accumulation of corrosion products in the HT system by removing soluble and insoluble impurities from the coolant,

- to remove accumulations of fine solids which may be released suddenly due to chemical, hydraulic or thermal transients,
- to minimize the level of radioactive contamination in the heat transport system,
- to maintain the pD (pH) log of the deuterium ion concentration of the D_2O at the required isotopic.

The flow through the purification system is provided by the HT pumps. It is taken from one inlet header on each loop of the HT system and is passed through one side of a heat interchanger, a cooler, a filter and an ion exchange column before being returned to the inlet of an HT pump through the other side of the interchanger. The D_2O is cooled to enhance ion exchange and the interchanger minimizes the heat loss. Flow can also be provided during shutdown by means of the shutdown cooling pumps.

The purification system has isolating valves in the inlet and outlet lines which permit the draining of the HT system without draining it. These valves close automatically in the event of a loss-of-coolant accident.

5.5 Power Supplies

One other system essential to the performance of the heat transport system is the electrical system. It provides Class IV power to the HT pump motors and Class II power to the instrumentation and control circuits of the system.

5.6 Instrument Air System

On loss of instrument air the liquid relief valves of the HT system fail open, to prevent possible over pressurization of the system.

5.7 Feedwater and Steam Systems

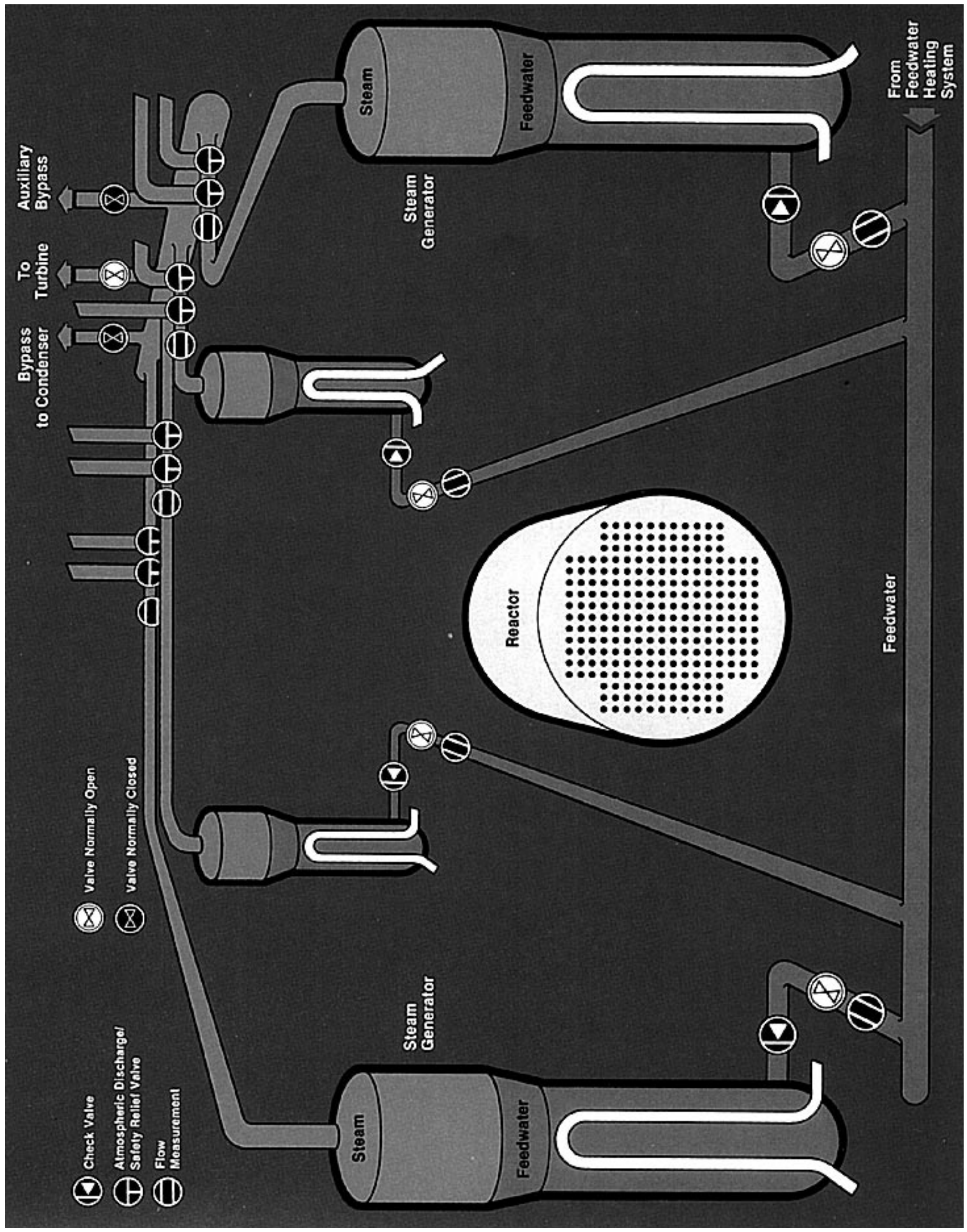
The feedwater and steam systems provide the principal heat sink for the heat generated by the fission reaction in the reactor in addition to serving the following process functions. Figure 12 shows a schematic diagram of these systems.

Feedwater from the feedwater heating system is supplied to each steam generator. The feedwater is pumped into the steam generators by three 50% capacity multi-stage feedwater pumps with the flow rate to each steam generator regulated by feedwater control valves. A check valve in the feedwater line to each steam generator is provided to prevent backflow in the unlikely event of a feed pipe failure. An auxiliary pump is provided which will supply 4% of full power feedwater requirements if the main pumps become unavailable.

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

The heat from the reactor coolant is transferred to the feedwater in the steam generator producing steam. Residual moisture in the steam is removed by the steam separators and the steam is led, by means of four separate steam lines, through the reactor building wall to the turbine steam chest.

Figure 12
CANDU 6 Steam and Feedwater Systems



The steam pressure is normally controlled by the turbine governor valves that admit steam to the high pressure stage of the turbine. If the turbine is unavailable, up to 70% of full power steam flow can bypass the turbine and go directly to the turbine condenser. During this operation, pressure is controlled by the turbine bypass valves. Auxiliary bypass valves are also provided to permit up to 10% of full power steam flow during low power operation.

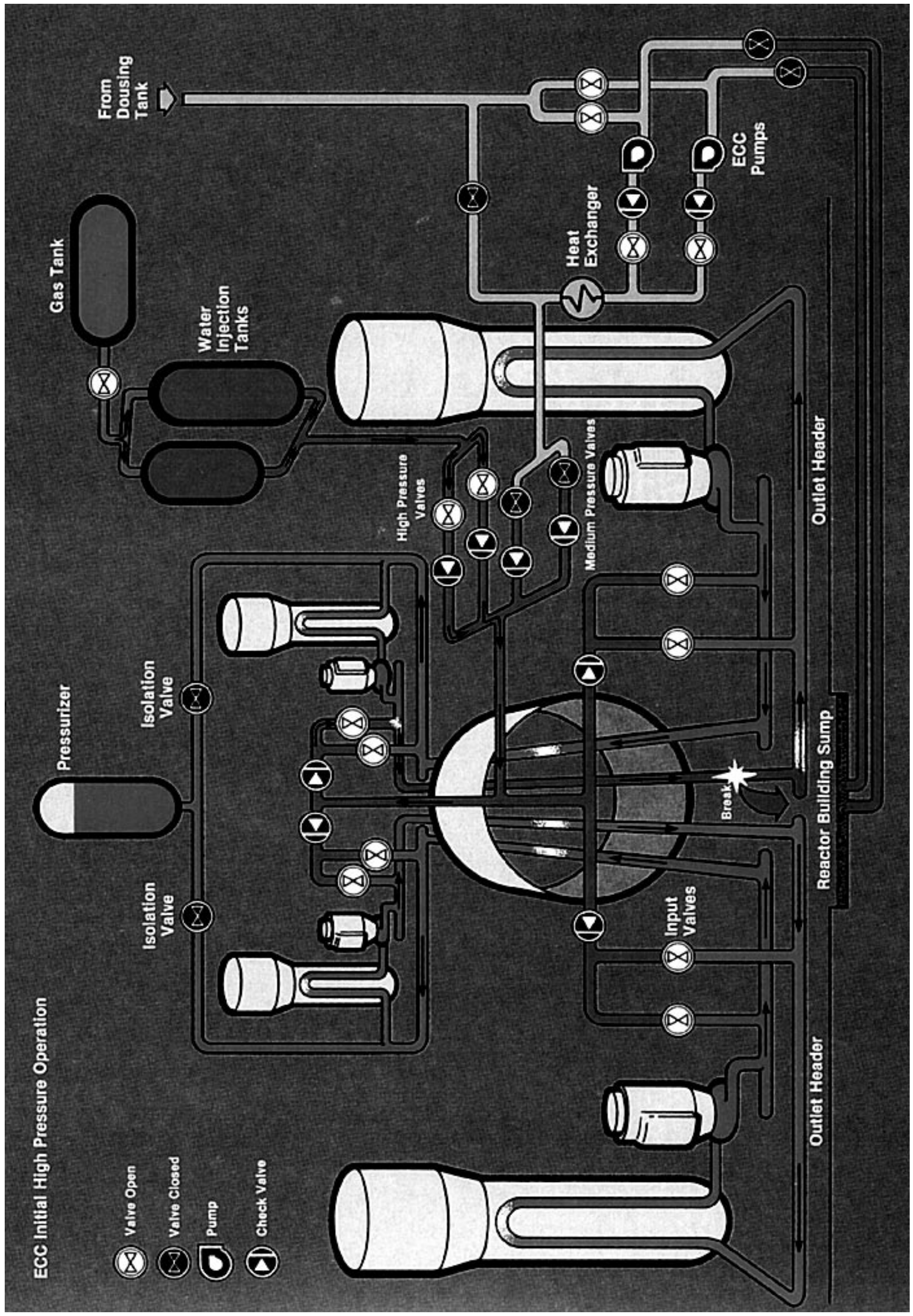
Steam pressure can be controlled by discharging steam directly to the atmosphere via four discharge valves which have a combined capacity of 10% of full power steam flow. These valves are used primarily for control during warm-up and cool-down of the heat transport system.

Overpressure protection for the steam system is provided by four safety relief valves connected to each steam main.

5.8 Emergency Core Cooling System

This is one of the major safety systems. The heat transport system is an integral part of the emergency core cooling system. Figure 13 shows a schematic of the emergency core cooling system and its connections to the heat transport system. The emergency core cooling system will be covered in detail in a separate lesson. Its function is to provide cooling water to the reactor core following a loss of coolant accident (LOCA, an event in which the heat transport system pressure boundary fails and coolant is lost from the system). The objective is to limit damage to the fuel and limit the release of radioactive materials from the station so that the applicable regulatory dose limits for the public are not exceeded.

Figure 13
CANDU Emergency Core Cooling System



6. Control and Monitoring

There are no controls as such on the heat transport system. The flow rate of the heavy water depends on the pump characteristics and the circuit resistance. The control of pressure and inventory is done by the Pressure and Inventory Control system which is described in detail in another lesson.

For the heat transport system, instrumentation and control circuits are used for:

- operation, from the main control room, of the heat transport circulating pumps, including brake and jacking oil control.
- monitoring of pump-motor set for bearing and winding temperatures, and vibration with alarm to the operator.
- providing a reactor stepback signal when a pump is tripped, as monitored by the motor switchgear.
- measuring differential temperatures between the reactor inlet and outlet headers to permit the reactor regulating system to calculate power in the non-boiling power range.

Other systems need to know the status of heat transport system parameters in order to carry out their functions. Parameters that are monitored are:

- temperatures at the reactor inlet headers as a process variable for the pressure and inventory control system,
- pressures at the reactor outlet headers as a process variable for the pressure and inventory control system,
- pressures at the reactor outlet headers to generate a high or low pressure trip signal for the shutdown systems,
- flows in some selected inlet feeders to generate a low gross flow trip signal for the shutdown systems,
- pressures at the reactor inlet and outlet headers to activate the emergency core cooling system,
- heavy water levels in the reactor inlet and outlet headers and in the pump suction line for use when maintenance is being done on pumps and boilers,
- temperatures on all outlet feeders to detect channel blockage in the non-boiling power range,
- gaseous fission products to detect failed fuel,
- delayed neutron activity on all outlet feeders to detect location of failed fuel.
- Boiler level trip circuit to protect H.T. system (no heat sink).

7. Overpressure Protection

There are no isolating or control valves in the heat transport system and therefore the system boundaries extend to the first isolating valve of all the attached systems. For the same reason the overpressure protection covers all the components of the HT system.

Overpressure protection is provided, principally, by liquid relief valves which operate in conjunction with Shutdown System 1 (SDS-1). The Reactor Regulating System (RRS) acts to minimize pressure transients.

Relief Valves

There are four 100% liquid relief valves, two connected to the reactor outlet header of each loop. They are fully instrumented globe valves and they discharge into the degasser condenser. The degasser condenser is itself protected from overpressure by two spring loaded relief valves which are also equipped with air actuators.

In sizing the HT relief valves the following process failures which could lead to HT system pressurization are considered. In each case it is assumed that the degasser condenser pressure is at the set point of its relief valves.

a) The D_2O pressurizing pumps are feeding at maximum rate and the bleed rate is zero. Inflow from the fuelling machines is also included.
b) The steam generator heat sink is lost. To be included are the swells due to decay heat and pump heat and the inflow from the fuelling machines.
c) Both of the first two cases happen at the same time. This is considered to be an emergency condition. The valves are sized to limit the system pressure to 110% of design.

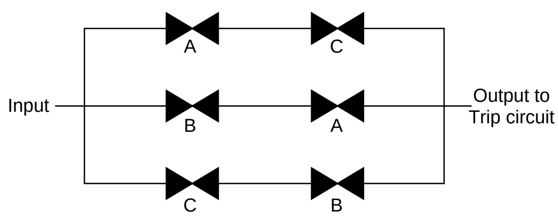
The sensing and control instrumentation for the liquid relief valves meets all of the requirements of the ASME code and has the same redundancy and independence as the shut down systems. The pressure transmitters for the valves are located in each reactor outlet header. The trip parameter logic of SDS 1 uses a signal from these same transmitters. The valves are activated by high pressure in any outlet header, their set point is 10.24 MPa(g) which is the maximum permissible outlet header pressure corresponding to the pressure tube design pressure.

The liquid relief valves fail open on loss of air to the actuator and on loss of electrical power. To ensure reliability, each valve is actuated by two pilot solenoid valves on different power supplies. The valve will open if one or both of the two solenoids is energized. The valves and/or control circuit are required to be regularly tested on power.

Reactor Regulating System (RRS)

The RRS acts to prevent overpressure of the HT system when the reactor is at power. A reactor stepback is initiated on high primary system pressure by dropping the mechanical control absorbers. The signal for stepback initiation is at 10.24 MPa(g) which is lower than the pressure set point for SDS1 initiation. Both signals are measured in the same header but different instrument loop components are used.

Typical Trip Channel Arrangement



2 out of 3 channel trip

Shut Down System

SDS-1 trips the reactor if the HT system pressure, as measured at the outlet header, exceeds 10.45 MPa(g).

There are three independent channels measuring the outlet header pressure. Any two of the channels tripping causes a reactor trip. Each trip channel has a pressure sensor in each of the four outlet headers. If any of these detects a pressure which exceeds the set point then the channel trips.

This arrangement provides not only redundancy but permits the instruments to be tested at power. The testing frequency is intended to ensure and overall system unavailability of less than 10^{-3} years per year.

8. System Operation - Normal Modes

The following discussion applies to the CANDU 6 plant.

100% Reactor Power

During normal heat transport system operation with the reactor at full power, all pumps will be operating, with the pressure at the outlet headers typically controlled at 10 MPa(a) by the pressure and inventory control system. The system temperature will be 266°C at the inlet headers and 310°C at the outlet headers (4% quality steam). The purification, collection, gaseous fission product and gland seal cooling systems will be on line while the hydrogen addition, sampling, recovery, delayed neutron and shutdown cooling systems are isolated but available for use as needed. The emergency core cooling system is poised for use in the unlikely event that it is needed.

Shutdown

Note: The following two sections cover the cooling down and warming up of the reactor and associated systems. To minimize stresses due to different parts of the systems cooling or warming up at different rates the maximum allowable rate of temperature change of the coolant is 2.8°C/min..

The sequence of events for a normal shutdown which could be for steam generator or pump maintenance is as follows:

- The steam from the steam generators is routed past the turbine directly to the condenser. This cools the HT system to 177°C in about 30-45 minutes.
- The isolating valves between the shutdown cooling system and the HT system are opened and the HT pumps are kept running. This forces a portion of the coolant flow through the shutdown cooling heat exchangers. Cooling is provided by service water. From 177°C the heat transport system is cooled to 77°C. This phase could last 6 hours or more and, toward the end, the cooling rate slows appreciably because the HT pumps begin to become the principal source of heat.
- At 77°C the heat transport pumps are shut down to remove their contribution to heating and the shutdown cooling system pumps are started. This cools the system to 54°C where it can be maintained for the duration of the shutdown With the HT pumps shut down the pressure in the system can be lowered by the pressure and inventory control system . At the end of this phase, the heat transport system can be drained to the level of the headers for maintenance of the steam generators and the HT pumps.

Startup

Startup of the HT system is almost the reverse of the shutdown procedure. With circulation being maintained at 54°C by the Shutdown Cooling system, the circuit is filled and pressure raised through the Pressure and Inventory Control system. The Shutdown Cooling system is then isolated and the HT circulating pumps are started up. With the heat generated by their operation the temperature of the circuits is raised gradually. During this period the reactor will be started up and power raised to approximately 5% rated power, to aid in raising circuit temperatures and to provide the necessary steam requirements in preparation for high power operation.

Operation in Solid Mode Control

To operate in solid mode means that either the pressurizer has lost its vapour space or has been valved out (isolated) and the HT system is completely filled with liquid. During cool down and warm up, when the reactor is below 5% full power, the pressurizer may be isolated and the pressure maintained by the feed and bleed of the pressure and inventory control system. A loss of pressure control while operating in this mode will cause rapid fluctuations in pressure.

For example, if the D_2O feed pumps are lost then the pressure will begin to fall rapidly because of the continuing bleeding from the system. Within minutes the

HT pumps will begin to cavitate and vibrate and will need to be shut down as soon as possible. The operator gets no advance warning of this kind of event because the feed pumps have no flywheels and will stop quickly, the first indication will be alarms showing a problem. For this reason it is recommended that both D_2O feed pumps be operated during solid mode pressure control to reduce the risk.
Considering another example, if, during operation in solid mode, the HT relief valves fail open. The result will be, again, a sudden drop in pressure with the HT pumps cavitating and vibrating severely.

Because the operator is unable to respond to this kind of event quickly enough it is recommended that operation in the solid mode be limited and avoided as much as possible.

9. System Operation - Abnormal Modes

9.1 Operation with a Failed Open Liquid Relief Valve

If one of the HT relief valves failed open while at power, D_2O discharges from the reactor outlet header to the degasser condenser until the pressure equalizes on both sides. There is no spill of D_2O through the degasser condenser relief valves since the setpoint of these relief valves [10.16 MPa(a)] is higher than the normal pressure of the reactor outlet header [10 MPa(a)]. The pressure and temperature of the HT system initially drop, and are restored to normal values by the Pressure and Inventory Control system.

There is a risk of spilling D_2O through the degasser condenser relief valves to the fuelling machine vault if a high pressure transient occurs (for example, due to a loss of Class IV power) following the fill-up of the degasser condenser. Therefore, continued reactor operation with a HT relief valve failed open is not recommended.

9.2 Operation with a Failed Variable Pressurizer Heater

The five pressurizer heaters HR1 through HR5 consist of four On-Off and one variable heater. The heaters are provided with quick disconnects such that heaters 1,3 and 5, connected to the `**Odd**' power supply, are interchangeable by interchanging the power cables from one heater to the other during shutdown. Similarly, heaters 2 and 4, connected to the `**Even**' power, supply, are interchangeable. In case the single variable heater fails, the task of controlling HT pressure, during normal reactor operation, ceases to be automatic. It then requires operator action to switch the fixed set of heaters on-and-off to maintain nominal set pressure. The drawback associated with manual pressure control, following the variable heater failure, is that the HT system pressure could drop equivalent to the pressure span covering the range of operation of the variable heater and the fixed heaters deadband, thus reducing the critical power ratio.

Thus long term operation with a failed variable pressurizer heater is not recommended.

9.2.1 Operation with Pressurizer Out of Service

A failure of the pressurizer at power levels greater than 5% FP would result in the pressure control system switching to the "solid mode". This mode uses the FEED and BLEED valves to control pressure. The feed and bleed valves do not have the capacity to control pressure effectively when in a transient condition thus the solid mode should only be used for a limited time when greater than 5% F.P.

9.3 Operation with Steam Generator Tube Leakage

Because of the high differential pressure between the heavy water in the steam generator tubes and the light water and steam on the shell side, tube leakage could result in the loss of a considerable amount of heavy water to the feedwater and steam system. This will result in the presence of radioactive contaminants, tritium, cobalt-60 etc. in the steam and feedwater system with consequent radiological hazards in the normally clean conventional side of the plant and possible increased releases of radioactivity from the station. In addition, continued operation with steam generator tube leakage is not recommended since it could result in complete rupture of the tube.

When the leak of a steam generator tube is confirmed the reactor must be shut down and the tube identified and plugged.

9.4 Operation with the HT Purification System Isolated

Should it become necessary to operate the reactor at power with the HT Purification system isolated, inventory control via the bleed lines is no longer possible. An alternate bleed can be set up via the degassing valves, the pressure and inventory control will be maintained by automatic operation of the feed valves. The changeover could, under controlled conditions, be effected with a minimal disruption to the HT system. However, prolonged operation in this mode is not permissible due to the resulting loss of coolant chemistry control and possible buildup of I-131.

9.5 Operation with HT Pump Seals Failed

As mentioned earlier, where the pump shaft penetrates through the pump casing there is a set of three mechanical seals, in series, to minimize leakage of hot coolant from the pump. A seal forms a path of high resistance to flow through it and can operate with a high differential pressure. However there will always be some small flow or leakage.

Mechanical seals have a limited life. Damage to mechanical seals occurs due to:

i) wear caused by the high pressure differential and particulate matter. As the hot coolant passes between the faces of the seal it flashes to vapour and because of the greater volume travels at a much higher velocity. That by itself can damage the seal faces; however if there is particulate material included then the wear is accelerated.

ii) deterioration and distortion caused by high temperatures and temperature transients.

The life expectancy of a seal package cannot be accurately predicted because failure in one seal will considerably reduce the life time of the remaining seals; for example, particulate matter released by the failure of one seal may accelerate the failure of the next seal downstream. It is recommended that seal packages be replaced at intervals of approximately two years to minimize forced outages due to seal failures.

A failed mechanical seal is defined as one which is damaged to such a degree that its leakage rate is higher than acceptable. If unchecked, such a condition can progress and extend to all three mechanical seals. The result would be hot flashing outflow from the HT system to the gland and hence, in part to the gland return line, to the seal leakage line and, if the backup seal has also failed or is leaking, to the motor stand. Leakage into the motor stand could result in damage to the pump motor and release of D_2O, tritium, and actuated corrosion products to the reactor building atmosphere.

Monitoring of seal performance is accomplished by noting changes in pressures and temperatures in the seal cavities.

Normally, the pump can be operated with the HT system hot and pressurized with one failed seal but close monitoring of the other two seals is required. With two or three failed seals, the pump should be stopped, immediately.

9.6 Operation with One Heat Transport Pump Blocked

The system has been designed to operate with one HT pump shut down. However, this is not considered to be a mode of continuing operation. Here is the sequence of events under this circumstance:

On loss of one HT pump, a signal, activated by the pump motor breaker, initiates a reactor stepback to 65% of the current power. The speed of the failed pump reduces to an equilibrium value of about 500 rpm; when the speed is below 800 rpm, the brake should be activated, and the pump stopped. If this is not possible then the pump can be allowed to freewheel for a short period of time (up to 100 hours). Coolant flow in the loop under these conditions is about 70% of normal flow.

The purification system connection to one loop should be closed as soon as possible. If the failed pump cannot be restarted and extended operation with the pump shut down is required, the corresponding pump in the other loop should be tripped and braked as indicated above and the purification system reconnected if appropriate.

9.7 Operation with Two Heat Transport Pumps Blocked

Here is what the design will permit if a second HT pump fails while operating with one already out of service:
If the second pump which fails is in the same loop as the first one then an immediate reactor shutdown will be initiated. Operation cannot be resumed until at least one of the failed pumps is repaired.

If the second failed pump is at the opposite end of the reactor in the other loop then the reactor must be shut down and one of the pumps repaired.

In all cases where there is a different flow in each loop the HT purification system must be isolated from one loop and if operation in this mode is prolonged then purification should be alternated between loops.

9.8. Recovery System Operation

In the unlikely event of a small coolant leak from the HT system, the D_2O recovery system will collect the liquid leakage from the HT system's components in sumps located in the Fuelling Machine Vaults. This leakage is sampled and, if not seriously downgraded, returned to be used by the pressure and inventory control system.

The operation of the recovery system will allow the HT system to continue operating for a limited time, or to be shut down under controlled conditions, providing the leakage rate is within the capacity of the pressure and inventory control system.

10. Emergency Conditions

10.1 Loss of Coolant Accident (LOCA)

A major piping break in the HT system will result in a rapid decrease in the system pressure in the defective loop, with a corresponding rise in containment pressure due to the escaping coolant flashing to steam. It is the purpose of the Safety Systems to ensure that, irrespective of the location or size of the HT system break, the following objectives are met:

a) Any release of radioactivity be limited to the containment area.
b) Adequate cooling of the fuel be maintained at all times.
c) Damage to HT system components be minimized.

To achieve these objectives when there may be very limited time for operator response, the corrective action required, via safety system operation, has been automated to give the following sequence of events:

a) The reactor will be tripped by SDS-1 and/or SDS-2.
b) The defective loop will be isolated from the other one to maintain one loop intact and minimize heavy water losses.
c) The containment isolation system will operate to isolate penetrations through the Reactor Building and close off all potential sources.
d) The Emergency Core Cooling system will be activated, on a LOCA, when the heat transport system pressure drops to 5.5 MPa and a circuit isolation system (independent of ECCS logic) closes the applicable valves to isolate the ruptured loop. Operation of the ECCS is discussed more fully in another lesson.
e) The Dousing system will be activated should the Reactor Building pressure rise sufficiently.

10.2 Post LOCA Operation of the HT Pumps

Following a LOCA (Loss of Coolant Accident) the HT pumps are required to continue running to provide maximum fuel cooling until thermalhydraulic conditions which are conducive to thermosyphoning are established. From a safety point of view, the time required for HT pump operation is dependent on the size and location of the LOCA.

The post LOCA operational requirements for the HT pump are:
a) The HT pumps are only required to operate for single failure events, i.e., for LOCA conditions, but not for LOCA with loss of Class IV power.
b) The operation of the HT pumps does not lead to a worse LOCA condition, e.g. shaft failure, hole in pump case. The failure of the mechanical seals is tolerable during LOCA conditions.

An evaluation of post LOCA HT pump operation has indicated that the pumps are capable of operation after a LOCA, and that the HT pumps satisfy all design requirements.

The respective equipment must contend with the adverse conditions associated with a LOCA - both the two phase flow in the primary system and the steam, temperature, pressure and radiation in the reactor building.

To ensure that the auxiliary services to the HT pumps are maintained following a LOCA, the following components of the main heat transport system have been environmentally qualified:
- HT pump motor supply power cables and cable connectors.
- Instrumentation (including cables and connectors) for determining pump performance.

Other post LOCA requirements that have been fulfilled include mechanical and structural integrity of the pump motor and all motor components.

11. Management of the Radiation Hazards

The source of all radioactive materials in the system is the core of the reactor and specifically the fuel. The heat transport fluid passes through the core, over the fuel and therefore is the means by which radioactivity can be transported to other parts of the plant. There are three principal sources of activity: fission products in the fuel, activation products arising from corrosion products in the coolant which pass through the neutron flux, and activation products produced in the coolant itself.

As the fuel is irradiated, fission products are formed in increasing quantities. These fission products are retained within the individual fuel elements of the fuel bundle unless there is a fuel element sheath failure, when some of them are released to the heat transport coolant. A small percentage of the 37-element fuel bundles (about 0.1%) at CANDU 6 and Bruce A have, historically, developed failures in the fuel sheathing (i.e., five bundles per year). These defects result in the escape of only a small fraction of the total inventory of the fission products from the defective element in the bundle. The escaped fission products, primarily the more volatile ones, enter the coolant and are distributed by the coolant to the surfaces of the HT system, to the HT purification system or they remain in solution. Thus, the majority of the escaped fission products can simply decay within the boundaries of the HT system.

Defective fuel bundles are located by the use of the delayed neutron and/or gaseous fission product monitoring systems and should be removed from the reactor promptly for subsequent storage.

Metals in contact with the coolant in the HT system corrode slowly. Some of the corrosion products dissolve or become suspended in the coolant and so may pass through the neutron flux in the reactor core and become activated. By controlling the alkalinity of the coolant and maintaining the pD value above 10.2, the deposition of corrosion products on the fuel element surfaces is minimized. There is a continuous exchange of dissolved and suspended materials between the coolant and the surfaces of the system and so there is a gradual buildup of activation products and as a result, radiation fields on the surfaces of the system.

Activation products are produced in the coolant itself. The principal radionuclides produced are nitrogen-16 from an (n,γ) reaction with nitrogen-15, oxygen-19 from an (n,γ) reaction with oxygen-16 and fluorine-17 from an (n,p) reaction with oxygen-16. These radioactive materials are short-lived e.g the half life of nitrogen-16 is seven seconds, and are only of concern during operation at power. The radiation fields from them decay quickly to negligible values shortly after shutdown. Tritium, and to a lesser extent carbon-14, are also produced in the coolant and become internal hazards if they escape from the system.

Because the HT system operates at high pressure and temperature and because it has many mechanical joints, such as end fittings, valves and pump seals, there is chronic leakage of coolant either as vapour or liquid. This leakage will carry both fission and activation products. Liquid collection systems and water vapour recovery systems are a means of control and serve as barriers to their release to both containment and the environment.

External and internal exposures of plant personnel are controlled by providing training for staff 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 dose 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.

Control of HTS Inventory and Pressure (Including Degasser-Condenser Level and Pressure Control)

Training Objectives

On completion of this lesson the participant will be able to describe;

- How the inventory is controlled over the range of zero power hot to full power.
- How the pressure is controlled during normal operation by the heaters and the steam discharge valves of the pressurizer.
- How the system pressure is controlled without the use of the pressurizer.
- How the isolated pressurizer pressure is controlled.
- How the level and pressure in degasser-condenser are controlled.

Table of Contents

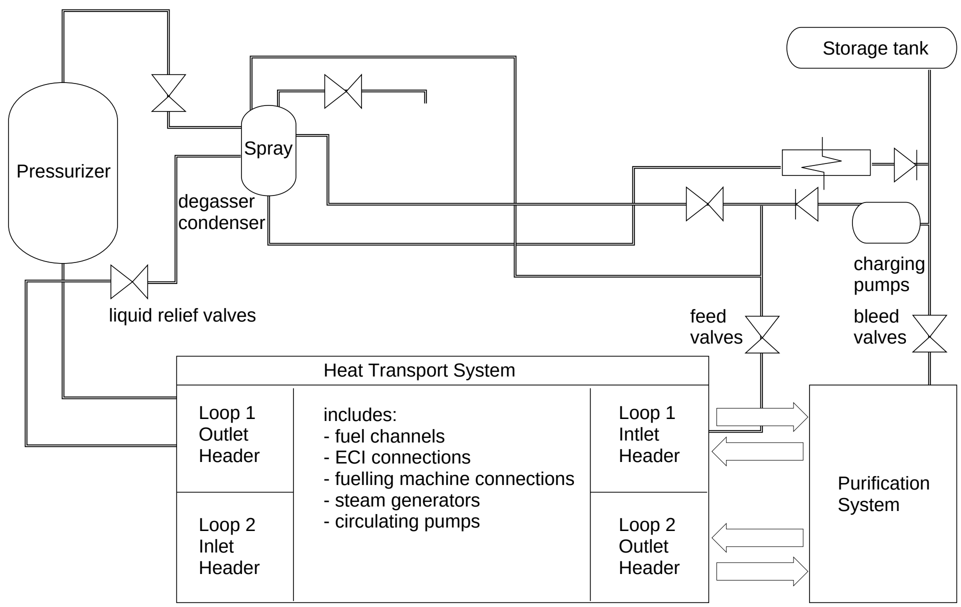
1.0 Introduction

The main function of the Inventory and Pressure Control System is to provide a reliable means of controlling inventory and pressure in the Heat Transport System (HTS). The principal components and their connections to each other are shown in Figure 1.

The major equipment consists of a pressurizer, a degasser-condenser and two pressurizing pumps, along with storage and purification systems.

Figure 1

Simplified Heat Transport Flow Sheet



The information given is typical of a CANDU 6 reactor. Some CANDU units do not make use of a pressurizer, Pickering is the only Canadian example.

The vapour space above the heavy water in the pressurizer is created by heating the water electrically. Since the water and vapour are at saturation conditions, the temperature of the water sets the pressure in the vapour space. The volume of the vapour space is designed to cushion pressure transients without allowing excessively high or low pressures.

The pressurizer also accommodates the change in volume of the reactor coolant from zero to full power. This permits the reactor power to be increased or decreased rapidly without imposing a severe demand on the feed and bleed components of the system.

When the reactor is at power, pressure is controlled by the pressurizer and coolant inventory is maintained by the feed and bleed circuit. At low reactor power or when the reactor is shut down, the pressurizer may be isolated and the pressure controlled by the feed and bleed circuit.

2.0 Functional Requirements of the System

2.1 Safety Requirements

- To maintain adequate inventory in the HTS to ensure heat removal from the fuel and thermosyphoning capability following accident conditions
- To provide overpressure protection of the HTS during accident conditions.
- To provide the low pressurizer level trip signal to the reactor shutdown system.

2.2 Process Requirements

- To accommodate the HTS coolant swell and shrink associated with warm-up, cool-down, power manoeuvring and unit disturbances.
- To limit pressure increases in the heat transport system during transients and to limit pressure decreases so that adequate Net Positive Suction Head (NPSH) on the main HTS pumps is ensured.
- To provide a means of heat transport system pressure recovery following sudden power reduction such as a trip or a stepback.
- To provide adequate relief capacity for overpressure protection of the main heat transport system as well as a means to contain any relief from it.
- To provide a means of degassing the HTS water.

3. Design Requirements for Control of System

3.1 Safety and Reliability Requirements

The integrity of the HTS against overpressure caused by too great a reactor power is assured by the protection provided by the two shutdown systems and the reactor regulating system.

In addition, as a safety requirement, some valves are required to close after a Design Basis Earthquake (DBE) to prevent loss of coolant.

The operational reliability of the system is assured by triplicating the most important loops and duplicating others so that failure of one loop does not incapacitate the system.

3.2 Functional Requirements

The basic functional requirements for the control of the HTS inventory and pressure are to;

- control reactor outlet header pressure under normal operating conditions.
- control heavy water inventory in the heat transport system by controlling pressurizer level.
- provide a facility for controlling the rate of degassing the heat transport system.
- provide controls for the overpressure protection of the pressurizer when it is isolated.

- provide signals of outlet header pressures to the reactor regulating system that initiate reactor stepback under certain conditions.
- provide an adequate display of system parameters to the operator together with alarm annunciation of important parameters.
- provide pressure control even if computers are unavailable.

In addition to these basic requirements, the heaters and the heat transport pumps require protection.
- The heaters in the pressurizer and the degasser-condenser must be cut-off on low water level to avoid overheating of the elements.
- The heat transport feed pumps must be tripped on low suction pressure to avoid cavitation damage.

3.3 Some Causes of System Overpressure

The heat transport system pressure could become greater than desired under the following upsets;
- if more water is fed into the system than is bled out. This could happen if the feed valves failed open or if the bleed valves failed shut.
- if the average temperature of the heat transport system is increased without allowing for the expansion of water (the swell). This could happen if;
  - the pressurizer heaters fail to shut off and the pressurizer temperature and hence its pressure increases.
  - regulation is lost and the reactor power increases, raising the average heat transport temperature
  - Class IV ac power is lost which causes HTS circulating pumps to stop, then heat is not transported to steam generators as before and the average temperature increases.
  - steam flow from the steam generators stops, then heat is not transported away from the reactor core, and the HTS temperature increases.
  - feed water flow to the steam generators stops, which will prevent heat from being transported from the heat transport system and its temperature will increase.

Overpressure in the HTS can be reduced quickly by either relieving water from the system to somewhere else; or by reducing the heat input through reactor trip; or by cooling the HTS system by discharging steam from the steam generators.

The purpose of the control system is to avoid using these drastic ways of pressure control during normal operation. This lesson describes the ways of maintaining control over the pressure in a way that allows normal power production.

4. Control System Overview

4.1. Heat Transport Inventory Control - Normal Operation

The function of the inventory control program is to maintain a constant mass inventory of heavy water in the heat transport system under different operating conditions. More details are given in section 5.1.1.

The level in the pressurizer is set by the positioning of the feed and bleed valves. The level setpoint is programmed to increase with power such that at zero power the level is just sufficient to cover the pressurizer heaters while at full power sufficient liquid and steam volume are available to dampen the effect of any transients, and provide D_2O for immediate return to the heat transport system.

The pressurizer level setpoint for positioning the feed and bleed valves is generated by the computer program using neutron power, heat transport system outlet header pressure and reactor inlet header temperature. From these variables a heat transport system mass balance calculation is made and the pressurizer level setpoint derived. For high reliability this control program can be run from either of the two unit computers. Triplicated level measurements from the pressurizer are fed to the computer where the median is selected.

4.2 Heat Transport Pressure Control - Normal Operation.

There are two modes of pressure control; "normal" operation using the pressurizer and "solid" mode without the pressurizer. "Solid" mode is outlined in section 4.4. More details of the pressure control under normal operation is given in section 5.2.1.

The pressurizer is the principal component in the pressure control of the HTS. It is a pressure vessel which is partly full of liquid D_2O with the remainder being saturated vapour in equilibrium with the liquid. Under normal power producing operation, the HTS pressure is controlled by the pressure in the pressurizer. This is done by use of the steam bleed valves and the heaters in the pressurizer. The control computers regulate these components so that;

- If the pressure is above the setpoint, the steam bleed valves on the pressurizer open to reduce pressure.
- If below the setpoint, heaters in the pressurizer are switched on to raise the temperature of the water and hence the saturation pressure.

During operation at power, the pressure control system covers a narrow pressure range, and takes its signals from four narrow range transmitters for high accuracy. Triplicated sets of pressure measurements taken from the outlet headers are used. The median pressure on each header is used by a control program which selects the highest of the four as representing the system pressure. It is this value that is compared to the pressure set point to develop the control action.

The control program operates in each of the two computers for reliability.

There are five heaters in the bottom of the pressurizer;

- one variable heater which is used under normal steady state conditions and provides fine control,
- four "on-off" heaters which are used for broader range control. These come on when the pressure drops below the proportional band setting of the pressure controller or if the water temperature, sensed just above the heaters, falls a predetermined amount below the normal saturation value - course control.

4.3. Interaction of Inventory and Pressure Control

During increases in power the pressurizer accommodates the resultant HTS swell. The rising water level compresses the vapour space above the water and pressure is maintained by the steam bleed valves opening. The water flowing into the pressurizer is below the saturation temperature for the controlled pressure. This reduces the thermal capacity of the pressurizer and its ability to reduce pressure transients. This condition is sensed by temperature detectors in the pressurizer and the heaters are turned on to increase the water temperature to the saturation value. The heaters are automatically shut off if sufficient water is not present to keep them covered.

4.4. "Solid" Mode Pressure Control (When Pressurizer is Isolated)

This mode is used when the reactor power is below 5% or when the operator needs to quickly change the pressure over a wide range.

Under these circumstances the pressurizer is isolated from the HTS and the HTS pressure can not be controlled by using the steam bleed valves and the heaters in the pressurizer.

In this case the HTS essentially contains no vapour and there is little damping of pressure changes by the liquid. This lack of damping has led this form of pressure control to be known as "solid" mode pressure control.

"Solid" mode control covers a wide pressure range and does not have the capacity to handle transients which may occur when the reactor is at high power. It should not be used when the plant is producing any substantial power. High reliability is therefore not required since it is not to be used under power producing conditions.

HTS pressure control in this mode is by a single channel analog system using signals from pressure transmitters located on the outlet headers and positioning the two feed valves and the two liquid bleed valves. The transmitters are not the same as those used for normal mode pressure control. The highest of the outlet header pressures is selected and is compared to the setpoint in a panel-mounted pressure controller.

With the pressurizer isolated, its pressure is controlled by a single channel pressure control loop controlling the steam bleed valves and the heaters. Overpressure protection is provided by tripping open the steam relief valves on high pressure. This protection is provided by triplicated pressure measurements on the pressurizer.

4.5. Degasser-Condenser Level Control

Level control on the degasser-condenser is by two separate analog systems each controlling the position of its own valve. The water drains from the degasser-condenser to the suction of the HTS feed pumps and the storage tank. Water is added from the degassing flow and during HTS pressure upsets from the pressurizer steam bleed valves.

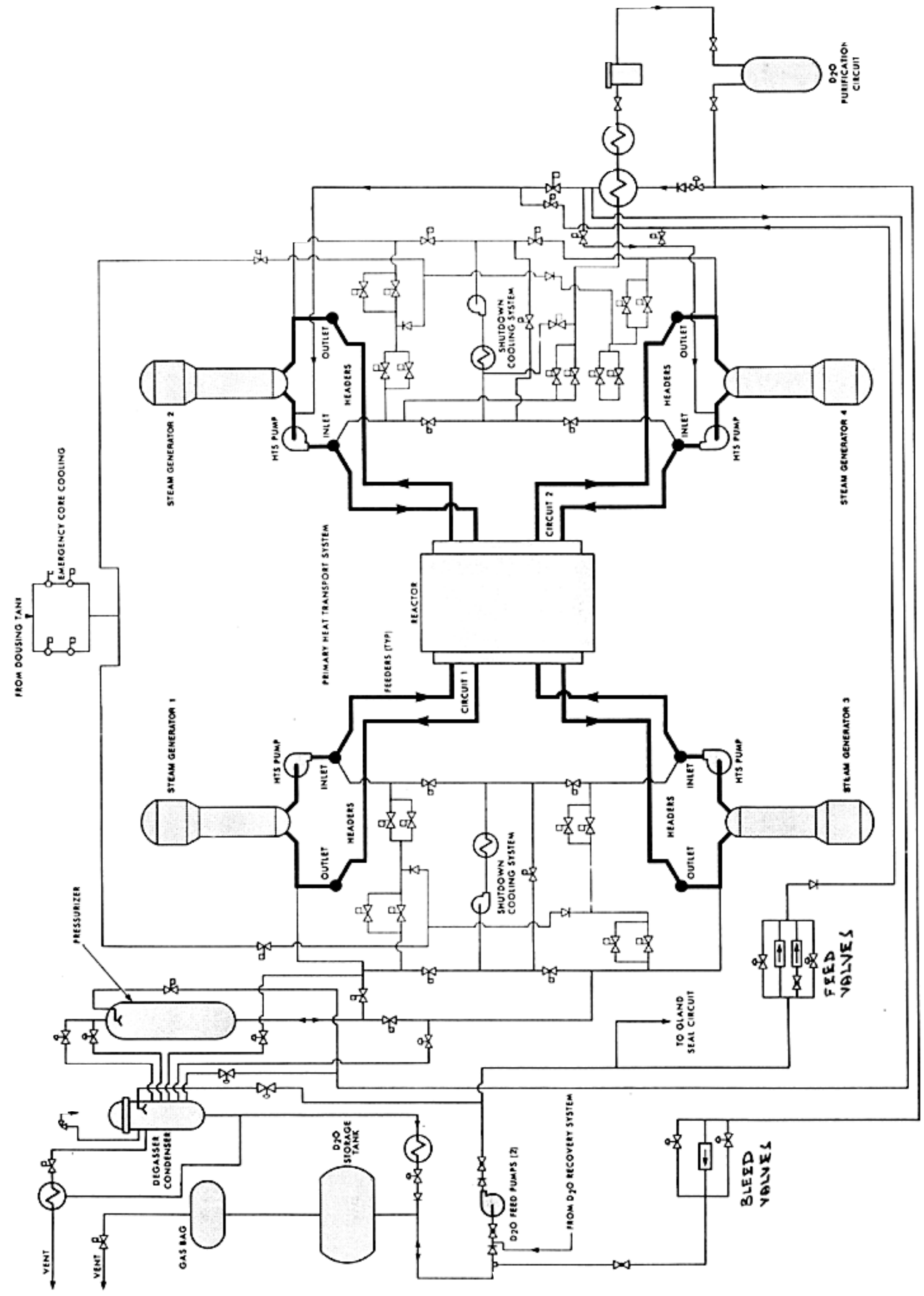
4.6. Degasser-Condenser Pressure Control

Pressure is controlled by condensing the vapour in the condenser. This is accomplished by a spray flow as demanded by the pressure controllers operating in two separate systems Normally there is no steam flow into the condenser, so no spray flow is required. A minimum pressure is maintained with electric heaters.

5.0 Control System Descriptions

This section provides details of the control of pressure, level and temperature of the HTS, the pressurizer and the degasser-condenser under normal and "solid" mode operations. The connections between the HTS, pressurizer, degasser-condenser and the feed pumps are shown on Figure 2, which is a greatly simplified flow sheet.

Figure 2
Simplified Flow Sheet



5.1 Heat Transport Inventory Control

5.1.1 Pressurizer Level Control Loop - Normal Operation

The inventory of the heat transport system is controlled by the use of the feed and bleed valves as dictated by a pressurizer level setpoint.

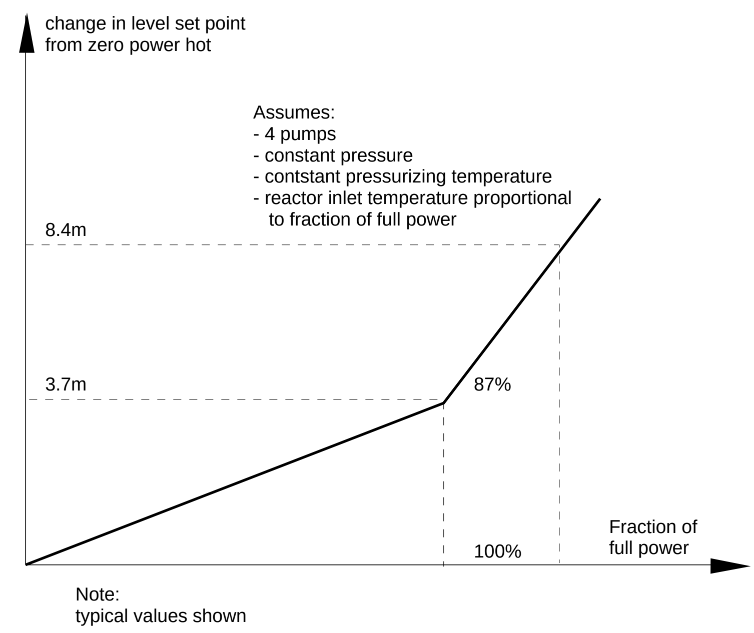
The system is designed so that the D_2O swell from zero power cold to zero power hot is taken to the storage system. At zero power hot there is sufficient liquid in the pressurizer to cover the heaters.

From zero power hot to 100% power, D_2O swell in the heat transport system is accommodated in the pressurizer, so that if a trip occurs the D_2O is available for immediate return to the heat transport system. The constant mass inventory program calculates the amount of swell in the heat transport system from 0% power hot to 100% power and follows the actual swell closely. This avoids excessive feed and bleed operation. Thus there is always enough D_2O in the pressurizer to handle a reactor trip.

There is a relatively large amount of boiling in the reactor core and this results in a non-linear swell characteristic. Figure 3 shows the effect of boiling on swell during startup.

Figure 3

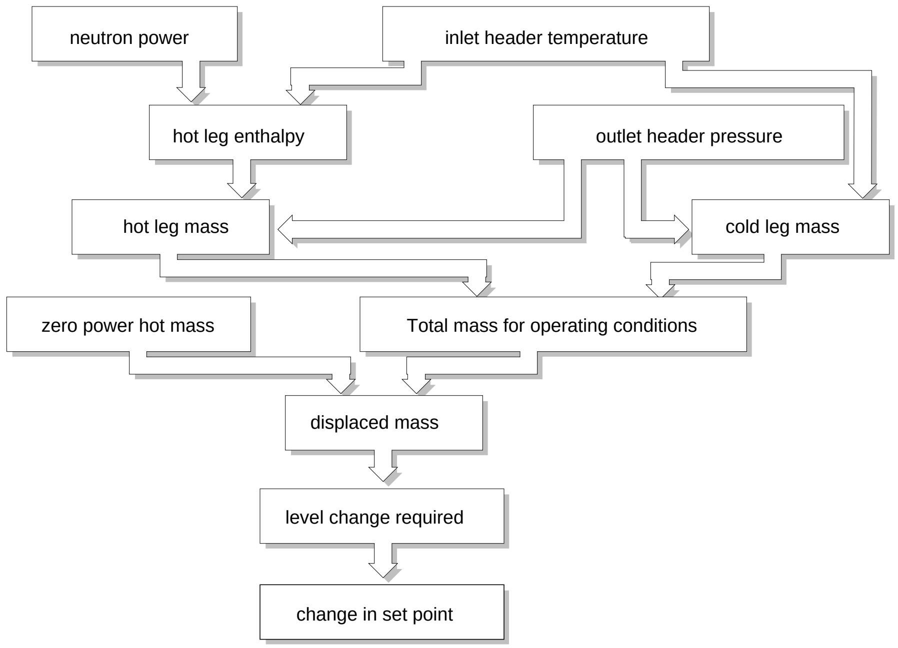
Pressurizer Level Set Point Change as a Function of Power



The pressurizer level setpoint for controlling the feed and bleed is generated by a computer program using neutron power, heat transport system outlet header pressure and reactor inlet header temperature. From these variables a heat transport system mass balance calculations is made and a pressurizer level setpoint is derived. Since the effect of boiler pressure is taken from the reactor inlet temperature, the effect of changing boiler heat transfer coefficient with time is automatically included. Figure 4 illustrates the elements of the control program.

Figure 4

Computation of Pressurizer Level Setpoint



The program calculates and sums the mass in the four cold legs and the four hot legs. This allows for different pressure, temperatures and steam qualities in the legs when operating with one pump out in a loop. One loop consists of two cold legs and two hot legs. At power with one pump operation of a loop and two pump operation of the other loop, there could be boiling taking place in only one of the hot legs of the one pump loop. This effect is automatically included in the program for any combination of operating pumps.

The amount of boiling at a given power level will depend on the pressure and temperature distribution throughout the HTS. During fast manoeuvring the change in the amount of boiling will depend on the pressure and temperatures at that power level. The control programme will allow for this in adjusting the level setpoint in the pressurizer.

Once the pressurizer level set point has been calculated the programme compares the computed setpoint to the actual pressurizer level. If actual level is lower than the setpoint then the feed valves will open, if higher, the bleed valves will open.

For reliability this loop is triplicated with median selection done in the computer.

The level measurement is corrected within the computer programme to compensate for D_2O temperature variations. This ensures the measurement is the true level at all times.

The pressurizer level is displayed on the CRT and on the main panel.

In order to protect the pressurizer heaters against overheating and consequent damage, the heaters will be cut-off when;

- the pressurizer level reaches a specified low level setpoint. The cut-off level takes into account the maximum instrument error as well as a safety margin. An alarm indicates heater cut-off on low level.
- the computers are down. The heaters are no longer required since the loss of both computers results in unit shutdown. The heater cut-off is implemented by the computer.

If the actual pressurizer level deviates from the programmed level by more than a certain amount the deviation alarm comes in. This alarm would indicate faults with the level program or gross leakage from the heat transport system. This alarm will be inhibited on low reactor power (<2%).

On excessive low pressurizer level relative to the setpoint the valves controlling the flow from the pressurizer to the degasser-condenser close.

The pressurizer level setpoint can either be "computer calculated" or "operator select". The choice is made from the computer keyboard. During normal operation the system will operate in the "computer calculated" mode. With the pressurizer level setpoint in the "operator select" mode, the desired setpoint is entered from the keyboard.

On a Class IV power failure or if both heat transport pumps failed in either loop, it is not possible to calculate the mass of D2O displaced into the pressurizer since the flow through the core is unknown. In this case, the level setpoint is initially increased by 10% to avoid hot bleed due to the additional swell in the core and then after 102 seconds the level setpoint is reduced to 4.8 metres (4.8 m is equivalent to 45% power level in normal operation), where it is held constant.

5.1.2 Feed and Bleed Valves

The feed flow into the main heat transport system is controlled by two valves. These valves are normally controlled automatically but can be controlled manually. The valves are designed to fail open. During normal operation, the feed valve opening will be determined by the level in the pressurizer. In the "solid" mode, valve opening will be determined from header pressure measurements.

The bleed flow from the main heat transport system is controlled by two valves in a manner similar to the feed valves. The valves are designed to fail closed.

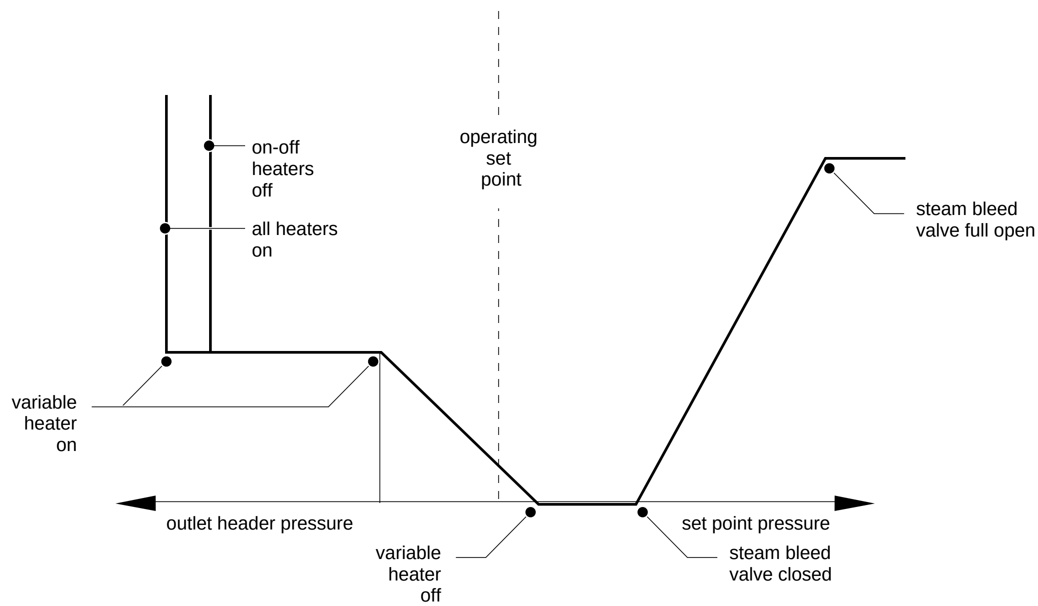
The operator can manually set the feed and bleed valves to any desired opening from the main control panel during pressurization and depressurization of the heat transport system.

In addition to direct manual intervention using the manual stations, the opening of the feed and bleed valves can also be set manually from the computer keyboard. This computer manual setting is used for bumpless transfer from analog manual to computer control.
To prevent a loss of coolant after a DBE, the operator must ensure the bleed valves are closed by operating the panel mounted controllers.

5.2 Heat Transport Pressure Control

The reactor pressure is controlled by control of the temperature and pressure in the pressurizer. Figure 5 outlines how the heaters and the steam bleed valves are used to control the pressure.

Figure 5
HTS Pressure Control



5.2.1 HTS Pressure Control Loops - Normal operation

Under normal operation, the pressure control system will hold the pressure in the heat transport system reactor outlet headers at 9.89 MPa(g) by control of the steam bleed valves and the heaters in the pressurizer. There are five heaters: one variable heater and four ON-OFF heaters.

The median measurement for each header is selected as representing the pressure in that header. The highest of the four medians is used in the pressure control program. The controller action is proportional.

The pressure transmitters, have a range of 6.0 to 12 MPa(g). Each of the four outlet headers has triplicated transmitters giving a total of twelve transmitters.

Pressure control will normally be by modulating the output of the variable heater so as to correct for heat loss through the pressurizer walls. The ON-OFF heaters are normally used to increase the temperature of the pressurizer water if it should be cooled by incoming swell. The control of these heaters for this purpose is described in section 5.2.3.

If the steam bleed valves develop a leak (say > 0.6% maximum flow), the 200 kW variable heater will be unable to cope and the pressure will drop until the ON-OFF heaters come on. This will result in poor pressure control. If the steam leak is excessive, one of the ON-OFF heaters is selected to be on continuously.

A low pressure alarm occurs if the reactor outlet header pressure is too low. To avoid nuisance alarms during reactor power reduction the alarm setpoint is made a function of neutron power. To avoid a nuisance alarm when operating with less than four heat transport pumps, the highest rational outlet header pressure is used.

Analog output signals controlling the variable heater and the steam bleed valves can be shown on the CRT display.

If the reactor outlet header pressure drops below 5.52 MPa(a), the pressurizer isolating valves, the feed isolating valves and the purification system isolating valves will be automatically closed. This isolates the main circuit from the pressurizer and from the purification system.

5.2.2 Steam Bleed Valves

Two valves control pressurizer pressure by directing steam from the pressurizer to the degasser- condenser and are normally controlled automatically.

In the normal mode, valve opening will be determined by the difference between the header pressure measurements, triplicated on each reactor outlet header and the pressure set point. In the "solid" mode, valve opening will be determined by the pressurizer pressure control loop.

5.2.3 Pressurizer Temperature Control

The temperature of the liquid in the pressurizer is monitored and controlled by three control loops.

When reactor power is increased, relatively cool water (the swell) flows from the HTS into the pressurizer. The temperature of this water is lower than the saturation temperature corresponding to the pressure in the pressurizer. The temperature control corrects this by controlling the ON-OFF heaters so that the water in the pressurizer is maintained about 3.5°C below the saturation temperature equivalent to the pressure set point. Under normal operating conditions the variable heater controls the HTS pressure.

When all four heat transport pumps are operating, the pressure in all reactor outlet headers is the same. With only one HTS pump operating, there will be a pressure differential between the two outlet headers in that loop. The header nearest the operating pump will be at the lower pressure. If this header, which now has a corresponding lower saturation temperature, is the one connected to the pressurizer, the pressurizer temperature will also decrease. As a result, the ON-OFF heaters would be turned on by the temperature controller. However, the heat transport pressure control system uses the highest of the outlet header pressures as its measure of actual pressure, and that pressure has not changed significantly. Therefore, as a result of the pump not operating, the corresponding temperature at which the heaters switch off is higher than the actual saturation temperature in the pressurizer and the steam bleed valves would be called upon to maintain pressure control. To prevent the heaters coming on under these conditions, the heater operating setpoint is made a linear function of the average of the pressures in the two reactor outlet headers connected directly to the pressurizer. This control loop operates only in the normal mode and is switched out during "solid" mode.

The temperature measurement is triplicated for reliability with median selection done by the computer. This temperature is also used for temperature correction of the pressurizer level used by the Heat Transport Inventory Control Program.

5.2.4 Pressurizer Heaters

The five pressurizer heaters consist of four ON/OFF heaters and one variable heater.

Each ON/OFF heater can be selected to be on, off, or auto. Under automatic control, the heaters will, in the normal mode of operation, be controlled by header pressure in conjunction with the pressurizer liquid temperature loop. With the handswitch in AUTO and the system in the "solid"' mode of operation (i.e., pressurizer isolated) the heaters are controlled by the pressurizer pressure.

The variable heater can be selected to be off, or controlled automatically.

Under all operating conditions (regardless of mode of operation and handswitches position) the heaters are disabled by computer if the level drops below the safe operating level for the heaters.

Heater low level protection relies on 48 V dc logic power. If 48 V dc is not available the heater should be shut off from the breaker.

After a DBE, the operator should ensure that all heaters are off by opening the breakers.

Failure of the variable heater means the pressure control will be by on-off heaters (at a lower pressure) until repair can be effected. The frequency of on-off cycling can be decreased by selecting some heaters off.

5.2.5 Reactor Outlet Header Wide Range Pressure Control ("Solid" mode operation)

During the "solid" mode operation the pressurizer is isolated and the control of heat transport system pressure is by means of feed and bleed valves. "solid" mode control is used by the operator to quickly pressurize or depressurize the heat transport system. It uses wide range instruments, the pressure transmitters in each of the four outlet headers have a range of 0 to 11.0 MPa(g).

Since the "solid" mode operation will not be used very often, the controls are not triplicated.

The highest of the four signals is selected and fed to two controllers, one for the control of the feed and bleed valves, the other for the control of the degassing valves.

As the system pressure drops, the feed valves move more open.

The valves between the outlet headers and the degasser-condenser are tripped open a very high pressure signal.

5.3 Pressurizer Controls

These control loops provide control of the pressurizer when it is isolated from the HTS, ie "solid" mode operation.

5.3.1 Pressurizer Wide Range Pressure Control

When the pressurizer is isolated from the heat transport system, i.e. in "solid" mode, its pressure is controlled by a wide range control loop. Since this control mode will be seldom used it has not been triplicated. The pressure transmitter range is 0 to 11 MPa(g).

The controller output is fed to both the steam bleed valves and the variable heater. The setpoint pressure is adjusted manually.

5.3.2 Pressurizer Pressure Relief Loop

The pressurizer is provided with two relief valves. They provide protection when the pressurizer is isolated from the heat transport system the heat transport relief valves are not available for its overpressure protection. Protection in this case is provided by two pressurizer relief valves. For reliability, a fully triplicated loop is used with current alarm units in each of the individual loops and 2 out of 3 logic.

When the pressurizer pressure rises above the setpoint, the pressurizer relief valves will open and remain open until the pressure drops below the setpoint. Alarms are provided through limit switch contacts when the valves are fully or partially open. CRT display of pressure is available on demand.

The range of the pressure transmitters is 6.0 to 12.0 MPa(g).

5.4 Degasser-Condenser Control

The degasser-condenser receives water and steam from;

- degassing flow taken from the two pump suction headers on the opposite side of the reactor to that of the HTS liquid relief valves.
- cooling and condensing spray taken from the discharge of the charging pumps
- steam bleed from the pressurizer
- liquid relief from the HTS

Normally the degasser-condenser can discharges its water to the HTS storage tank and the suction of the feed pumps after cooling by the degasser-condenser cooler. In extreme conditions, overpressure relief is to the fuelling machine vault floor. Any relief will be collected in the D2O recovery tank and pumped back to storage via the recovery pumps.

5.4.1 Degasser-Condenser Level Control

The D_2O level in the degasser-condenser is controlled by regulating the flow out of the condenser through two valves, each driven by its own level measurement. In order to prevent high temperatures in the storage and feed systems, temperature control loops override the level control signals to the valves. For redundancy the loops have separate power supplies and route separation. The valves are designed to fail closed. To prevent a loss of coolant after a DBE, the operator must ensure that the valves are closed.

The lower of the two level measurements, corrected for temperature is used to cut off the degasser-condenser heaters on low level.

The corrected levels are available for CRT display on demand. Annunciation is given when the heaters are disabled and when the degasser-condenser level is high.

5.4.2 Degasser-Condenser Pressure Control

Under normal operating conditions the degasser-condenser pressure is controlled by varying the flow of the cooling spray into the tank. There are two loops providing pressure control, differing only in the set point applied. The pressure setpoints are interchangeable without affecting control and alarm annunciations. The interchangeability of setpoints provides capability for having uniform wear of both spray valves. Each loop contains a wide range and narrow range pressure measurement.

The narrow range transmitter covers 0 to 6.0 MPa(g). The signal is used to;
- control the pressure by positioning the spray valve and operating the heaters.
- close the degassing valve on high pressure.

Annunciation is provided by the narrow range loops on high pressure, or on irrational spray valve position, ie if one or both valves are not in the position expected by the degasser-condenser pressure.

The wide range pressure transmitter has a range of 0 to 11.0 MPa(g). Its output closes the spray valve on very high pressure and activates annunciation.

5.4.3 Degasser-Condenser Pressure Relief

The principal cause of the degasser-condenser being over pressurized is;
- the inflow from the HTS feed pumps, either directly via the sprays
- indirectly via the heat transport system and the liquid relief valves.

Two spring loaded relief valves with pneumatic actuators provide over pressure protection for the degasser-condenser under two conditions. First, they provide protection for the degasser-condenser itself and are set to relieve at 10.6 Mpa(g). Second, their pneumatic actuators provide adequate overpressure protection for the main heat transport system during the unusual condition when only two heat transport pumps are operating and they are on the side nearest the reactor outlet headers that have the HTS liquid relief valves. Under this condition the liquid relief valves are at the lowest HTS header pressure.

When the pressure in the either of the other outlet headers rises above a setpoint, (11.27 MPa(g)), the degasser-condenser relief valves are tripped open through 2 out of 3 logic using signals from fully triplicated pressure loops. The range of the pressure transmitters is 6 to 12 MPa(g).

The degasser-condenser relief valves have a backup bottled air supply.

A CRT display of pressures in these loops is available on demand. A CRT annunciation will come on when any one of the relief valves is open by this pressure signal.

5.4.4 Degasser-Condenser Degassing Control

The term degassing is the name given to the process where by gas comes out of solution when the pressure in a system is reduced.In the heat transport system radiolysis of the coolant takes place while in the reactor core.The result is the formation of D_2 and O_2 gases.

Under normal conditions, in the heat transport system degassing will be generally confined to two areas:
- The D_2O storage tank
- the Degasser-Condenser (Bleed Condenser)

Both these areas have D_2O liquid in thermal equilibrium with the D_2O vapour above.

In the heat transport storage tank the cover gas is helium.But H_2/D_2 gas is also present due to the degassing of the radiolysis process.

A quantity of greater than 4% D_2 could cause an explosive mixture.
To control this the operator is required to purge the D_2 from the storage tank when ever the concentration of D_2 approaches a predetermined value (ie:2%). This process prevents the build up of D_2 gases in the storage tank and the possibility of an explosion.Chemical sampling of the storage tank is done on a routine basis by the chemical department.

In the Degasser-Condenser the cover gas is saturated D_2O vapour with some O_2, D_2/H_2 and fission product gases (Xe and Kr).

These gases come out of solution from the heat transport D_2O when it flashes to steam when entering the bleed condenser or via the steam from the pressurizer steam bleed valves.

To control this in the Bleed Condenser there is a take off to a system called the Off-Gas system.The Bleed Condenser is valved into the Off-Gas system and the gases are removed.

To control this in the heat transport system which contains a Degasser, a degassing flow is established to the Degasser-Condenser from the heat transport system or from the pressurizer bleed flow. The vapour gas mixture is directed to a vent condenser, then to the vapour recovery system.Hence the problem of gas accumulation in the Degasser-Condenser is eliminated.

5.5 CRT Displays

Process flows, temperatures and pressures for the pressure and inventory control system can be displayed on the CRT. In addition, bearing and winding temperatures, and vibration of process equipment can be displayed.

5.6 Alarms

All alarms are actuated through the computer in the form of an annunciation message on the CRT. In addition some important alarms are displayed on windows.

A window alarm is given on "very high" degasser-condenser pressure actuated by a current sensitive alarm unit which also gives a CRT annunciation and trips both spray valves closed.

A window alarm comes on if any one of the pressurizer isolating valves is not fully open while the reactor power is greater than 5% F.P.

5.7 Effect of Other Systems Malfunctioning

This section describes the effect on the inventory and pressure control system when other systems malfunction.

5.7.1 Loss of Instrument Air

The heat transport relief valves and pressurizer relief valves will fail open on complete loss of odd and even air supplies. On loss of odd or even air supply, the respective pneumatic valves (control valves and pneumatic isolating valves) go to their failure positions as shown in the system flowsheet.

5.7.2 Loss of Class IV Power

The pressurizer heaters are supplied from Class IV power. Upon loss of Class IV they would not be able to maintain the pressure. If the power cannot be restored, the operator will transfer to "solid" mode control and use the feed pumps to maintain heat transport system pressure.

The two feed pumps, which are on Class III power, will stop on loss of Class IV but will restart automatically when Class III power is available. Failure of the Class III power to the running pump would require operator action to bring in the other feed pump.

5.7.3 Loss of 120 V ac Class II Power

The output of controllers and manual control stations drops to zero on the failure of their 120 V ac supply. This results in the control valve going to its failure position shown in the flowsheet.

The current alarm units fail to the alarm condition when their 120 V ac supplies are lost.

The pressurizer and the degasser-condenser relief valves will fail open only if 120 V ac power supplies are lost to 2 out of 3 of their pressure control loops.

5.7.4 Loss of 48 V dc Power

Failure of the 48 V dc power supply results in the loss control for the pressurizer heaters and the feed pumps from the control room. Low level protection of the pressurizer heaters and the very low suction pressure trip for the feed pumps will be lost. Heaters should be turned off as soon as possible from the circuit breakers to prevent their destruction.
The degasser-condenser heaters will trip on failure of 48 V dc relay power.

The pressurizer relief valves and the heat transport relief valves fail open on a general loss of odd and even control power.

A general loss of odd and even control power would result in a reactor trip combined with loss of the low level pressurizer heater protection. The heat transport relief valves would fail open and cause the water level in the pressurizer to fall below the heaters, i.e., the pressurizer would empty. All the heaters would be called to operate due to the low pressure but they would not come on due to loss of 48 V dc power supply. Any heater that happened to be operating at the time would probably burn out. Of course, such a loss of power is extremely unlikely and in normal operation only the variable heater would be in operation so would be the only one affected.

5.7.5 Loss of 40 v dc Power

Failures in the 40 V dc power supply to the loops with computer control or with an alarm will be picked up by the computer rationality check. Loops for indication only do not have rationality check.

5.7.6 Loss of Computers

The functions of the boiler inventory and pressure control system are duplicated in the two computers. On the failure of the first computer (DCCX) the control functions are transferred to the other computer (DCCY). All computer controlled loops in this system can be transferred to manual computer keyboard control if a malfunction in the loop makes this necessary.

On failure of the computers, all computer contacts will open. This will cause the pressurizer and degasser-condenser heaters to shut off and the steam bleed valves on the pressurizer to close.

The pressurizer relief valves remain available since they are independent of computer with their setpoint about 410 kPa higher than that of the liquid relief valves. The liquid relief valves will open which may bottle up the degasser-condenser on temperature override and bring in the "very high pressure" window alarm (independent of computer).

On failure of both computers the operator must switch the system to the "solid" mode when the reactor power falls below 5% and the D_2O level in the pressurizer falls close to its zero power hot level. This provides the required control to the valves normally on computer control. If the operator does not switch to "solid" mode the HTS bleed valves will go closed and its feed valves will open. As the heat transport system shrinks (reactor is shutdown by its control devices) the pressurizer level will drop down to its zero power hot level. Then as the heat transport feed pump continues to operate, the level will start rising. This will result in adiabatic compression of the vapour space in the pressurizer and possible opening of the relief valves to pass water to the degasser-condenser. In turn the degasser-condenser relief valves may open, discharging may hot D_2O into the fuelling machine vault.
All CRT displays are unavailable.

5.7.7 Heater protection

For the protection of the heaters under all operating conditions (regardless of mode of operation and heater handswitches position) the heaters are disabled if the level drops below a certain setpoint. In "solid" mode, failure of the transmitter to low or the loss of 40 V dc power supply would bring the heaters on. This event will be annunciated by computer rationality check and operator should control the heaters manually. Overpressure protection is provided by the pressurizer relief valves and hardware logic.

6.0 Summary

- The HTS inventory between zero power hot and full power is constant. The shrink and swell from temperature change is accommodated by changing the level set point in the pressurizer to match the thermal conditions in the HTS.
- The HTS pressure is controlled during full power operation by controlling saturation temperature in pressurizer through action of heaters and steam bleed valves. The pressure is controlled over a narrow range.
- The HTS pressure can be controlled over wide range by feed and bleed valves and feed pumps without use of pressurizer. This is used during startup or when the operator needs to change the pressure over a wide range. This mode is not normally used at full power.
- The degasser-condenser receives water from the pressurizer steam bleed valves, the HTS liquid relief valves and returns water to the storage system which includes the feed pumps and the feed and bleed valves.

Shutdown Cooling System

Training Objectives

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

- Draw a simple flow diagram of the shutdown cooling systems for a CANDU 6.
- State the principal safety related functional requirement of the system.
- State two important process related functional requirements of the system.
- Describe for a CANDU 6, at least three alternative cooling paths for the heat transport system.
- Describe the principal difference between the CANDU 6 shutdown cooling system and the systems of the Bruce 'B' and Darlington stations.
- State the two main operating modes and for the 'cooling mode' describe the two pumping options.
- Describe for a CANDU 6 an abnormal cooldown of the HT system using the shutdown cooling system.

Table of Contents

1. Shutdown Cooling Systems

Normally the initial cooldown of the heat transport system (HT) is accomplished by using the steam generators and their associated Steam Discharge Valves (SDVs). At temperatures below 177˚C this process become ineffective. The CANDU 6 Shutdown Cooling System (SDCS) is provided to cool the Heat Transport System (HT) from 177˚C to 54˚C and hold the system at 54˚C for an indefinite period of time. It is also designed to provide core cooling with the heat transport system drained to the headers to permit maintenance of the boilers and the heat transport pump internals.

The shutdown cooling system is also capable of cooling the system from 260˚C, zero power hot, under abnormal (emergency) conditions.

The shutdown cooling system maybe used as a heat sink for cooling of the intact loop, in two loop CANDU 6 reactors, after a Loss of Coolant Accident (LOCA), and for alternative cooling in the event of a feedwater line break or a steam line break outside the reactor building.

In Ontario Hydro Bruce reactors, the cooling function of the HT system is assumed by two systems, the Shutdown Cooling System and the Maintenance Cooling System as described in Section 3.1. It should be noted in all operating CANDU reactors, initial cooldown is performed through the depressurization of the steam generators.

Generally station Operating Policies and Principles (O.P. & P.'s) will have the following requirements for the shutdown cooling system:

- The shutdown cooling system is rated as an "**Alternate Heat Sink** " for the purposes of the O.P. & P.'s.
- The shutdown Cooling system must be available when the heat transport system temperature is > 100˚ C.

2. Functional Requirements

2.1 Safety Related

The safety requirements of the shutdown cooling (SDC) system are as follows:

- To remove decay heat from the fuel following normal shutdown and after certain accident conditions (e.g. feedwater line break).
- To be qualified for Design Basis Earthquake (DBE) to ensure pressure boundary integrity of the HT system when the system is connected to the HT system.
- To be qualified to remain fully functional in the harsh environmental conditions caused by a LOCA, or a steam line break within containment.

This system is required to remove decay heat from the intact loop following reactor shutdown and initial cooldown.

2.2 Process Related

To cool the heat transport system after a shutdown and following an initial cooldown by steam rejection, to a temperature suitable for maintenance.

- To remove decay heat from the reactor during shutdown.
- To maintain the heat transport system at a temperature suitable for maintenance for any desired length of time.
- To provide a means of draining, refilling and controlling the level of the heat transport system to allow maintenance of the heat transport pumps and/or the steam generators.
- To be able to cool the core with the heat transport system drained to header level to permit maintenance to be carried out on the heat transport pumps and the steam generators.
- To provide purification of the heat transport system heavy water during all phases of SDC system operation, except when the heat transport system is drained to the headers.
- To be capable of cooling the heat transport system from 260˚C under abnormal conditions.
- To be capable of operation after failure of a main steam line inside containment.
- To retain pressure boundary integrity in the event of a Design Basis Earthquake.

3. System Description

The main function of the shutdown cooling system is to cool down the HT system and maintain it in a cold shutdown state as long as it is required by rejecting heat to the service cooling water system. The systems are also capable of lowering the level in the HT system so that maintenance can be carried out on the HT pumps and the heavy water side of the boilers. Each reactor adopts a different arrangement of pumps, heat exchangers, piping and valving to accomplish this objective.

3.1 System Configuration In Various CANDU Stations

A CANDU 6

The Candu 6 system consists of a pump and a heat exchanger at each end of the reactor. This reduces heavy water hold-up since long piping runs between the two reactor faces are not required. Each pump and heat exchanger connects between the inlet and outlet headers of both heat transport loops. Figure 3.1 shows the normal flows through the HT system with the SDC system poised (filled with D_2O but not pressurized). The design is such that cooldown can be achieved using either the heat transport pumps (flow direction is from inlet headers to the outlet headers via SDC heat exchanger and the SDC pump bypass, see fig. 3.2) or using the shutdown cooling pumps (flow direction is from outlet headers to the inlet headers, see fig. 3.3). In both cases, the pressure at the inlet headers is sufficient to force water through the core to the opposite outlet headers.

The shutdown cooling system is located inside the reactor building. The entire system is below the reactor header's level. The main isolating valves from the heat transport system are normally closed and the pump isolating valves are normally open when the reactor is operating. The system is kept depressurized and full of D_2O. (see fig. 3.1)

The cooldown using the heat transport pumps is shown in figure 3.2. In this mode some or all of the heat transport pumps are in service and they produce a reverse flow through the shutdown cooling heat exchangers. In this mode purification is supplied in the normal HT flows.

The longer term cooling with the shutdown cooling pumps in service is shown in figure 3.3. There is a reversed flow (approximately 55 %) passing through the heat transport pump and boiler with the remainder passing through the reactor core. When the reactor is shut down and the shutdown cooling system is operating, HT purification is available by taking the flow from the discharge of SDC pump and passing it through the tube-side of the interchanger, cooler, filters, ion exchange columns, shell-side of the interchanger and back to the suction of the SDC pump.

Spring loaded pressure relief valves on the pump suction header (see figure 3.1) provide overpressure protection during shutdown cooling isolation. During reactor shutdown, the shutdown cooling system is connected to the heat transport system and thus the overpressure protection which is provided for the HT system also provides overpressure protection for the shutdown cooling system.

B. BRUCE

The Bruce plants use two systems to achieve all shutdown cooling objectives. The first system is known as the "Shutdown Cooling System". This is a light water system and is used to lower the HT temperature to around 90˚C. The second system is known as the "Maintenance Cooling System". This is a heavy water system and is used to both lower the HT system temperature to the mid 50˚C range and to lower the HT level for maintenance.

The Bruce A and B shutdown cooling system takes advantage of the separate preheater in the HTS for shutdown cooling (Bruce B is shown in figure 3.4). Heat rejection is via the four heat transport system preheaters to a closed demineralized light water loop comprising the shell side of the preheaters, the tube side of the two shutdown coolers and the two shutdown cooling pumps. Heat is rejected from this closed loop to service cooling water on the shell side of the shutdown coolers. The shutdown cooling system is a part of the feedwater system (H_2O) and designed to the same standards. The normal feedwater flow path is interrupted by closing motorized valves. The HT system flow is maintained by the HT pumps. Hence the HTS must remain pressurized and the heat transport circulating pumps must be in service. The system must be capable of removing both core and pump heat.

If it is necessary to depressurize and/or drain the Bruce B HT system, the shutdown cooling system is not adequate. A separate circuit called the maintenance cooling system is provided (shown in figure 3.5). The system consists of a one-loop circuit containing a heat exchanger and two pumps connected between the reactor outlet headers and inlet headers. This is a heavy water circuit. Heat is rejected from the HTS in the heat exchanger to service cooling water. The maintenance cooling system is normally brought into operation after the shutdown cooling has reduced the reactor outlet header temperature to approximately 93˚C. However, it is designed, for a limited number of times, to cool the HTS from 177˚C.

The above applies to both Bruce A and B, but Bruce A has a different style of boiler where the four boilers (steam generator) are joined by a common steam drum, at each side of the reactor.

C. Darlington A

The Darlington A shutdown cooling system (figure 3.6) is similar to the CANDU 6 shutdown cooling system with the two circuits combined into one. Heavy water drawn from all the outlet headers to three pumps in parallel, rejects heat to service cooling water in two heat exchangers in parallel and then returns to all the inlet headers. An arrangement of 3 x 50% pumps were chosen instead of 2 x 100% pumps to reduce upset in the event of a pump failure, i.e., flow is not completely stopped when two 50% pumps are operating and one fails.

D. HT Low Level Operation

From the above it can be seen that the CANDU 6 has the two SDC systems in series, while Bruce and Darlington use a parallel design. The advantage of the CANDU 6 is in the reduced length of piping and consequent reduction in heavy water holdup. The Hydro reactors use a parallel pump, heat exchanger combination with all ROH headers tied together and all RIH headers tied together. This arrangement has the advantage of reduced upsets and/or D_2O spills while on level control.

Figure 3.1
SDC System out Of Service

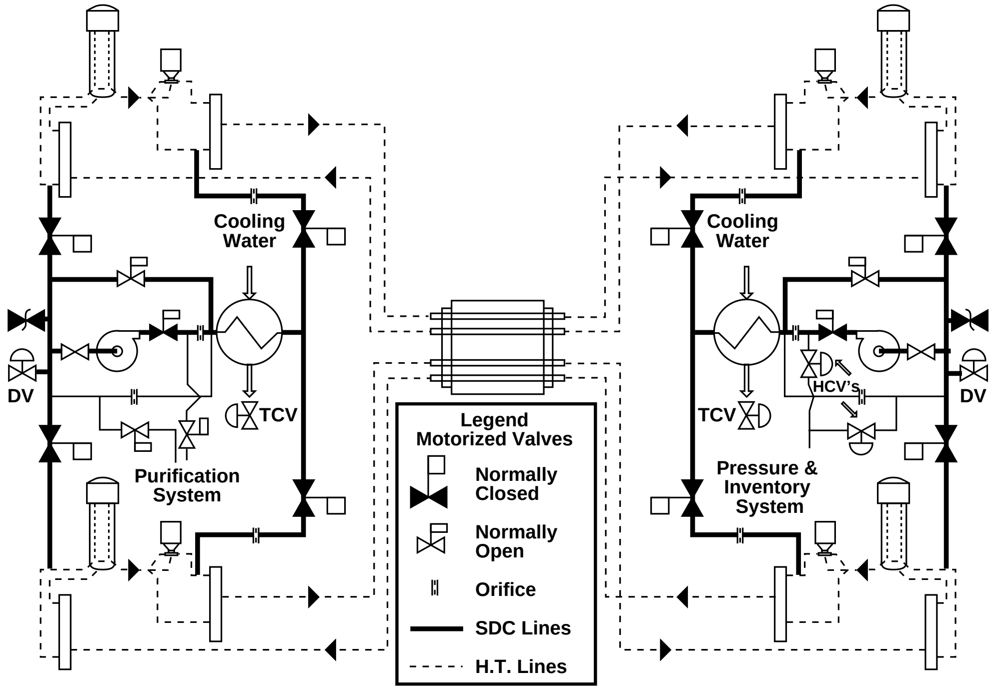


Figure 3.2
SDC System - Flow supplied by heat transport circ. pumps

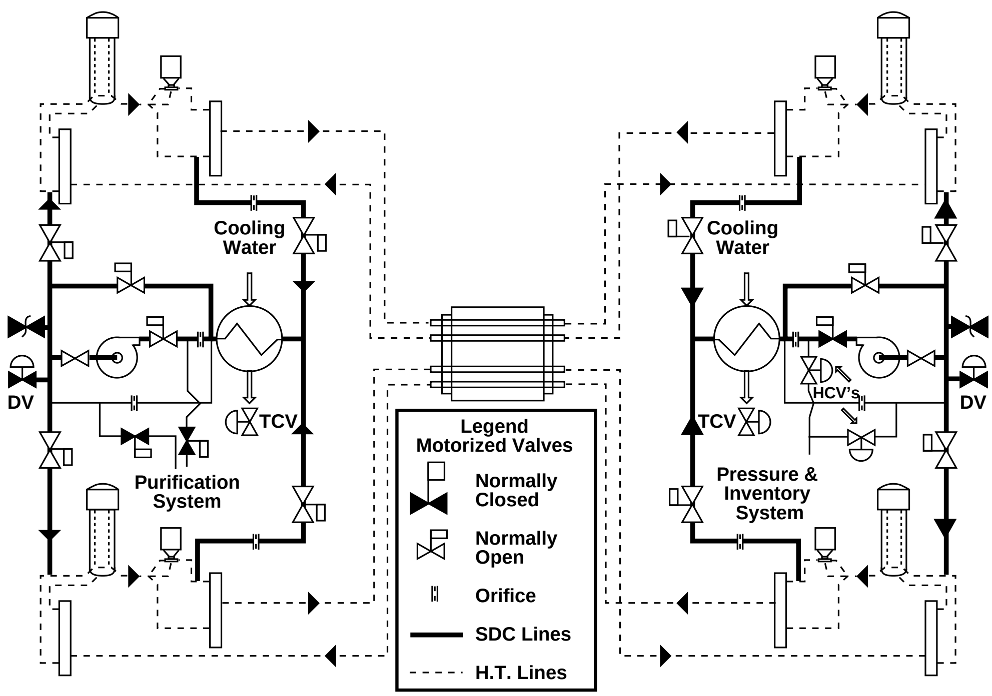


Figure 3.3
SDC System - Normal Operation

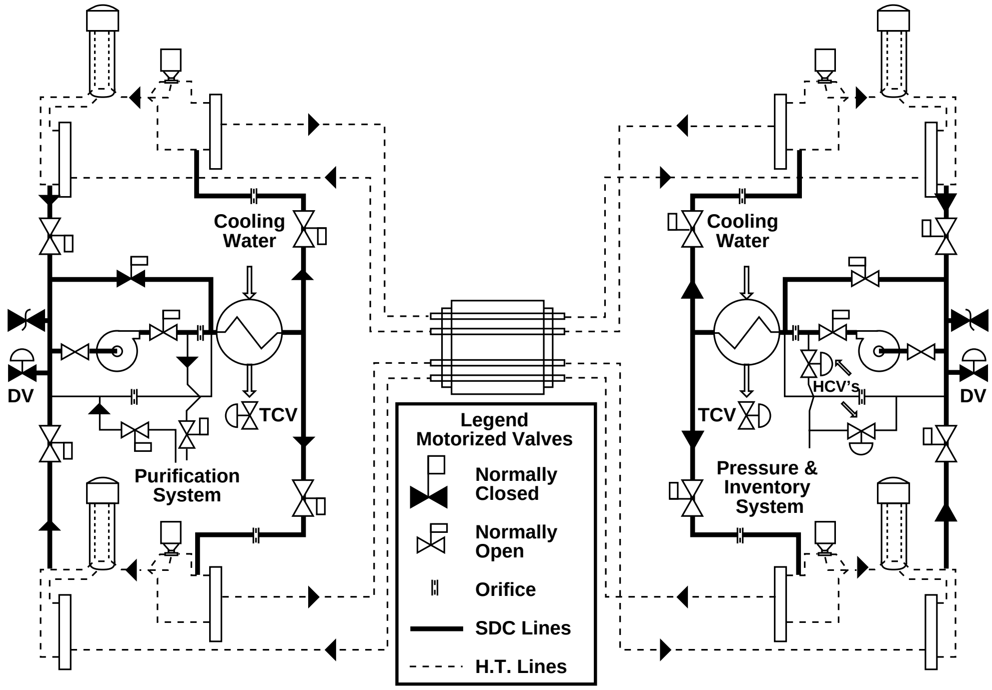


Figure 3.4
SDC Bruce NGS

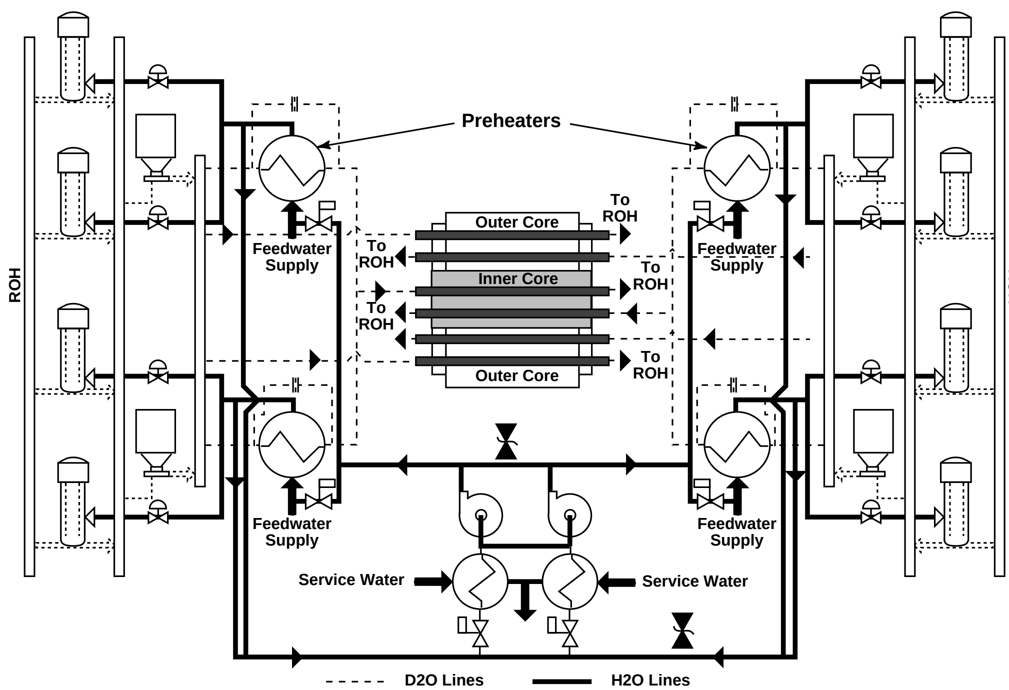


Figure 3.5
Maintenance Cooling System - Bruce B

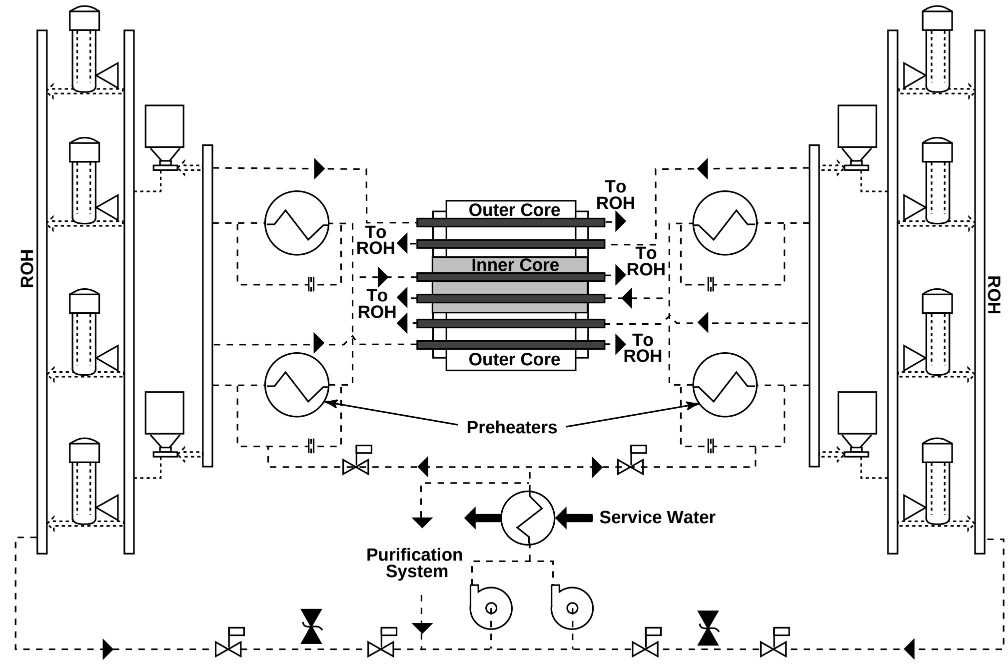
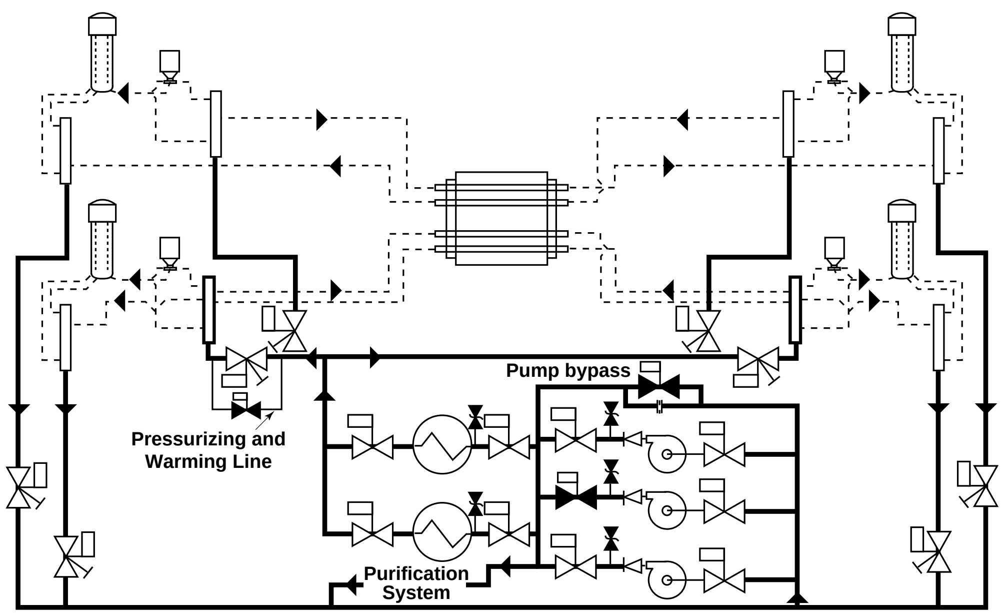


Figure 3.6
SDC System - Darlington NGS



4. Major Equipment/Components (CANDU 6)

Major components of the CANDU 6 shutdown cooling system are the SDC pumps, the SDC heat exchangers, valves, piping, and freeze jacket assemblies.

4.1 Shutdown Cooling Pumps

A 100% capacity vertically mounted, centrifugal type, electrically driven pump is provided for each shutdown cooling loop. The rated capacity of the pump is 182 *l*/s at a head of 73.8m. The drive motor for the pump is rated at 300 HP. The pump/motor unit is normally accessible during reactor operation for inspection and repair. During reactor shutdown, the SDC pumps are accessible for inspection but not for significant repair.

The SDC pumps are designed to meet the requirements of ASME Code Section III. These pumps are equipped with two primary seals and one backup seal in series. The backup seal is designed to prevent tritium leakage to the reactor building atmosphere. The backup seal will also prevent gross leakage to atmosphere, with the pump either stopped or coasting down after the primary seals have failed. The seals are assembled as a single cartridge assembly for fast removal and replacement.

The two primary seals are face type seals, each suitable for operation at full pressure. During normal operation, each of the two primary seals operates at approximately one-half pressure. In the event that one of the primary seals fails, the remaining seal will operate at full pressure until the pump can be stopped for seal repair. The backup seal is located immediately above the pump cover flange outside the pressure envelope of the pump and is designed to operate without lubrication.

The heavy water for the seal cooling is supplied by the SDC pumps themselves. An auxiliary impeller mounted on the pump shaft, above the pump bearing, recirculates a flow of D_2O in a loop through the pump bearing for cooling and lubrication, then through an external heat exchanger before it is injected into the pump's two seal stages, and out of the pump gland through a seal staging valve. To compensate for the seal staging flow, an equivalent amount of D_2O from the pump casing flows into the recirculation loop and is cooled by the SDC heat exchanger before being injected into the pump's first seal cavity.

4.2 Shutdown Cooling Heat Exchangers

The SDC heat exchangers are horizontal U-tube and shell type heat exchangers. The SDC heat exchangers are used to cool D_2O flowing in either direction between the inlet and outlet headers of the heat transport circuit.

The tube-side of the heat exchanger is designed and constructed to meet as Class 1 requirements of the ASME Code Section III whereas the shell-side is designed in accordance with the Section VIII, Division 1.

Each heat exchanger is cooled by a recirculated cooling water (RCW) flow of 303 l/s . This flow is also available under Class III power conditions. The supply pressure is 0.69 MPa(g) at the reactor building wall, which ensures that high tube-side temperatures do not lead to localized boiling in the shell side of the heat exchanger.

4.3 Valves

All the shutdown cooling system valves are designed to withstand full system temperature and pressure. Except for valves which have live load double packed stem seals, all valves are of the bellows seal type. The isolating valves are wedge gate valves with electric operators. Check valves are of the swing disc type with metal to metal seat.

All valves have been designed as per ASME Section III, Class 1 requirements and are rated at the appropriate ANSI rating.

Those valves required to function and retain pressure integrity during or immediately following a DBE have been qualified to Seismic Category B. The rest of the valves in the system are qualified to Category A.

4.4 Piping

Piping is fabricated and installed in accordance with AECL Specification NCS-5 using schedule 80 carbon steel for piping 4" and below, schedule 120 for 6" only, and using schedule 100 for piping 8" and over. The material of the piping is carbon steel ASME SA106 Grade B.

Because the head loss through the reactor is negligible while friction loss through the purification system is appreciable, an orifice is provided in the pump discharge line so that there will be sufficient pressure to force flow through the purification system when the shutdown cooling system is operating and the heat transport system is full; and to force flow through the pressure and inventory control system when the shutdown cooling system is operating and the heat transport system is being drained.

4.5 Freeze Jacket Assemblies

Since the lines on the primary heat transport system side of the shutdown cooling system isolating valves can not be drained, provisions for forming freeze plugs in the lines are provided to isolate the valves from the main circuit to permit valve maintenance. Freeze plugs are effective at differential pressures of about 0.34 MPa or less so that maintenance on the isolating valves is only possible during reactor shutdown with heat transport system cooled and depressurized.

Both external freezing jackets and internal freeze plugs forming equipment are provided, as either one of them by itself may not be adequate to freeze the line if there is flow through the valve.

5. Layout

The shutdown cooling system is located inside the reactor building.

There is a shutdown cooling pump and a heat exchanger at each end of the reactor, connected between the inlet and outlet headers of both heat transport loops. The system is entirely below header level. The system's isolation valves from the HT system share a steel platform with the emergency core cooling system isolation valves on each reactor face. All valves can be maintained using the same overhead monorail. The heat exchangers are located on separate platforms underneath the platforms on which the valves are supported. These platforms are not accessible during normal operation.

The SDC pumps are located in pits within rooms adjacent to the valves' platforms on each side of the reactor. The pumps are shielded using a combination of concrete and steel slabs, which permit accessibility during plant operation.

The system consists of eight 10" lines, which interconnect the reactor inlet headers and the reactors outlet headers, for the two heat transport loops at each end of the reactor. The piping from the reactor outlet headers is routed to the pump suction, from the pump to the heat exchanger inlet, and finally to the reactor inlet headers.

A connection is provided to the HT purification system to maintain D_2O purity when the shutdown cooling system is in operation. Lines are connected to the pressure and inventory control system to maintain constant level in the headers when the system is drained to the headers for maintenance of the heat transport pumps or boilers.

Manual valves are provided for isolating the shutdown cooling pumps whenever required. Eight motorized isolation valves are provided to isolate the hot and pressurized heat transport system from the cold and depressurized shutdown cooling system during normal reactor operation.

There is a local discharge pressure gauge for each shutdown cooling pump to provide information during commissioning and give an indication to the operator if the system is pressurized or not.

Orifices, downstream of the shutdown cooling pumps, are provided so that there will be sufficient pressure to force D_2O through the purification system, and the pressure and inventory control system.
A by-pass line with orifices is provided to by-pass the shutdown cooling pumps during all modes of system operation. This by-pass line helps to limit the cooling water temperature rise in the heat exchangers.

Drain lines to the D_2O collection system are provided.
Emergency Core Cooling System connections are provided between the reactor headers and the shutdown cooling isolation valves.

6. Control, Monitoring And Diagnostics

Control and instrumentation in the system are designed to:

- provide control for cooing the primary heat transport (HT) system from 177˚C to 54˚C
- control the HT system temperature at 54˚C
- indicate and control the level of the depressurized HT system to permit maintenance to be carried out on the pumps and boilers
- control and monitor the operation of the shutdown cooling pumps
- monitor the operation of the heat exchangers
- provide manual control of the motorized valves
- provide auto-manual control of the SDC pump motorized bypass valves
- warn the operator of abnormal conditions in the system
- automatically open pump by-pass valve on the tripped pump if one of the pumps should trip during cooling while the heat transport system boilers are full or partially drained
- automatically open both pump by-pass valves should either one of the pumps trip during cooling while the heat transport system is drained down to the reactor headers
- operate properly inside the reactor building after failure of a main steam line inside containment.

6.1 Temperature Control

Temperature control loops, downstream of the heat exchangers (on the shell side), are used to control the cooling water flow through the heat exchangers when the shutdown cooling system pumps are operating. An alarm is provided if the temperature differences between the inlet and outlet of the heat exchanger is less than or equal to 10˚C. The alarm is conditional on shutdown cooling pumps running to prevent spurious alarms during normal reactor operation when heat transport pumps are running and shutdown cooling isolation valves are closed. When the shutdown cooling system is in the "alternate" mode, the temperature in this loop will be sufficiently above the setpoint of the temperature controller that the cooling water control valve will be held fully open. The maximum cooling water flow is needed for the "alternate" mode for maximum cooldown.

6.2 Level and Inventory Control

The heat transport system can be partially drained. Means have been provided to control the level in the heat transport system and to allow for draining down to header level. Lines are provided in order to drain the primary fluid between the primary heat transport pumps and the boilers. These lines are connected to the primary heat transport pump suction lines and permit nitrogen to enter the boiler leg, breaking the syphon effect and thus ensure proper draining of the boilers. This is because the only system vent is via the heat transport system pumps.

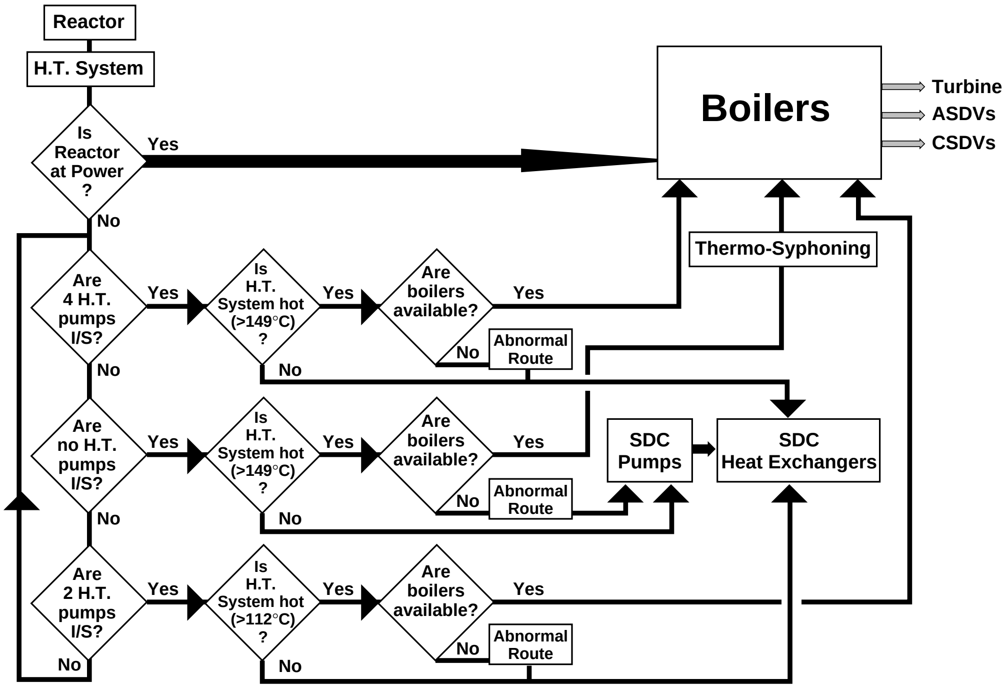
As the level approaches the desired value, the level in the heat transport system is maintained by using manually adjustable control valves. These valves are connected to the D_2O feed pump suction line. If the level is below the required point, then the D_2O makeup valve is opened thus directing the flow from the storage tank to the heat transport system. Since the D_2O storage tank is at a much higher elevation than the headers, D_2O will flow by gravity into the system. On the other hand, if the level is above the desired value the flow is directed from the heat transport system to the storage tank.

Level measurements are provided on the reactor headers and on the lines between the boiler outlet and pump suction.

7. System Operation

There are several flow paths for cooling down the HT System. They are illustrated in Figure 7.1 and discussed below.

Figure 7.1
HT Alternate Cooling Paths



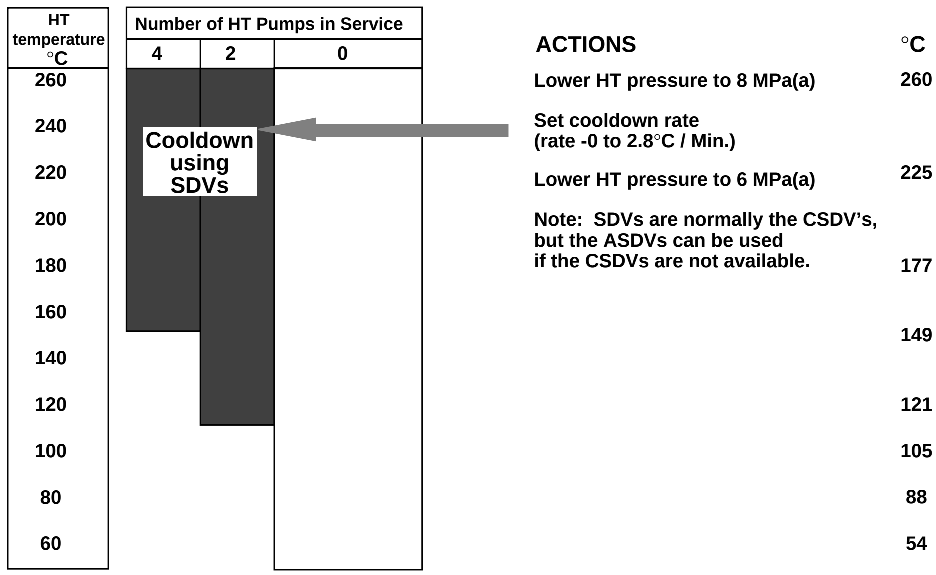
Heat Transport Pressure Changes on Cooldown

After reactor shutdown, the HT system is at 10 MPa(a) and 260˚C. In order to limit the potential for pressure tube rupture, the reactor cooldown procedure requires the operating pressure of the heat transport system to be reduced from 10 MPa(a) to 8 MPa(a) before the start of reactor cooldown. Boiling is not a problem as the temperature of the reactor outlet header (ROH) is now close to the reactor inlet header (RIH). The lowering of HT system pressure increases the critical crack length in a pressure tube. Lower pressure is obtained by lowering the setpoint in the pressure and inventory control (P&IC) system.

Cooldown at 8 MPa(a) from 260˚C is achieved by steam discharge through the CSDVs (or the ASDVs if the CSDVs are not available). When the HT system is cooled below 225˚C, the HT system pressure setpoint is lowered from 8 MPa(a) to 6 MPa(a). Normal mode of pressure control continues to be used.

Cooldown with SDVs Only (initial cooldown phase)

Figure 7.2
Cooldown of HT System via SDVs Only



As can be seen from fig. 7.2 cooldown to the 149˚ to 112˚ C range can be accomplished by using the SDVs (CSDVs and/or ASDVs) without the use of the shutdown cooling system. This is the normal process used prior to placing the shutdown cooling system into operation.

The shutdown cooling system is normally full of D_2O and is isolated from the HT system by eight valves. Prior to opening these valves and connecting the shutdown cooling system to the HT system, the operator should ensure that the shutdown cooling system is completely filled with D_2O via the purification connection.

The shutdown cooling pumps must be used when the heat transport system is depressurized.

7.1 Operating Modes

There are two main operating modes, a) Standby with the reactor operating from 0 - 100% power and b) Cooling with two options for circulating the D_2O; b_i) using the Heat Transport Circulating Pumps, or b_{ii}) using the Shutdown Pumps.

System with Reactor at Power

During normal operation with the reactor at power, the shutdown cooling system is isolated from the HT system. It is kept full with heavy water at 38˚C and pressure at or just above atmospheric. Because there is a large pressure differential across the shutdown cooling system isolation valves there may be some leakage past the valves. Lines are provided to drain this leakage to the leakage collection system.

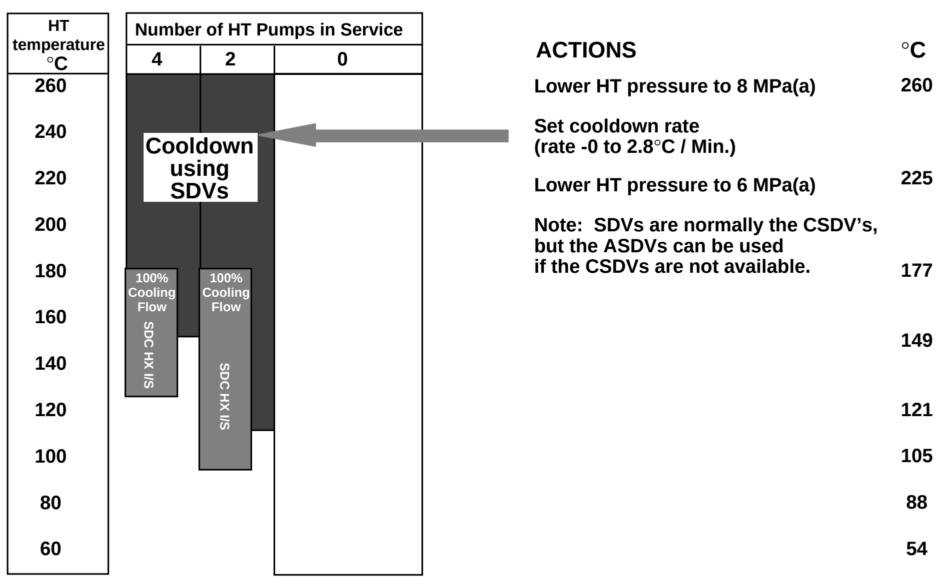
Cooldown Options (Reactor at 0% Power)
There are two normal cooldown options available after a reactor shutdown. The initial phase of the two options is similar and involves the use of the SDVs to lower the HT system temperature from 260˚C at various rates up to a maximum of 2.8˚C/min During this phase, the HT pumps circulate the coolant through the steam generators.

7.2 Option 1 (using the HT pumps)

After the initial cooldown phase, cooldown using HT pumps and the SDC heat exchangers can be initiated from a HT system temperature of 177˚C . However, it only reduces the HT system temperature effectively to 121˚C with four HT pumps being used or 88˚C with two pumps being used. Further cooldown has to be carried out through the use of the SDC pumps and heat exchangers (option 2). The normal temperature ranges are illustrated in figure 7.3 below. As can be seen a lower minimum equilibrium temperature can be reached by only using two HT pumps (pump heat input down by a factor of two).

These pumps can also be used without the benefit of initial cooldown see section 7.5.2 **Abnormal Cooldown** using HT pumps and SDC HX's (boilers unavailable).

Figure 7.3
Cooldown of HT System via SDVs and SDC HXs



Pneumatic drain valves ("DV"in figure 3.1) are closed. The SDC pump discharge isolating valves are opened and then the reactor outlet header isolating valves are opened to pressurize the SDC system. The inlet header valves are opened one at a time to allow flow to pass through the SDC heat exchangers. Cooldown is continued by using the heat transport pumps to circulate the heavy water from the reactor inlet headers to the outlet headers via the SDC heat exchangers, the SDC pumps and the SDC pump bypass lines. Reverse flow through the SDC pumps and recirculation line restriction orifices does not cause boiling on the secondary side of the heat exchangers. During this mode of operation, the anti-reverse ratchet (anti-rotation device) on the SDC pump motor shaft prevents reverse rotation.

The heat transport system is cooled down from 177˚C to 121˚C using four HT pumps, and to about 88˚C using two HT pumps. At HT system temperature below 88˚C the heat input from the heat transport pumps significantly reduces the rate of cooldown. The SDC pumps must then be used for further cooldown to 54˚C.

Because of design deficiencies, the configuration of the system has changed from that of figure 3.1 to figure 3.2. (see the discussion below on maintenance cooling)

The heat transport pumps circulate the heavy water around the heat transport system with a portion bypassing the core through the shutdown cooling system where the heat exchangers provide the necessary heat sink.

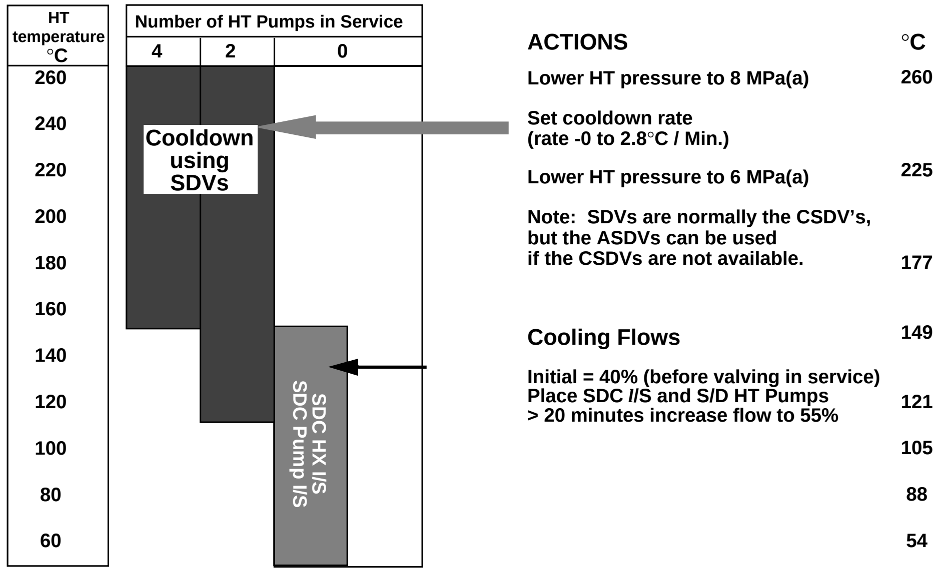
The flow through the shutdown cooling system depends on the heat transport system header to header pressure drop. This increases slightly as the heat transport system temperature decreases, since the volumetric flow is constant.

7.3 Option 2 (using the SDC pumps)

In the initial cooldown phase, the HT system temperature is normally brought down to 149˚C first through the CSDVs. The SDC pumps and heat exchanger are then brought into service and cooldown to 54˚C is then carried out. The normal temperature ranges are illustrated in figure 7.4 below.

These pumps can also be used without the benefit of initial cooldown see section 7.5.1 **Abnormal Cooldown** using SDC Pumps (boilers unavailable).

Figure 7.4
Cooldown of HT System via SDVs and SDC System



Cooldown from 149˚C (300˚F) and placing the SDC System in service

After the initial cooldown phase with 4 HT pumps in service the HT system will be in the 149˚C range.

In order to warm up the SDC system loop and to reduce the subsequent thermal shock in the reactor inlet headers when the SDC mode is brought into operation, the following procedure is followed:

While the SDC system is still isolated, the cooling flow to the two SDC heat exchangers is brought up to 40% full flow.

After that the inlet header SDC system isolation valves are opened to pressurize the system, then the outlet header SDC isolation valves are opened. D_2O flows from the reactor inlet headers via the SDC heat exchanger to the reactor outlet headers. Then all the reactor inlet header isolation valves are closed. The SDC pumps are subsequently started and flow is recirculated around the bypass lines.

The heat transport pumps are tripped and one minute later the SDC system isolation valves at the reactor inlet header are opened one at a time to bring the SDC system into operation (if a header valve fails to open,the heat transport system can reject decay heat by thermosyphoning).

In approximately twenty minutes after the SDC system has been valved in (the SDC heat exchanger outlet temperature reaches approximately 105˚C). The cooling water flow is increased from 40% to 65% full flow.

Twenty minutes after the cooling water valves have been modulated to 65% (the SDC heat exchanger outlet temperature reaches approximately 85˚C , full cooling water flow can be provided to continue the cooldown to 54˚C.

A dwell time of twenty minutes is necessary to allow the heat system components to reach a steady state temperature before the next cooldown phase takes place.

Cooldown from 121˚C and placing the SDC System in service
After the initial cooldown phase 2 HT pumps at one end of the reactor are shutdown and the HT system can now be cooled to the 121˚C range as the pump heat input has been reduced.

In order to warm up the SDC system loop and to reduce the subsequent thermal shock in the reactor inlet headers when the SDC mode is brought into operation, the following procedure is followed:

While the SDC system is still isolated, the cooling flow to the two SDC heat exchangers is brought up to 40% full flow.

The SDC circuits are valved in. The flow through the SDC circuits are opposite, ie; on the side of the reactor with the operating HT pumps the flow is reversed as in a) above, while the other side the flow is in the normal direction. The SDC pump freewheels. When it is desired to place the SDC pumps in service the two remaining HT pumps are tripped and the SDC pumps are placed in service.

After the SDC heat exchanger outlet temperatures lowers as in a) above the cooling flows are increased as above.

7.4 SDC Pumps in service - Flow Patterns and/or Partial Draining

When the SDC pumps are used at low pressure (0 - 0.34MPa) the HT system core temperature should be below 77˚C to avoid boiling if a one pump trip were to occur. In order to reduce the ingress of tritium into the reactor building, the temperature should be as low as practical to reduce the D_2O vapour pressure.

a Heat Transport System Full and at Low Pressure
In this case, the shutdown cooling pumps circulate water to the reactor inlet headers but about 55% of the total flow bypasses the core via the heat transport boilers and pumps. Because of the core by-passing, the temperature rise across the core is about 81˚C at 2% decay power although the bypass flow through the heat transport pumps and boilers reduces the temperature at the outlet headers. It is therefore necessary to keep the system pressure at a high enough value at the ROH to avoid channel boiling and possible flow stagnation. Flow stagnation in individual channels is possible at low pressure because, at the very low driving head available, any boiling in the channels will result in a greatly increased drop in the channels and outlet feeders.

In the event of a failure of one of the shutdown cooling pumps in this mode of operation, the failed pump is automatically by-passed by opening the corresponding by-pass valve to the 40% open position. This will increase the outlet header temperature slightly but does not result in any boiling in the channels.

b Heat Transport System Partially Drained

Core by-passing can be eliminated by lowering the level in the heat transport system. For this mode to work the syphon effect through the boilers must be broken by adding gas (nitrogen) into the boiler tubes and the level in the boiler can be lowered. There are two different modes of operation. In both cases, the heat transport system core outlet temperature should be below 77˚C to avoid any possibility of boiling during draining. The operator selects the "LOW LEVEL" position to enable the high level alarms.

1. Heat Transport System Not Vented

This mode of operation can be used for long term cooling with the heat transport system depressurized. The outlet header pressure set-point is ramped down to about 340 Kpa and the heat transport feed valves are then closed by placing them on manual control. The shutdown cooling system valve (connected to the purification system) is then opened and the heat transport bleed valves are opened by placing them on the manual control. To assist in draining the system, nitrogen is added through the HT pump glands as the D_2O is pumped, by the shutdown cooling pumps, to the D_2O storage tank via the pressure and inventory control system. As the level approaches the required value, the valve is closed. Immediately following this closure, the bleed valves are closed. The HCV valves are manually adjusted to obtain and maintain the correct heat transport system level. By partially draining the system when the temperature has been brought to the desired value, core bypassing is avoided.

In the event of a failure of one of the shutdown cooling pumps in this mode of operation, both bypass valves are opened to the 40% open position. This increases the outlet header temperature but does not result in any boiling in the channels.

2. Heat Transport System Vented

This mode of operation is used when a pump seal must be changed on a heat transport pump. The level is lowered to below the pump bowl but the boilers need not be fully drained.

When level indication loops indicate that the level in the pump suction line is below the pump, draining must be stopped.

All four heat transport pumps must be vented.

The level in the boiler will be higher due to the partial vacuum in the upper portion of the U-tubes. Prior to removing the seals, the heat transport pump shaft is dropped to provide a seal adequate for a pressure differential of 102 Kpa to guard against inadvertent spilling of D_2O. The bypass valves operate on failure of a shutdown cooling pump as noted in (1) above.

c. Heat Transport System Drained to Header Level
When boiler manhole covers have to be removed, or a pump shaft has to be removed, then the heat transport level must be dropped to header level.

Before the boiler manhole cover is removed the pump bypass valves must be opened to 33% of their open position and the SDC pump on the same side of the boiler is then tripped manually. These steps are necessary to eliminate the possibility of D_2O flooding the boiler head when bungs are being installed. Once the boiler manhole is opened, the two primary inlet nozzles of the boiler are blocked with portable inflatable pipe bungs. After the bungs are installed and before boiler maintenance is carried out the tripped SDC pump is restarted.

During the course of maintenance on the boiler, if a SDC pump becomes unavailable, both pump bypass will remain open such that no detrimental rise in the D_2O level in the boiler head will occur. The failed SDC pump can then be isolated for repair if necessary and heat transport system maintenance can be continued in this mode. This measure ensures flow through the core.

7.5 Abnormal Modes of Operation

7.5.1 Abnormal Cooldown from 260˚C Using The Shutdown Cooling Pumps

This is a special case in which the CSDVs, the ASCVs and the heat transport pumps are not available to cool the system down to 177˚C .

With the heat transport system full, the shutdown cooling pumps bypass valves are kept in their normally closed position. The shutdown cooling pumps are started and the flow circulated around the small by-pass lines for about 1 minute. The main shutdown cooling isolation valves are then opened and the cooldown is continued as described in cooldown from 149˚C. see section 7.3. (see fig. 7.5)

7.5.2 Abnormal Cooldown from 260˚C Using The Heat Transport Pumps and SDC HXs (Boilers Unavailable)

This is a case in which neither the CSDVs nor the ASDVs are available to cool the system down from 260˚C to 177˚C and hence the heat transport pumps are used to cool the system from 260˚C to 149˚C or 88˚C as described in section 7.2. (see fig. 7.5)

7.5.3 Crash Cooling

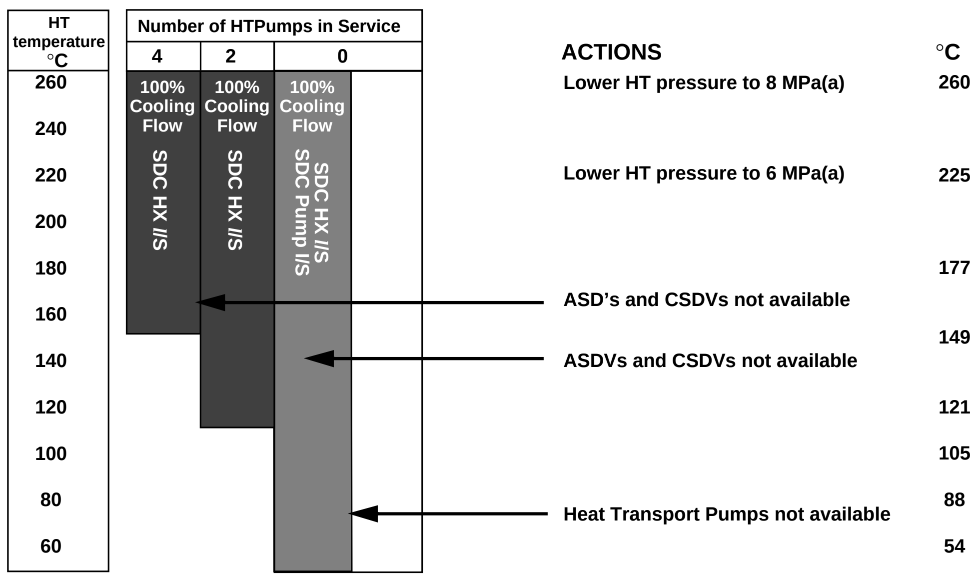
Crash Cooling is the name given to the operation in which the secondary side (steam generators) is depressurized at the maximum rate in order to place the shut down cooling system in service. This is a very serious situation and requires the shut down cooling system to become the primary heat sink.

Crash Cooling could be done by the operator by simply opening the steam discharge valves. This however is not a normal procedure and is not done unless called for in the station procedures.

Crash Cooling is called upon only when the emergency core cooling system is initiated. At this time the E.C.C. automatically opens the steam discharge valves and the secondary side pressure is reduced to allow shut down cooling to be placed in service.

Figure 7.5

Abnormal Cooldown of H. T. System



8. Protection Against Overpressure

Overpressure Protection

During normal reactor operation, the shutdown cooling system is cold and depressurized and is isolated from the hot and pressurized heat transport system. Relief valves provide overpressure protection. During reactor shutdown, the shutdown cooling system is connected to the heat transport system and thus the heat transport system overpressure protection provides overpressure protection for the shutdown cooling system as well.

When the pressurizer is connected to the heat transport system the reactor safety systems in conjunction with the cushioning effect of the steam in the pressurizer provides the necessary overpressure protection for the heat transport system.

The liquid relief valves also act as an overpressure protection device during this period. When the reactor is shutdown, the pressurizer is disconnected from the heat transport system and the liquid relief valves alone provide overpressure protection for the system.

9. Interdependencies with other Systems

A number of other systems are required to function for the successful operation of the SDC system as described below.

9.1 Heat Transport System

The HT system supplies reactor coolant to the SDC system for cooldown and decay heat removal purposes. It also maintains the reactor coolant level at a desirable setpoint by draining or filling via the SDC system during maintenance mode operation for heat transport pump or steam generator maintenance.

Provides overpressure protection for the system and system components when the SDC is connected to the HT system.

9.2 Pressure and Inventory Control System

In the heat transport partially drained mode of operation, the level in the heat transport system is maintained by using manually adjustable control valves. These valves are connected to the D_2O feed pump suction line which are part of the Pressure and Inventory Control System. If the level is below the required point, then the D_2O makeup valve is opened thus directing the flow from the storage tank to the heat transport system. Since the D_2O storage tank is at a much higher elevation than the headers, D_2O will flow by gravity into the system. On the other hand, if the level is above the desired value the flow is directed from the heat transport system to the storage tank. The storage tank is also part of the Pressure and Inventory Control System.

9.3 Heat Transport Purification System

During normal reactor operation the purification flow is taken from upstream of the shutdown cooling isolation valves and is returned back to the heat transport pump suction.

When the reactor is shutdown and the shutdown cooling system is operating, purification is available by taking the flow from the discharge of the shutdown cooling pump, passing the flow through the interchanger, cooler, filters and ion exchange columns, and back to the suction of the shutdown cooling pump..

Purification is not available when the heat transport system is partially drained. This results from the difficulties encountered in level control for this mode, if purification is made available. However, this mode of operation is not likely to be frequent or of extended duration. In addition, normal shutdown cooling and purification can be resumed once the heat transport system is refilled and before the reactor is restarted.

9.4 Heat Transport D_2O Sampling System

The purpose of the D_2O sampling systems is to permit the operating staff to extract representative specimens of process fluids to evaluate the performance of various systems. A determination of the heavy water isotopic concentration, the concentrations of dissolved solids, suspended solids, and chemical impurities is required to permit the purification and chemical treatment systems to be operated to their maximum potential. In addition, a knowledge of system radioactivities is acquired, which assists in process and radiation protection.

9.5 D_2O Leakage Collection System

This system collects heavy water leakage from double-packed valve stems, pump seals and inter-gasket cavities from the SDC as well as from other HT and auxiliary systems. It is also used to collect drainage and to provide a means of venting equipment in these systems.

9.6 Electric Power Supply Systems

The SDC pumps and motorized valves are supplied from the Class III power bus. On a loss of Class IV power, the Recirculated Cooling Water (RCW) pumps and the SDC pumps trip. As a result, the SDC system becomes unavailable for about three minutes. After three minutes the RCW pumps and SDC pumps are restarted on Class III power and the SDC system becomes available again to cool the HT system. Class II power is used to supply Instrument power in the system.

The Emergency power supply meets the following requirements:

a. To meet the two group separation requirements.
b. To provide the capability of long term decay heat removal and for Group 2 post accident monitoring.
c. To cater for Site Design Earthquake following a postulated loss of coolant accident, the EPS system must provide an alternate power supply to the ECC pumps.
d. A target unavailability for the supply of power from the EPS system to any load is 10^{-2}.
e. To allow a simple design, the specified starting time for the EPS generators is 20 minutes after the common mode event.

9.7 Recirculated Cooling Water System

This system provides normal cooling water to the SDC heat exchangers and pump motor bearings. On loss of RCW, the SDC system is not available for cooldown of the HT system. In this event the steam generators are established as the heat sink.

10. Potential for Radioactive Release and Radiation Hazard to Operator

The deposited corrosion and fission produce activity and tritium from the heat transport heavy water constitute the principal radiation sources in this system. The short-lived induced activities are present around the bleed cooler, but since it is located in inaccessible areas, this radiation source is not significant.

The valves galleries are located in inaccessible areas when the reactor is at power; the pumps are situated in rooms which have controlled access and the rooms are in dried areas, i.e., served by the D_2O Vapour Recovery system to recover D_2O and reduce tritium-in-air concentrations.

D_2O Handling

Objectives

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

- State why the HT D_2O recovery system is preferred over the ECC system for maintaining cooling of the fuel in the event of a moderate coolant leak.
- State the general location of the recovery tank in the reactor building, and the reason for this. State where the discharge lines from the recovery pump typically go.
- State what indication on the HT D_2O storage tank allows the coolant leakage rate from HT D_2O storage tank allows the coolant leakage rate from HT system to be found.
- If a feeder pipe were to break on power, state approximately how much of the leakage would be recovered as liquid and drain to the recovery system and how much would be vaporized and collected by the D_2O vapour recovery system. Assume the containment pressure does not trigger the dousing system.
- State the typical temperature range of D_2O recovered in D_2O recovery tank from a hot HT system leak.
- State what is the most important check made on the D_2O in the collection tank before it is pumped out. State the restriction on the isotopic of collection returns to the HT system.
- State the short and long term consequences of returning a single collection tankful of pure H_2O to the HT system.
- State two reasons why the HT D_2O collection rate is larger than in the moderator D_2O collection system.
- State a circumstance by which it is possible to have H_2O leak into a HT D_2O collection tank.
- Briefly describe the action of a molecular sieve in the recovery of heavy water from atmosphere containing moisture.

- State the purpose
  - The Downgraded D_2O Transfer System
  - The D_2O Cleanup System

b) The following major components in:

The downgraded D_2O Transfer System

- Downgraded D_2O storage tanks
- Oily downgraded D_2O storage tank

The D_2O Cleanup System

- Oil/water separator and pump
- D_2O cleanup feed pumps,
- Charcoal filter
- Ion exchange columns
- D_2O cleanup product tanks
- D_2O cleanup product pump
- Vent tank
- Head tank
- Strainers
- Filter

Table of Contents

Introduction

A large part of the field operators' daily work load, deals with the handling of D_2O. This section will deal with the various aspects of D_2O handling including safety precautions, radiation protection, and analysis techniques.

Heavy water is a restricted and a costly material worth approximately $450 a kilogram. Any D_2O added or removed from the system or shipped from a station must be sampled for isotopic and correctly weighed. Failure to do this results in paper losses that affect the economic operation of the plant. To aid in the accuracy of D_2O inventory, all stainless steel D_2O storage drums used on each station have a listed tare weight which is to be used when calculating the net contents of a drum. Whenever D_2O is removed, added or shipped from the station the appropriate form, ie., withdrawal, addition or shipping advisory, is filled in and forwarded to the technical unit so they can update their records as to the amount of D_2O on the station.

Correct sampling and recording of isotopics enable the, D_2O supervisor to calculate upgrading costs and determine which drums/tanks require de-oiling or shipment for upgrading.

After any drum has been sampled, the drum head statement should be updated and the drum bung sealed to prevent any downgrading or evaporation.

Isotopic, Boron, Gadolinium, Curie Content

Prior to adding D_2O to either the moderator or HT system, analysis must have been completed to determine D_2O isotopic. D_2O having isotopic less than required must not be added to the moderator or heat transport system without prior consent of the First operator. Consequences of adding low isotopic water may result in increased neutron absorption thus causing a decrease in reactivity.

If the downgrading is severe enough and enough negative reactivity is introduced, the reactor will go sub-critical and poison out. This results in enormous economic penalties in terms of D_2O upgrading and unit down time.

If the unit does continue to operate with slight downgrading, a penalty is paid in increased fuel burnup.

D_2O isotopics are routinely carried out by operations (Chemical Lab) using the infrared spectrometer.

Prior to adding D_2O, samples are submitted to the chemical unit for determination of isotopic, boron, gadolinium and curie concentration. Boron is a strong neutron absorber and is used in some Nuclear Generating plants as a source of negative reactivity to compensate for the lack of Xenon poison at startup after a long shutdown. This allows the reactor to achieve full power before Xenon equilibrium levels are achieved. As the Xenon builds up, the boron is removed from the moderator by ion exchange resins.

Gadolium is a stronger neutron absorber than Boron. It is used in the moderator systems for Shutdown System 2 and for guaranteeing the reactor shutdown (Poisoned Out). Gadolium will burn off at the same time as Xenon builds up. It is removed from the moderator by ion exchange resins.

Adding water to the moderator without first determining the boron and gadolium concentration could have a similar effect as downgrading in that negative reactivity is introduced which could result in the reactor going sub-critical and poisoning out. Although poisoning is unlikely, economic penalty is paid in increased fuel burnup.

D_2O added to either the moderator or HT system must first be analyzed for curie content to ensure that high curie moderator water is not added to the HT system. Since most leakage to atmosphere will occur from the high pressure HT system, it makes sense to maintain a low curie content in this system and reduce tritium uptakes.

Another impurity that can show up in D_2O is oil. Reactor grade D_2O should be analyzed for oil prior to addition, as the hydrogen and oxygen in the oils will disassociate via radiolysis. This will cause the deuterium levels in the D_2O and helium cover gas to rise considerably above 3%. Deuterium, like hydrogen is flammable and has an explosive range of 4% to 74% well.

- The limit for deuterium concentration in the Heat Transport and moderator systems is < 3%. At 4%, the cover gas must be purged with helium.

Hazards (Radiological)

When working with D_2O we must remember there are radiological hazards associated with the D_2O systems, these being:

a) Tritium

Tritium is an activation product and will be present in both the moderator and heat transport systems. These systems will include all open or leaking D_2O systems, fuelling machines and equipment, D_2O handling facilities and ventilation systems. Always remember the "rule of thumb" 1 Ci/kg (3.7×10^{10}Bq/kg) = 1500MPC(a) which leads to 1ALI. Tritium builds up in the moderator at a rate of 2 Ci (7.4×10^{10}Bq) per year of reactor operation.

Station staff must always wear the proper protective equipment and use the appropriate respirators.

b) Contaminants

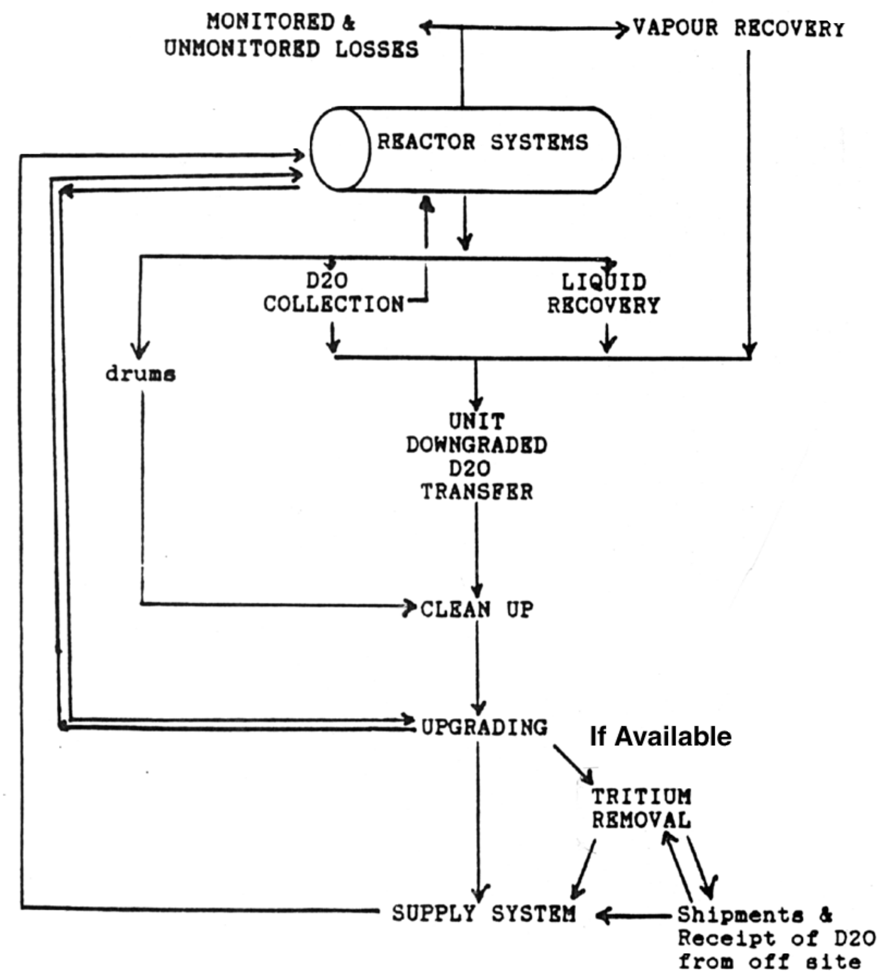
The heat transport system and moderator systems may contain activation products or fission products. Fission products may be present due to fuel defects, and activation products from the system itself or corrosion products (impurities in the liquid).

Care should be taken to wear proper protective clothing and use required instruments as specified in Radiation Protection Procedures.

General Rules:

1. Containers for heavy water must be clean and must always be tightly covered to prevent evaporation losses and downgrading.
2. Spills of heavy water must be immediately and thoroughly mopped up and saved.
3. Any liquid of unknown origin must be assumed to be heavy water and handled as such until analysis proves otherwise.
4. A record must be kept of all the heavy water movement into, within, and out of the station.

Figure 1
A typical Station D_2O Cycle



Purpose of the System

The D_2O Liquid Recovery System is designed to recover, by gravity, heavy water spilled in the Reactor Building (RB), the Reactor Auxiliary Bay (RAB), Heavy Water Management Areas where leakage is likely to occur during operation or maintenance.

System Overview

D_2O Liquid Recovery System provides a means of recovering D_2O spills or leaks of Heavy Water by a system of piping from floor drains in rooms where leakage is likely to occur to sumps or tank whence it can be returned to station inventory.

The floors in rooms are sloped to the floor drain which has a Moisture Detector (beetle) mounted in the cavity which will activate an alarm when wetted to indicate leak location. The leaked/spilled D_2O can then be drained to sumps in Heavy Water Management Areas and Downgraded D_2O Transfer Tanks.

Mechanical seal leak-off lines of D_2O pumps and pump casing drain lines are tied into the D_2O Recovery System.

All piping from floor drains and utility drains drain to sumps. All system piping in the unit drains to Downgrading D_2O Storage Areas.

System Major Components and their Function

1. Floor drains
2. Piping and plug valves
3. Collection sumps
4. Sump pumps
5. Strainers
6. Drum filling stations

1. Floor drains

These are mini sumps in rooms where there is a possibility of D_2O leaks/spills. The floor is usually sloped to the floor drain which is covered with a steel plate with drilled holes and a mesh to stop debris from getting into drain pipes. The beetle is mounted in the floor drain piping.

2. Piping and plug valves

Piping from each floor drain provides a flowpath to collection sumps or tanks. Plug valves are provided downstream of floor drains and are normally closed so that a spill in one room will not backup into lower rooms if sumps/tanks fill. Valves also prevent D_2O/DTO vapours from sumps and tanks venting to rooms.

3. Sumps

These sumps are situated on the lowest level in the building

4. Sump pumps
These are usually vertical cantilever sump pumps and are in the sumps. Discharge piping from the pumps allow the content of sumps to be pumped to Active Drainage, Downgraded Storage Tanks or to Drum Filling Station. A pump or pumps situated in D_2O collection in the unit with discharge piping to Active Drainage, Downgraded D_2O Transfer Area or Drumming Station.

5. System strainers
"Y" type wire mesh strainers are provided at the floor drain inlet piping to each collection tank. Strainers should prevent buildup of radioactive solids in the tanks. These strainers are a potential collection point for active particulate matter and debris.

6. Drum filling stations
Piping is provided to drum the contents of collection tanks.

Radiological Hazards

When investigating after a beetle has alarmed, personnel are to take precautions and ensure radiation procedures are followed. Radiation Hazard exists during maintenance on the system or its components.

Conventional Hazards

Conventional hazards occur when rotating equipment and operating valves in very congested areas. There is a possibility of high temperature D_2O leaks. Station staff must know the hazards and exercise required precautions.

Heat Transport D_2O Recovery

i Purpose of System

The purpose of the HT D_2O recovery system is to return enough HT D_2O escaping from the HT system in the event of a moderate piping rupture to provide for adequate coolant flow in the fuel channels, until the HT system can be cooled down and depressurized.

A "moderate" leak is defined here as one that does not allow the HT system pressure to fall below the saturation pressure. In other words, the rate at which coolant escapes from the leak is less than the rate at which it can be returned to the main system, at normal system pressure.

This system is then designed to avoid the need for ECC system operation during moderate HT leaks. The ECC system is intended for use in the event of a major break in the HT system, and its use involves serious consequences. ECC's operation would result in a large downgrading of moderator D_2O if moderator injection were used, or a large downgrading of HT D_2O if H_2O were used. Hence substantial upgrading costs would be incurred. Furthermore, in the case of a moderate coolant leak, crash cooling of the HT system would likely be required in order to reduce system pressure to ECC's injection pressure quickly enough to prevent fuel damage. HT components might be damaged due to the system pressure & temperature, but flow will be at a state where the leak can be eventually isolated and then repaired.

ii System Description

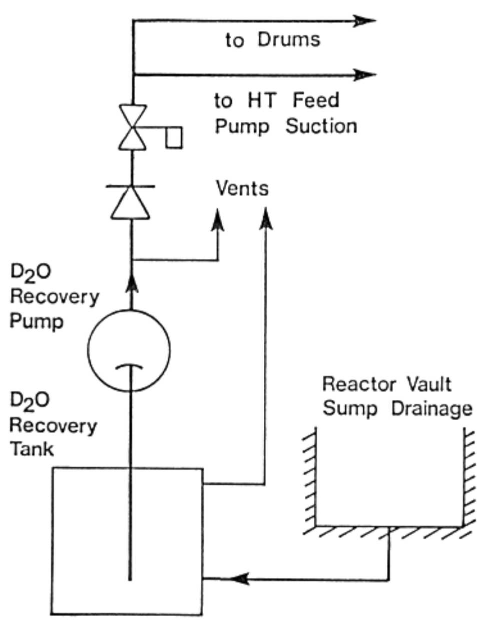
The system (see Figure 1) consists of a D_2O recovery tank located at a low point in the reactor building. Water, from a break, drains via gravity into this tank.

A HT D_2O recovery pump is provided to pump recovered D_2O back to the suction of the HT pumps (or HT D_2O storage tank) via a check valve and/or a motorized isolating valve. The recovered D_2O is then pressurized and returned to the system by the feed pumps. the motorized isolating valve opens when the recovery pump starts, and closes when the pump shuts down.

If the HT leak rate is small, then the recovered water can be pumped to D_2O Handling for chemical cleanup and /or upgrading. The D_2O storage tank inventory would then be used to supply HT D_2O until the leak could be isolated. Such action would prevent any downgrading and contamination with insolubles of HT D_2O, which would occur to some extent if recovered D_2O were returned directly to the HT system.

Figure 2

Simplified HT H_2O Recovery System



iii Operating Features of System

From HT system break at power, two phase (ie, liquid and vapour) fluid will be ejected. Initially ≈60% will be liquid and ≈40% will be vapour (steam). As the HT D_2O cools down in the main system, the liquid percentage of ejected fluid will increase. The steam buildup results in a pressure rise inside containment. Containment (reactor building or vacuum building dousing) operation will be triggered, if this pressure rise is large enough to be handled via the recovery system, containment is not likely to be triggered.

Although the liquid draining into the recovery tank is hot (≈80˚C - 100˚C) typically, the drainage system is designed to provide adequate suction head for the recovery pump.

Vent lines on the recovery tank and pump discharge allow air to be vented from these components prior to pumping back the recovered coolant. Thus the probability of airlocking is minimized. These lines, shown in Figure 1, vent into containment.

iv Detection of HT D_2O Leaks

It is useful to summarize the range of HT D_2O leaks, from small to large, and to describe how they are detected and which system(s) would be necessary to cope with the problem.

(1) Small Leaks

Small HT D_2O leaks due to bad seals on channel closure plugs, or on other equipment such as valves or pumps, will be indicated by an increased collection rate from HT vapour recovery driers. Decrease in HT D_2O inventory will also indicate a leak: HT D_2O storage tank level falling will indicate this. (Closure plug leaks can be traced directly by using the ultrasonic leak scanners on some of the fuelling machine designs). Leaks of this nature will not require recovery system operation.

(2) Moderate leaks

These will be leaks from ruptured instrument lines up to 1cm in diameter. They will likely require recovery system operating varying from intermittent pumping, to D_2O handling, to continual recirculation back to the main system. Indication of leaks of this nature may be one more of the following:

- low storage tank level
- high vault pressure alarm
- beetle alarms in drainage sumps
- reactor building liquid recovery system tank high level alarm
- increased collection rate in D_2O vapour recovery collection tank
- high inlet dewpoint to D_2O vapour recovery driers

Whether the recovery system is used to pump to D_2O handling or to the main system will depend on whether adequate HT coolant inventory is available in the storage tank (or can be transferred from other reactor units) to allow for HT cooldown, a procedure lasting about an hour. Storage is preferred, because it prevents contamination (with foreign matter) and downgrading of the HT D_2O remaining in the HT system. However, if adequate inventory is unavailable to compensate for the leakage, the recovery system would be used to return the D_2O to the feed pumps. If the feed capacity is adequate to maintain pressurizer tank level then the recovery system operation will be adequate to cope with the leak.

To confirm whether pressure control has stabilized, Station staff check HT pressure/temperature readings with saturation values in steam tables. Note that an indicated constant pressure does not necessarily mean pressure is being controlled - it may simply mean that the water is boiling, which is to be avoided.

(3) Large Leaks
For large breaks, the above indicators will all still be evident, with the additional triggering of containment and ECCs.

Heat Transport D_2O Collection

I Purpose of System

The main purpose of the HT D_2O collection system is to collect, by gravity drainage, normal leakage from HT equipment collection points via a closed piping system. The System also pumps the collected D_2O back to the HT system.

As with the moderator D_2O collection system, this system will minimize leakage to plant atmosphere. this is especially important for HT equipment as there are many more leak-off collection points than in the moderator system. Also the higher pressure of the HT system and some of its auxiliary systems will result in more leakage than in the low pressure moderator system.

II System Description

Typical collection points will be from:

(1) Pumps
The collection points will be:
- pump seals
- pump intergasket leak-off lines
- pump drain/vent lines

The pumps involved will be:
(a) main HT pumps
(b) HT pressurizing pumps
(c) HT shutdown cooling pumps

(2) Heat Exchangers
The collection points will be :
- heat exchanger drain/vent lines
- heat exchanger intergasket leak off lines

The heat exchangers involved will be:
(a) shutdown cooling HX
(b) bleed cooler
(c) bleed condenser
(d) main boilers
(e) emergency (back up) HT pump gland cooler (some CANDU designs)

(3) Valves
The valves with leak off collection points (from stem seal packing or, occasionally, double gasket cavities on flanges) will be:
(a) HT main system isolating valves (if used)
(b) feed and bleed system isolating and control valves
(c) HT shutdown cooling system isolating valves
(d) HT system pressure relief valves

A typical HT D_2O collection system is shown in Figure 1. The collection points drain into the HT D_2O collection tank which is usually situated at a low level in the reactor building to facilitate good gravity drainage.

III Operating Features

During normal operation the leakage points typically will provide enough leakage to fill the collection tank every few hours. Increases in the collection rate will usually indicate deterioration in some particular pump/HX/valve-leak off point. Flow gauges on the collection lines will enable the particular leak-off line to be traced.

As many of the collection points are collecting hot HT D_2O leakage, the collected D_2O has to be cooled in the collection tank. Cooling is provided by immersed tubes circulating water. This is a potential source of HT D_2O collection downgrading if a leak occurs in these tubes. Hot D_2O vapour from the collection tank is also cooled in a vent condenser, as shown in figure 1, and condensed D_2O drained back into the collection tank. The vent condenser is also a potential source of D_2O downgrading.

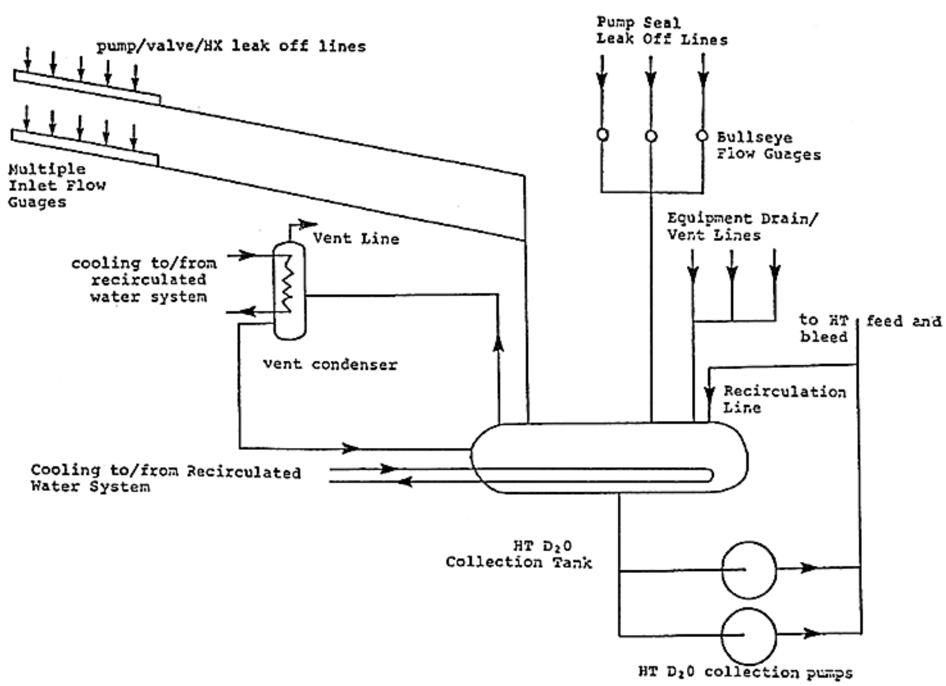
When the collection tank is almost full, a high level alarm will come in and the tank D_2O should then be recirculated using the D_2O collection pump to provide good mixing, enabling a representative sample to be obtained at the heat transport D_2O sample station. Usually the system is provided with 2 x 100% collection pumps for higher reliability due to the more rapid collection rates in this system compared to the moderator system.

The D_2O is checked for the various parameters discussed in the D_2O sampling system. Of most importance, however, is isotopic. The isotopic of the returns is usually specified to be within 0.1% of the current unit isotopic before it can be returned to the HT feed and bleed system. Authorization by the control room operator is required. The return point is usually the bleed filter inlet (ie, the purification system inlet).

The isotopic of the returns is not as important as far as reactivity is concerned as that of the moderator returns. For the HT D_2O the reactivity will change by about 0.05mk for each 0.1% change in the bulk HT isotopic. As a specified example, pumping back to the main system one collection tank full of H_2O would result in a downgrading of about 1.3%

If a HT collection sample is not satisfactory it is pumped to drums ready for chemical clean-up and/or upgrading.

Figure 3
Typical HT H_2O Collection System



Exercise

1. Calculate the resulting isotopic HT D_2O if the contents of the HT D_2O collection tank are completely pumped back into the HT in the following cases (A calculator is necessary to do this accurately.)
   (a) HT collection isotopic is 99.00%
   (b) HT collection isotopic is 98.90%
   (c) HT collection isotopic is 97.00%
   (d) HT collection tank is full of H_2O
   State the immediate and long term consequence of (a), (b), (c) and (d).
   Assume:
   (a) initial HT system isotopic is 99.00%
   (b) total HT system mass is 160Mg
   (c) mass of water in a full collection tank is 200kg
2. How often would you make an isotopic check of the collection tank D_2O?
3. Which equipment has the largest normal leak rate?
4. In which ways may H_2O get into the collection system?

Heavy Water Leakage -Recovery by Molecular Sieves

Heavy water leaked from CANDU systems is recovered by two basic methods:
1. Liquid Recovery (Collection) systems.
2. Vapour recovery systems to remove heavy water from the air leaving the plant and areas such as reactor vault atmospheres.

Heavy water vapour is recovered by Molecular Sieves (Driers).

All air leaving the plant and air from areas which are likely to contain heavy water vapour is passed through filters, commonly called molecular sieves.

The packaging material of these filters is an inorganic polymer in granular form. The infrastructure of the polymer has cavities or holes such that linear molecules are able to line themselves up and pass through but non-linear molecules are trapped. The significance of this is that Nitrogen and Oxygen (i.e. air) may pass through the sieve, but water or nitrogen oxides from radiolysis of air are trapped. Thus, all water vapour is removed from the air, both heavy water and light water, as well as the small amount of nitrogen oxides present. At some point in time the sieve becomes saturated and will no longer function. For this reason, sieves are run in pairs, one in service, the other being regenerated (having the water removed) or on Standby.

Regeneration is done with a stream of preheated air which expands the cavities allowing the water to escape. The resulting warm moist air is passed through a condenser for D_2O recovery. The condensate will be downgraded from the normal atmospheric humidity which was also trapped on the sieve.

The service/regeneration cycles may be depicted schematically.

Service

Reactor Vault Air → Molecular Sieve → Dry Air

Regeneration

Air → Heater → Molecular Sieve → Condenser → Waste Air

Condenser ↓ D_2O Recovery

2.0 Purpose of the System

The purpose of the system is to recover D_2O vapours from the reactor vault and fuelling machine areas in order to:

1. minimize D_2O losses for economic reasons
2. maintain tritium and airborne contamination at a low level to:

- minimize internal dose to personnel
- limit gaseous emissions to the environment

The safety related function of the systems is to maintain containment integrity by:

1. maintaining containment at negative pressure with respect to atmosphere during normal operation.
2. closing the containment isolation valves related to this system on receipt of a button up signal.

Vapour recovery systems are Group A Safety Related Systems.

3.0 System Overview

There is one Reactor Vault and Fuelling Area Vapour Recovery System (where applicable) for each reactor unit. The system contains vapour recovery dryers which draw moist air from the vault and fuelling machine duct and pass it through a molecular sieve desiccant bed. The air is dried in passing through the desiccant and following this the air is returned to containment. The system also dries the air from the shutdown cooling heat exchanger rooms.

Containment Dewpoint

The parameter to be controlled for vapour recovery systems is DEWPOINT. Control of dewpoint is achieved directly or indirectly depending on the operating mode. Maintaining dewpoint at a low level maximizes D_2O recovery and minimizes losses.

Containment Activity

The vault atmosphere will normally contain tritium, due to chronic leakage, and airborne contamination. By maintaining containment dewpoint at a low level, the concentration of tritium is also controlled. As air is circulated through the system, a large filter assembly removes airborne particulate contamination and iodine if present in low concentrations.

Containment Pressure
Containment is maintained at negative differential pressure with respect to atmosphere, ie., the powerhouse. This is done so that clean air will leak into containment rather than airborne contamination. Leaking out into the powerhouse.

4.0 System Equipment and Operation

The system contains up to four single Bed Area Driers and one double bed Exhaust Drier. One or two of the Area Driers normally operate to circulate air and remove moisture from the containment atmosphere. Located on the supply line to the Area Driers is a large Filter Assembly that removes airborne contamination from the air supply to the driers.

System Orientation
The Fan on the Exhaust Drier assembly maintains containment at negative pressure. The fan takes its suction from the air return line to containment. The flow to the Exhaust Drier is adjusted by a control valve. The control valve is throttled to achieve the flow needed to maintain containment within the required negative pressure range Dried purge flow from the Exhaust Drier is directed to the containment exhaust stack of the Powerhouse Ventilation system. The air flow through this stack is monitored continuously for environmental emissions.

Air is circulated through the system by a fan on the operating area drier unit. Moist air from containment is drawn through the desiccant (molecular sieve) tower where moisture is removed by absorption. The absorption process generates heat. Therefore, the hot, dried air is passed through a condenser for cooling prior to reaching the circulating fan. The fan exhausts the air through a three-way valve which directs the flow back to containment.

As the desiccant continues to absorb moisture, it eventually becomes saturated and absorption stops. At this point, the desiccant must be regenerated.

During regeneration, the position of the three-way valve changes to cause air to recirculate past the Heater, which becomes energized. The heated air passes on to the drier and drives the absorbed D_2O out of the desiccant. The moistened air stream then travels to the condenser where the vapour condenses. The condensed liquid is directed to the downgraded D_2O Transfer System.

At the end of regeneration, the heater trips but the fan continues to circulate air through the desiccant to cool it. Following cooldown, the fan trips and the drier goes on standby.

4.1 Dryer Cycle

The foregoing has described a drier cycle. The cycle commences when the drier enters the absorption phase. This is followed by regeneration, then cooldown. At the end of cooldown, the drier enters standby. Normally, one or two Area Driers operate in absorption while the others are in regeneration, cooldown, or standby.

For the Exhaust Drier, normally one of the two desiccant beds operates in absorption using a Fan, while the other is in regeneration using a Fan for cooldown, or standby.

4.2 Dryer Modes

The driers may be operated in any of three modes; auto, timed, or manual. Auto is the preferred mode of operation requiring a minimum of operator attention. The operating mode selections are carried out locally at the drier panels.

4.3 Containment Button Up

The Containment Button Up System monitors vault and fuelling area pressure and activity. If pressure and/or activity increase to the button up set point, as it will in the event of a Loss of Coolant Accident, the Containment Button Up System will cause the Containment Isolation Valves associated with the Reactor Vault and Fuelling Duct Vapour Recovery System to close. This action isolates the vault from the vapour recovery system to minimize the spread of activity from the vault. A containment Isolation Valve also isolates vapour recovery from the Downgraded D_2O Transfer System.

4.4 Other Connected Systems

The following support systems are required for the operation of the Reactor Vault Vapour Recovery System:

- Electrical
  - 600vac cl IV
  - 600vac cl III
  - 120vac cl II
  - 48vdc cl II
- Downgraded D_2O Transfer System
- Service Water System
- Powerhouse Ventilating System
- Containment Button Up System
- Common Processes and Unit Computer

4.5 Equipment Location

The D_2O vapour recovery supply line exists usually in the corner of the vault and the return line re-enters the vault. The Filter Assembly is located near the drier room. The Area Driers and Exhaust Driers are located in a drier room (area).

5.0 Safety Considerations

5.1 Operating Constraints

Operating Policies and Principles are usually established with respect to the containment system to:

a. Ensure containment is available when required.
b. Provide guidelines for actions during abnormal situations.
c. Control modification, maintenance, and leakage on the system.

The reactor vault and fuelling duct D_2O vapour recovery system is an extension of containment and therefore the containment OP & Ps apply to it as well.

5.2 Radiation Hazards

a.) Tritium

Tritium may be a hazard, especially when equipment is opened up for maintenance. Personnel should use appropriate radiation protection procedures and equipment when working where tritium may exist. Prior to opening up the system for maintenance, the drier should be regenerated to ensure there is a minimum amount of water vapour in the system.

In the event of D_2O spill in the Area/Exhaust Drier rooms personnel should leave the area and initiate cleanup within three minutes.

b.) Airborne Particulate

The desiccant bed will become a radiation hazard as particulate accumulates. Proper shielding and maintenance procedures are required when adding or removing desiccant.

The particulate, HEPA, and activated carbon in the filter assembly will require replacement periodically. Shielding, personnel protective clothing and maintenance procedures will be required in this situation.

5.3 Conventional Hazards

a.) Mechanical

The system contains rotating equipment and pneumatic valves.

b.) Electrical

The system is supplied by 600V power to fans and heaters and 120V power to panels and instrumentation. The normal precautions when dealing with electrical equipment are followed.

c.) Thermal

The heaters and piping downstream of the heaters will be hot during the regeneration and cooldown phase. Station staff must not touch this equipment if the insulation has been removed or tampered with.

d.) Chemical

Desiccant used in the driers can cause irritation or burns to skin, respiratory tract, and eyes if contact occurs. Suitable protective clothing and breathing protection should be worn when handling desiccant.

Sampling Techniques

Introduction

Various techniques are used for sampling D_2O. It may be from a drum, off the floor or at a sample station (SS). Regardless of the source of the sample, the equipment used to obtain the sample should be flushed sufficiently and the container used to transport sample to the lab should also be flushed properly. These two objects can be the same but often the sample is obtained using a syringe or aspiration bulb or straw and then put into a sample bottle.

Hazards and Precautions

Since some of the heavy water containers being sampled will contain high curie water, proper protective equipment must be used to minimize radiological hazards. Because of the potential radiological hazards, housekeeping is of particular importance at these sample locations. Any spills of heavy water must be cleaned up as soon as possible. When sampling high isotopic (reactor grade) D_2O for verification of station receipt or shipment; station staff ensure the sample bottle is filled completely. This avoids downgrading the sample with atmospheric air or H_2O that may be trapped in the sample bottle.

Bulk Sample from Sample Station (Glove Box)

Whenever a sample is required from systems such as Moderator or Heat Transport a different sampling systems technique is used. For every system the procedures may vary; therefore station staff should determine the correct procedure and apply it.

The reason for this variation in procedures is due to differing system pressures, contents etc. The procedures themselves vary as to flow rates recirculation time, amount of sample required.

Heavy Water Spill or Leaks

To sample water from a spill or leak the depth and extent of the spill will determine the method used for sampling.

The procedures are covered in the operating procedures and are usually discussed in the section on liquid recovery.

When taking or collecting samples station staff must wear proper protective equipment for Radiological and Conventional hazards and follow proper procedures.

General Precautions

The following is a brief summary of the general operating procedures; the proper manuals should be read for a more detailed explanation.

- Approved out of system D_2O containers (drums, TDO packages, tankers) used for storage of D_2O must display a D_2O Container Label with copy routed as indicated on the label listing contents and analysis.
- All D_2O containers used as temporary storage for D_2O (drums, pails, carbons) shall be analyzed and labeled to identify contents as completely as possible.
- As a minimum any analysis of D_2O shall consist of weight of fluid, isotopic, and Ci of Tritium.
- Station D_2O inventories are usually performed monthly whenever possible, and must be performed quarterly as a minimum.
- Station D_2O recovery and loss reporting shall be performed on a shift basis for daily reports and averaging for calculations of D.E.L.'s (Delivered Emmision Levels)
- All water with a D_2O isotopic of less than 0.3% ***may*** be disposed to active liquid drainage and recorded as a D_2O loss.
- No heavy water should be stored within 4 meters of 'New Fuel' enclosures.
- Tritium is a whole body irradiator. Protective clothing must be worn when handling tritiated D_2O to prevent ingestion, skin-wetting or inhalation. Refer to Radiation protection Procedures (Module 3) for appropriate clothing.

Caution

Unless the actual tritium concentration is known, any D_2O must be assumed to have a high tritium concentration until proven otherwise.

Miscellaneous Operations:

Decanting

Decanting is a procedure used to remove D_2O from drums of oil. The only way that the decanting process can be done successfully is if the drum has been sitting motionless for at least 24 hours. This will allow time for any water mixed with the oil to separate out. During decanting, the drum must not be moved around as this will cause the oil to mix with the water and defeat the purpose of the exercise.

Always Remember:

a) Downgraded D_2O handling involves the same procedures and precautions as reactor grade D_2O handling.
b) Any liquid of unknown origin must be assumed to be heavy water and handled as such, until analysis proves otherwise.
c) Extreme care must be exercised when handling moderator D_2O. We have concentrations in excess of 20 curies per kilogram. Working unprotected at a moderator D_2O spill is like working in a 100 rem per hour field. Rule of thumb: 1 curie of D_2O = 1500 MPCa of tritium.
d) D_2O often contains activated corrosion products. Always take external gamma surveys when transferring D_2O.

Deuterations and De-deuterations

Introduction

The following will only be a general explanation of the meaning of deuteration and de-deuteration. Commonly referred to as DEUT/DE-DEUT.

We use resins in IX columns for chemical control. in order to remove the used or spent resin from columns we must use water as a carrier or flushing system.

The systems use heavy (D_2O) water therefore heavy water must replace the light water in IX columns this is deuteration.

Deuteration

Deuterations are performed on ION Exchange Columns used in our heavy water system. The basic idea is to replace H_2O with D_2O. Prepare a column or hopper by adding the required resin and then puddle. Connect the hopper to the deuteration system ensuring the proper connection. The idea basically is to put D_2O(heavier) in at the bottom and push H_2O out of the top slowly. D_2O connections should therefore be made at the bottom and H_2O connections at the top. When Deuteration is complete the container is ready for use.

Chemical analysis will confirm that deuteration has taken place.

De-Deuteration

A de-deuteration is the opposite of a deuteration light water is injected in the top of the container and heavy water extracted from the bottom. The purpose of de-deuteration is to retrieve the D_2O from the resin before the resin is disposed of. As in the deuteration process the H_2O flow is important as well as having collection space available.

General Concerns

- Proper radiation procedures should be followed.
- A **slow** constant flow should be established and maintained until collected water is of required isotopic.
- Ensure sufficient collection is available when performing procedures.

Heavy Water Recovery

Introduction

Immediate action is required when a heavy water leak is detected, to confirm the escape. recovery operations must then follow immediately to maintain the workers dose ALARA. Depending on the size of the spill or leak get help if necessary.

Heavy water leaks or escapes are generally divided into 2 categories, they depend on magnitude, and are usually divided as to wet surface and major escapes.

For wet surface the use of sponges, mops, etc, can be utilized to pick up the liquid from the surface. The liquid is then transferred to carboys covered pails etc.

For major leaks or escapes use a Vac-u-max system (wet vacuum), which transfers the liquid directly to drums.

Recovery Equipment

D_2O recovery equipment cabinets are usually placed or located at strategic locations, usually at potential leak points, also where they are readily accessible to the workers.

Portable cabinets or mobile cabinets are usually on site and available at the unit.

Leak Search

Where required inaccessible and some accessible areas are sampled for tritium atmosphere. Tritium implies D_2O and the drums or tanks are designed and installed to collect and remove D_2O. These areas will have a particular concentration of MPC(a)s of tritium while operating, and this is sampled routinely. When MPC(a)s start to increase, for no known reason;. monitors may not have alarmed, detectors for leaks have not alarmed but the increase in isotopic of drier collection increases, then something is leaking D_2O to the atmospheric and it must be found.

Two methods “Hot Leak Search” and “Cold Leak Search” are used to discover the origin of the leak.

Hot Leak Search

This is carried out with the HT system hot and pressurized. It has the advantage of allowing a leak to issue steam, whose vapour cloud is visible on entering the area. It also will allow a very small leak, such as a crack or pinhole, to remain evident. Cooling the system could cause a leak to disappear completely. Of course, steam leaks are very dangerous and eyes (and ears) are the best detectors, rather than hands. When a vapour cloud is spotted, a piece of cloth attached to a stick or broom handle can be used to find the actual source. Hot leak searches are carried out on shutdown and before startup. This type of search will reveal a leak that, when cold, could be absorbed into lagging that may take days to soak through.

Proper radiation and protective equipment is always worn when a leak search is done, and a flashlight is a handy item. A leak may be in a very inaccessible spot and this requires a degree of caution. Most often, leaks are very hard to find (hence the name “SEARCH”) and it is handy to know before hand which equipment was previously leaking and /or has been worked on. A "Route" should be planned, and leak searches carried out with a "buddy".

Cold Leak Search

If a leak occurs on a normally "COOL" system (below saturation temperature), a cold search is done. These water leaks are evident as drips from equipment or "WEEPERS". Water will form droplets at valve stems, bonnets, joints, flanges, unions, kinks in tube connections, pump shaft seals and many other places.

Time and effort, along with a good knowledge of the system, and experience are helpful. Of course, sometimes a hot system will not reveal a leak that would be visible if the system were cooled, but kept pressurized. For this situation a cold, pressurized leak search would be appropriate. Sometimes leaks will have stopped completely, but deposits such as rust or lithium will be left behind and this will indicate a leak.

Piping, instruments, gauges, etc, are often located in rooms where they are not readily accessible. Minor leaks occur that can be very hard to detect. In these cases, portable tritium monitors may be used. Because the rooms are continuously ventilated, the concentration will be much higher in the area nearest the leak, and the monitor will lead you right to it, if adequate response time is given.

Recovery Phase

When a leak has been discovered or it becomes a small undetected leak the systems will collect the D_2O. The systems used are vapour recovery and liquid recovery, both these systems will be dealt with in this lesson.

D_2O Recovery:

Introduction

Some D_2O will escape chronically from the Moderator or heat transport systems as a normal part of routine operations. "Higher" or "Acute" levels will occur if there are equipment or operational problems.

Typical sources are:

Chronic	**Acute**
- pump seals	- failed welds
- valve packing	- failed ice plugs
- Fueling operations	- valving errors
- system sampling	- HX tube failure/leaks
- equipment draining for maintenance	- pump seal failure
	- valve packing failure
- minor leaks from Hx tubes fittings	- Gasket leaks
- swageloks etc.	- Failed swagelock or other mechanical fittings

The heavy water loss rates and upkeep for a unit must be known hour by hour or day by day therefore all leaks must be recovered.

If for example you discover a leaking valve, you should note the valve identification for reporting and repair. We must also determine the leak rate in drops/min, is it streaming?, spraying?, to accomplish this we may require the use of carboys(buckets) or some other containers to determine the leak rate. We must also note where the water is leaking to, ie floor drain, sump, onto a motor, into lagging etc.

All this information may be required in order to, if a particular course of action is needed, to contain and clean up the leak. The actions are usually listed in the Station Operating Manuals.

Leaks are usually cleaned up as soon as possible. Often a simple procedure such as tightening a valve gland nut will do, but more often a comprehensive plan is required. This may involve isolation of equipment, depressurization, ice plugs, special monitoring or some form of prefabricated collection system. Spilled water can be wiped up by use of sponges, mops, pails, etc, and also by the use of vac-u-max.

Recovery

The basic “Recovery” is by the following means:

Recovery from Surfaces

Heavy water escapes can be divided into two categories depending on magnitude. These are: Wet surfaces and major escapes.

For wet surfaces, sponges, self-wringing mops, carboys with funnels and plastic pails with air tight lids are to be used. The heavy water transferred from the surfaces by the sponges or sponge mops to carboys and pails is retained for D_2O inventory as part of D_2O losses.

Vac-U-Max Equipment (Water Vacuum)

A Vac-U-Max air venturi tank kit is provided for use in recovering appreciable amounts of escaped D_2O. By means of a vacuum cleaner head and hose, this equipment will transfer heavy water from flooded floors, directly into standard heavy water drums. Without the vacuum cleaner head the hose may be used to transfer from drum to drum, which will occur at the rate of 5 kg/s.

Remember

Although immediate action may be required the ALARA principle will apply and all actions must follow, with the minimum necessary total dose uptake, and always request help if necessary.

To meet the requirements of the liquid recovery two systems or methods have been built into the system. One is the liquid recovery system, the other is the Vapour Recovery system, we have discussed these systems in detail during your other lessons.

Liquid Recovery

Introduction:

When designing a station certain principles are applied for liquid recovery. Beetles are used where possible in pipes or cavities as a means of early leaks detection. Dikes are installed in certain key areas to contain likely D_2O leaks. Insulation is applied to pipes to prevent condensation build up, and the use of directing the escape by the shortest routes through sloped floors to drains or recovery sumps.

System Operation

We use two basic modes of operation, unit D_2O collection from moderator and heat transport systems can be returned via collection back to the systems. If the system collection is not chemically acceptable to return, its pumped through pipework to the Heavy Water Management Area.

D_2O which finds its way to common floor drains goes directly to the D_2O downgraded transfer tanks for that area.

Procedure

The procedures for sampling, pumping or transferring to or from collections are covered in the station manuals and should always be followed.

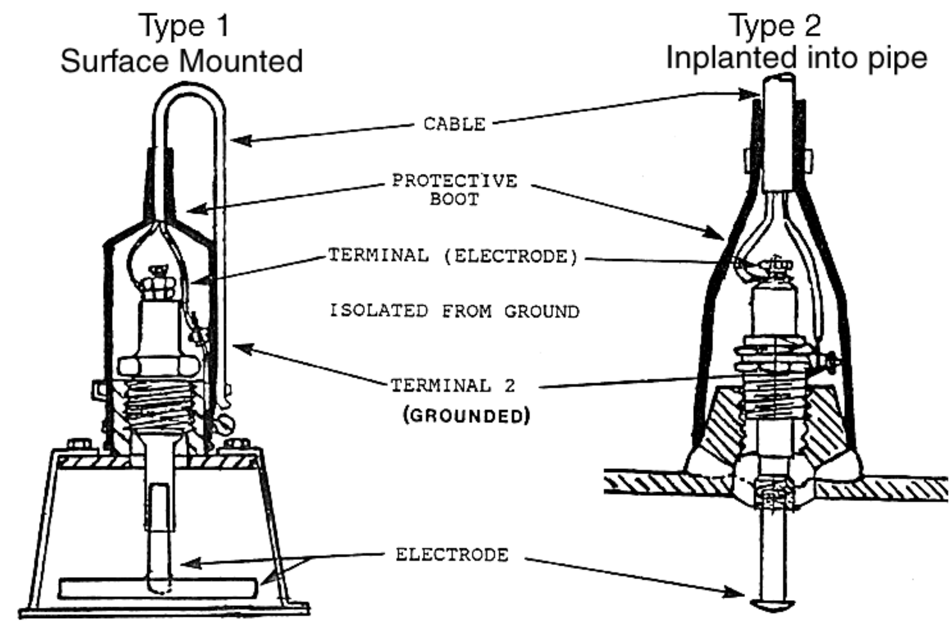
The normal mode of operation for the liquid recovery system is to have water recovered via floor drains to be collected in a tank. All of this water will eventually go through the Station Upgrader.

Water collected from the floor drains would be segregated according to the tritium concentration. This is done by collecting low tritium water, from Heat Transport and related systems, and collecting high tritium water, from Moderator and related systems.

Water would be transferred to drums rather than tanks if it is required to store the water independently of the downgraded D_2O transfer system. This may be required if the water is particularly dirty or if the transfer header is out of service.

Beetles

Figure 4
Moisture Sensor (Beetle)



Beetles are provided with 24 VDC from the moisture indicating alarm units. This voltage is considered safe, avoid bridging the gap in the beetle as the reflex action caused by the low level (24V) may lead to injury from shocks.

Routines
Accessible (on power) beetles will be periodically tested on a call up routine. Floor and tray mounted beetles will be tested usually semi annually, in line beetles will be tested usually quarterly.

In accessible areas beetles will be tested during shutdowns.

Liquid Recovery Hazards:
Liquid can be present in many areas of the plant and in various states therefore all:
a) Radiological precautions and procedures will apply.
b) Electrical precaution and procedures will apply.
c) Conventional precautions and procedures will apply.

Vapour Recovery

Purpose:
1 We recover D_2O vapours from the reactor vault and fuelling machine areas in order to:
   a) Minimize D_2O losses for economic reasons.

b) Maintain tritium and airborne contamination at a low level to:
   1) minimize internal dose to personnel
   2) limit gaseous emissions to the environment.

2. The system will also:
   a) Maintain containment at negative pressure with respect to atmosphere during normal operation.
   b) Close the containment isolating valves on the system upon receipt of a button up signal.

This is a safety related function to maintain system containment integrity.

System:
To remove the moisture from containment atmosphere in the Reactor Vault and Fuelling Areas we use at least two systems normally.

Field Routines:
Inspect the dryer control panel, for temperatures, pressures, dewpoints, and indicating lights to ensure normal operation as per the operating manual.

Hazards:
All radiological, conventional and electrical hazards could be present due to equipment operation and placement therefore all precautions and procedures must be followed.

Non-Routine Operation
(procedure included for reference only)
Shutdown State:

- dryers are off
- fans are not running
- heaters are off
- three way valves are in the regeneration position

Start-Up
(procedure included for reference only)

1.

- open containment isolation valves
- establish/check Area Dryer valve positions
- establish/check valve positions
- on the local common and Area Dryer panels set up dryer handswitch positions for auto, timed or manual operation:
- check that the selected dryer enters absorption
- check the directional valve repositioned for absorption
- check local panels for annunciation conditions and investigate if required

Rationale:
the above actions and checks are carried out to place the Area Dryers in normal operation

2. Start Up - Exhaust Dryer
- confirm contaminated exhaust fan is running
- confirm at least one Area Dryer is running

Rationale:
- at least one Area Dryer must be operating before the exhaust Dryer will start.
- on the local Exhaust Dryer panel set up dryer handswitch positions for auto, timer or manual operation:
- check that local alarms are clear
- check PV is open
- in field or MCR check flow to contaminated exhaust
- to place the system in normal operation for maintenance of containment pressure.

Preferred Operating Mode - Automatic

1. The preferred mode of operating is auto.
Rationale:
This mode will minimize the time spent operating the equipment. Also, auto mode controls dryer cycles based on dryer outlet dewpoint. This is the most direct parameter to control to maintain D_2O losses and tritium concentration at a minimum.

Exhaust Dryer
In automatic control mode, the Exhaust dryer operates continuously alternating between two desiccant beds. In manual operation the operator must perform this switching function.

The same considerations regarding operating mode for the Vault Dryers apply to the Exhaust Dryers.

Vault Pressure
The vault pressure control loop operates a control valve to throttle flow to maintain containment pressure in the range -2.5 to -3.5kPa.

Shutting Down:

The dryers can be shut down for maintenance on a containment isolation valve or maintenance on the dryer.

Note:
Shutting down the Reactor Vault and Fuelling Area Vapour Recovery System will eventually lead to loss of negative pressure in containment (unless and APRV is operated).

If time permits, dryer beds should be regenerated prior to shutdown.

Rationale:
This will maximize dryer availability during start up.

D_2O Cleanup/ Downgraded D_2O Transfer Objectives

1) The downgraded D_2O Transfer System

The downgraded D_2O transfer system is a station wide network of pumps, pipe lines and tanks used in the collection, transfer and storage of downgraded heavy water from Moderator and Heat Transport Systems. the recovered D_2O is transferred from the unit to the collection system located in a typical heavy water management area. It is the collection system in the heavy water management area that will be addressed.

The purpose of the Downgraded D_2O transfer system is to transfer (and store) downgraded D_2O from various systems in the reactor building, fuelling facilities Auxiliary area, and the Central Service area to the D_2O management area. the system consists of a station-wide network of pumps, pipelines and tanks which serve to transfer the downgraded heavy water to a set of storage tanks in the heavy water management area, prior to being fed into the D_2O cleanup system and or the D_2O station upgrader system.

2) The D_2O cleanup system

The purpose of the D_2O cleanup system is to purify the heavy water received from the downgraded D_2O transfer tanks in the heavy water management area (H.W.M.A.). This water is of varying isotopic concentration and may contain oil, particulate, dissolved ionic or organic impurities. Recovery heavy water is handled in batches, based on presence or absence of oil and heavy water isotopic concentration. After cleanup, the D_2O is passed to the upgrader where the isotopic concentration is increased to reactor grade.

The purpose of the system is to remove oil, particulate, and dissolved ionic and organic impurities from the D_2O recovered and transferred to the downgraded D_2O transfer tanks. The purified D_2O may then be passed to the upgrader or tankers or drums. The D_2O cleanup system can also receive D_2O from a unit moderator purification system and if below specification it is purified prior to upgrading.

b) the following major components:

The downgraded D_2O transfer system

1) Downgraded D_2O storage tanks

All the recovered downgraded heavy water from the Unit is pumped to a storage tank. These tank or tanks are each of nominal capacity are located in the Heavy water management area (H.W.M.A.).

2) Oily downgraded D_2O storage tank

All the downgraded D_2O from the floor drains of the unit reactor building which are collected separately from other recoveries in the unit are transferred to a separate tank. This tank is of a nominal capacity and is located in the HWMA.

The D_2O cleanup system

1) Oil/water separator and pump (O/S)

This unit coalesces and removes down to approximately 10mg/kg of oil in D_2O (10ppm) in an emulsified state. The unit also filters out solids greater than 5 micron. D_2O processed through O/S is routed to the storage tanks.

2) D_2O cleanup feed pumps

Feed pumps transfer downgraded D_2O from either storage tanks to a Charcoal Filter, FR1.

3) Charcoal filter (FR1)

The downgraded D_2O collected in storage tank /tanks is routed through FR1 to remove organic impurities and any remaining traces of oil and particulate contaminants.

4) Ion exchange columns (IX)

The downgraded D_2O processed by FR1 is then passed through two ion exchange columns (IXI and or IX2). These columns remove the cation and anion impurities in the D_2O.

5) D_2O cleanup product tanks (TK)

Downgraded D2O that has been processed through IX1 and/or IX2 is routed to either of the D2O cleanup product tank or tanks

6) D_2O cleanup product pump

The product is used for recirculating product tank content for sampling or returning contents to storage tanks if contents are off specification.

7) Vent tank

The Downgrading D_2O transfer and the D_2O cleanup system equipment is under pressure equalized to the Vent tank where in turn is pressure equalized to the vent chiller. The vent tank has been detected early by a beetle installed such that if the vent lines from downgraded D_2O transfer or D_2O cleanup becomes flooded the problem is detected early by a beetle which is installed in the tank Appropriate action can than be taken to find the source of liquid in the vent tank.

8) Head tank
The head tank is filled with demineralized water and used during the de-deuteration process of spent resins and the charcoal filter in the D_2O cleanup system.

9) Strainers (STR)
Oily heavy water from drums is passed through either cleanup strainer before it is routed to FR. These strainers remove bulk oil and participate contaminants.

10) Filter (FR)
Oily heavy water from drums passed through STR is then routed through FR. FR is a cartridge type filter that will remove bulk sources of oil. After passed through FR the D_2O is routed to storage tanks.

Note:
Not all CANDU 600 system may be exactly as described, the components listed are the basis of a cleanup/downgraded system.

Fuel

Training Objectives

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

- Outline the historic design evolution of CANDU fuel from NRX to CANDU 600.
- Tabulate the basic components of a fuel bundle and understand the basic design philosophy.
- Understand the limits and relationship of bundle power and fuel burn-up.
- Tabulate the fuel compatibility requirements with respect to the heat transport system.
- Tabulate the fuel compatibility requirements with respect to the fuel handling system.
- Tabulate the fundamental criteria fuel imposes on reactivity control requirements.
- Tabulate the relative levels and requirements of an irradiated fuel storage bay to cope with fuel cooling, shielding and containment.
- Understand the basic steps of fuel fabrication from U_3O_8 concentrate to final shipment to site.
- Understand the power history of typical CANDU fuel bundles, and relate the post fuel defect history by type of failure with relationship to burn-up, power level and power ramping.
- Relate fuel cost in mills/Kwhr.
- Discuss the merits of irradiated fuel disposal versus reprocessing for enriched fuel cycles.

Table of Contents

1. Introduction

The CANDU heavy-water reactor (PHWR) fuel bundle uses two main materials: Zircalloy and UO_2. When loaded with UO_2 pellets the bundle weighs about 25 kg of which more than 90% is uranium dioxide fuel. The bundle design has resulted from 35 years of development work involving 500 man-years and costs estimated to be in excess of $600 million (1993 Canadian dollars).

Prominent features of the CANDU bundle are:

- high density natural UO_2 pellets, which ensure dimensional stability; this assists with remote, on-power handling of fuel;
- thin-walled collapsible Zircalloy-4 cladding for neutron economy and improved heat transfer. Neutron economy ultimately means reduced electricity costs, and improved heat transfer means better fission gas retention;
- no gas plenum; the fuel performance codes ELESIM and ELESTRES predict that no plenum is necessary, thus maximizing the fissile content per bundle. Extended burnup experiments and commercial experience have confirmed this;
- high-integrity resistance welding of end-caps;
- Canlub graphite interlayer between the UO_2 pellets and Zircalloy cladding, which improves the tolerance of the fuel to power ramping;
- induction-brazed spacer pads, which maintain separation of the fuel elements without the need of complex, expensive spacer grids; and
- simple bundle structure. This is possible because the pressure tube supports the fuel bundle, and all reactivity control mechanisms are external to the fuel channel.

The use of heavy water as a moderator provides the opportunity to use natural UO_2 for heavy-water CANDU reactors. Thus this fuel cycle does not require expensive enrichment facilities.

This lesson reviews reactor systems with which the fuel has to interface, and from which the design and performance requirements are derived. These requirements are written into technical specifications which define the standards. The extensive research information then has to be assembled in accordance with these requirements, to produce the reference fuel design. The success of the design is demonstrated by extensive computer code assessments, type tests, and the very successful operation of the commercial heavy-water power reactors.

2. Evolution of CANDU Fuel Design

2.1 Design Description

Figure 1 shows typical heavy-water reactor fuel bundles in use today. It is reliable and relatively inexpensive, and has low neutron absorption. Each bundle is fabricated from seven basic components. The fuel elements contain UO_2 pellets (natural uranium) in a Zircalloy-4 sheath. A graphite layer (Canlub) on the inside surface of the sheath reduces the effects of pellet-clad interaction (PCI). End caps are resistance welded to the ends of the sheaths to seal the element. End plates are also resistance welded to the end caps to hold the elements in a bundle assembly. Spacer pads are brazed to the elements at their mid-points, to provide the desired inter-element spacings. The bundles are placed in rows of 12 (CANDU-6 and Pickering reactors) or 13 (Bruce and Darlington reactors) in horizontal pressure tubes. Each bundle is spaced from the pressure tube by bearing pads brazed near the ends and at the mid-point of each outer element. Beryllium metal is alloyed with the Zircalloy to make the braze joints.

Each 37-element bundle, which is the standard design for the Bruce, Darlington and CANDU-6 reactors, is 495 mm long with a nominal weight of 23.7 kg. Each bundle contains a nominal mass of about 19 kg of natural uranium; it is easily lifted by hand.

2.2 Evolution of Fuel Bundle Design

In Canada, the development of power-reactor fuels began over 40 years ago and is based on the experience of test reactors NRX and NRU at Chalk River. Initially the test reactors utilized natural uranium in metal form. The metal fuel rods however proved to be dimensionally unstable, with large elongation rates under irradiation making it unsuitable for power reactors. In the 1950's research continued with the development of UO2 oxide which proved to be very stable dimensionally under irradiation and has been the standard fuel for all CANDU reactors.

Figure 1
CANDU Fuel Bundles-82.5mm Dia x 495mm Long

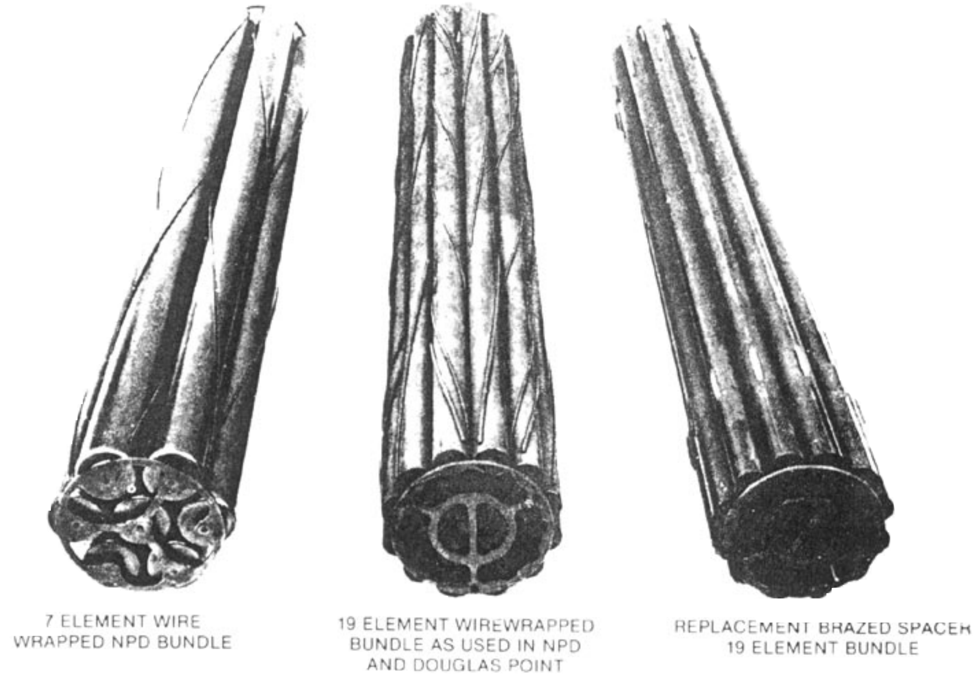
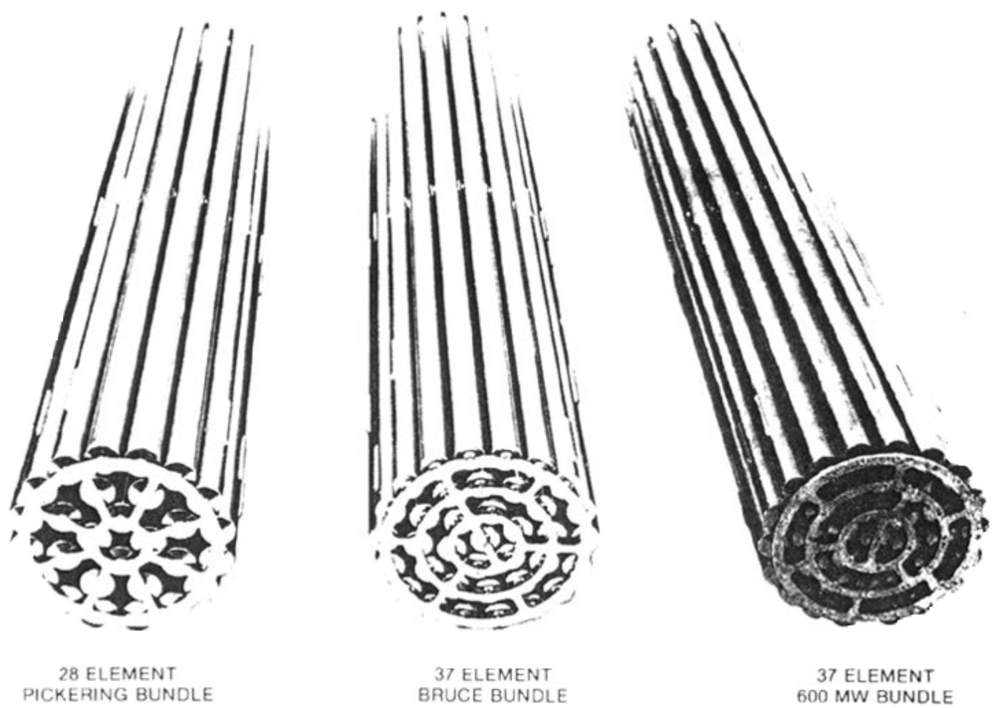


Figure 2
CANDU Fuel Bundles-102mm Dia x 495mm Long



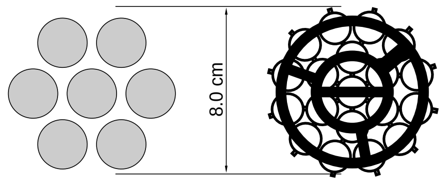
The goal of high neutron economy, set in the 1960s, is met with a fuel bundle consisting of the fuel material itself, and a minimum of containment material. The original fuel charge for the first CANDU power reactor NPD (Nuclear Power Demonstration, a 22 MW reactor) consisted of 7-element bundles in the outer zone of the reactor core and 19-element bundles in the inner zone. These original bundles were made with "wire-wrapped" elements. The replacement design of fuel for the NPD reactor and the later KANUPP, Douglas Point and RAPP reactors was based on a "brazed spacer" design of element; these original and replacement bundles are shown in Figure 2. Performance data from these reactors were used in the evolution of CANDU fuel, to develop the Pickering 28-element bundle, and the 37-element bundles, shown in Figure 1. The cross sections of these bundles are shown in Figure 3. Note the Douglas Point and NPD 19-element bundle cross section also shows the positions of the inter-element spacers and bearing pads, and the (solid) outline of the end plate which holds the elements in the bundle assembly. All bundles have similar spacers and end plates, omitted from the other cross-sections for clarity.

Figure 3 gives the nominal bundle powers and the mass ratio of UO2 to Zircalloy. While the power has increased from 220 kW (NPD bundle) to 900 kW (Bruce bundle), the mass ratio has decreased from 11.1 to 9.4. This will be redressed in future bundle designs such as the graded bundle, or reduced sheath-thickness bundle-designs.

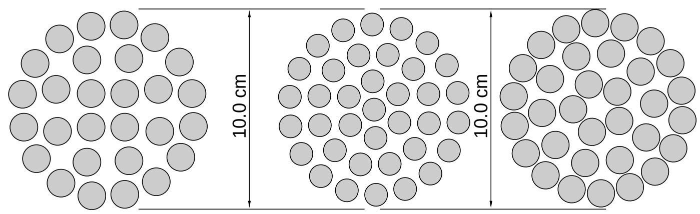
The four designs of CANDU fuel bundle in current production are:
- 28-element bundle for the Pickering reactors;
- 37-element bundle for the Bruce and Darlington reactors:
- 37-element bundle for the CANDU-6 reactors.

The principal requirements and designs of these bundles are similar. The requirements which have the strongest influence on these designs are:
- fuel channel size;
- specific power requirements;
- the fuel-handling machine detailed requirements.

Figure 3
Basic Data for CANDU Fuel Bundles



Reactor	N.P.D.	N.P.D. & Douglas Pt.	
Number of Rods/Bundles	7	19	19
Rod Diameter mm	25.4	15.25	15.22
Nominal Power	220	221	420
Mass Ratio UO_2/Zircaloy	11.1	10.2	10.1



Reactor	Pickering	Bruce & 600MW		Graded
Number of Rods/Bundles	28	37	37	33
Rod Diameter mm	15.19	13.08	13.08	13.30/14.48
Nominal Power	640	900	800	-
Mass Ratio UO_2/Zircaloy	11.1	9.4	9.4	10.6

Note:
End plate and interelement spacers and bearing pads are shown for NPD and Douglas Point bundle, omitted from other bundles for clarity

The NPD bundle was designed to operate inside an 80 mm pressure tube at a bundle power in excess of 220 kW, and to interface with the fuel-handling system of the "latch" type - also used in the Bruce reactors. The Pickering bundle uses the same element size as the NPD bundle (15 mm diameter), but in a 28-element configuration. The larger bundle, 100 mm diameter, can operate at a power in excess of 700 kW. In this case, the fuel-handling system is of the same generic design ("side-stop" type) as that of the CANDU-6 reactor. The CANDU-6 reactor bundle has the same overall dimensions as the Pickering bundle but uses a smaller element diameter (13.08 mm, the same as that for the Bruce bundles) to allow more than 1000 kW to be produced by one bundle (2). However, from a performance and licensing viewpoint the bundle designs are similar.

Features of the CANDU fuel design, and the year of their first use in production fuel, are shown in Figure 4.

2.2.1 Fuel Design Characteristics

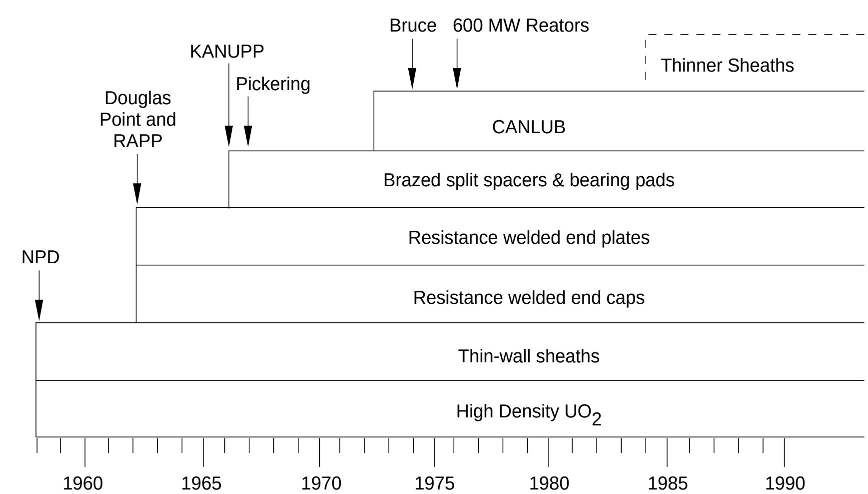
An important characteristic of CANDU fuel elements is the use of thin-walled Zircalloy fuel sheathing. The minimum wall thickness of 0.38 mm has been used continuously since a portion of the first charge for NPD used this thickness. The sheath properties must be controlled during manufacturing to ensure retention of the desirable properties. For example, corrosion resistance, ductility, strength are affected by manufacturing processes such as brazing. Because this thin sheathing is designed to collapse under the normal operating pressure of the coolant, it was recognized from the start of the program that pellet-clad interaction (PCI) would occur.

PCI can cause unacceptable sheath strains, for example, because of in-reactor densification of the UO_2. Thus, the need was established for high density "irradiation-stable" fuel pellets which avoid this problem. High density UO_2 fuel pellets, typically 97% of theoretical density, have been used since the beginning of the program. Control of density during fabrication is essential to maintain good thermal behaviour of the fuel. Hence fission gas release is controlled within the typical low operating ranges for CANDU fuel.

High density fuel, together with the later introduction of Canlub coatings of the inside surface of the sheath have maintained the potential PCI problem at an insignificant level in CANDU fuel. The Canlub coating reduces the effects of pellet-clad interaction, which are greatest during a power ramp. Such ramps can occur during a reactor power manoeuvre and more normally during refuelling. There is also evidence to suggest that the Canlub material acts as a chemical "getter" for the corrosive fission products, and is an important aspect of the fuel's resistance to stress corrosion cracking (SCC), particularly at high burnups. High quality, consistent Canlub coatings are required to maintain good resistance to SCC and good capability to handle power ramps.

Figure 4

Evolution of CANDU Fuel Features



To assist in maintaining low centre temperatures and hence low gas release rates from the high-powered UO_2 pellets, a fine surface finish must be provided on the pellets and sheath inside surface. This maximizes the heat transfer between the UO_2 pellets and the contacting sheath.

Prior to end cap welding each element is filled with (typically 100%) helium. Although used for leak detection during fabrication, the helium also enhances the pellet-sheath contact conductance at the start of life.

The end plates hold the fuel elements together in a bundle configuration. The end plates are made as thin as possible to minimize the amount of neutron-absorbing material in the assembly and minimize the gap between the UO_2 fuel in adjacent bundles. Recent experience shows that the end plates must allow differential expansion between elements in the bundle and resist fatigue cracking due vibration of the elements and bundle caused by the high coolant flows typical of modern reactors. These opposing requirements must be satisfied by careful design of the end plates and control of fabrication parameters such as weld positions and strengths to ensure adequate performance in the reactor. The resistance weld to each end cap is the weakest structural link and the first to break if the bundle is dropped or mishandled.

The bundle is supported on bearing pads, approximately 2.5 mm x 25 mm, which are brazed near the ends and at the centre of each outer element. The pads support the bundle in the fuel channel and fuel handling system, and prevent mechanical damage to the sheath. The pads are designed to minimize surface damage to the pressure tube caused by sliding, fretting, or corrosion. Positioning of the pads affects vibration performance of the fuel in the reactor. At the same time the pad positions are governed by fuel handling requirements.

The surface shape and condition of the pads affects how the pads interact with the pressure tubes. Interaction occurs during refuelling (sliding) and during normal operation (fretting and corrosion). Although advanced pads are being studied (e.g. low heat flux pads), careful fabrication of the current design of pad will ensure minimal damage to the pressure tube due to these effects.

The introduction of these and other design features, such as brazed spacers, (Figure 4) shows that the basic designs of fuel bundle have remained unchanged since the introduction of Canlub in 1973.

2.3 Development in Support of Fuel Design

When a new fuel design is to be introduced, an experimental irradiation program is initiated. Depending on the complexity of the changes in the fuel design compared to past experience, the program would be supplemented as required by in-reactor single element tests, out-reactor tests, fuel fabrication development tests, and may or may not include material property or development tests. In any case, prototype bundle irradiations are initiated many years ahead of each new CANDU reactor design.

An extensive program of fundamental work has been underway in Canada for many years; over 1,000 individual fuel elements, some highly instrumented, and over 500 bundles, some with demountable outer elements, have been extensively tested in in-reactor loops. The work has included fundamental tests on UO_2 irradiation properties, material properties of Zircalloy sheathing, and critical heat-flux tests in Freon and high-pressure steam and water loops. The current CANDU fuel designs were thoroughly tested at the prototype stage and have demonstrated excellent performance in the many commercial CANDU power reactors. Type tests which were used in the past are discussed below.

2.3.1 In-reactor Type-tests

Capsule irradiations were begun at the Chalk River Laboratories (CRL) many years ago in the early 1950s. These were followed by element testing of nominal diameters:

- 20 mm (used in NPD first charge),
- 15 mm (NPD, Douglas Point and the Pickering reactors), and
- 13 mm (Bruce and 600 MW reactors).

The element designs were then used in prototype bundles which were irradiation tested in the loops of the NRU reactor at CRL. The following aspects were included in these tests:

- irradiation at high element ratings to high burnups,
- dimensional stability with fission gas pressure above coolant pressure,
- power boost performance and fuel defect performance,
- element length, diameter and bowing, and their effect on dimensional stability of the bundle,
- Zircalloy material properties including hydriding and oxidation, and internal stress corrosion cracking,
- UO_2 performance (grain growth, swelling, and fission gas release),
- performance of the graphite Canlub layers,
- fretting and corrosion of the outer elements.

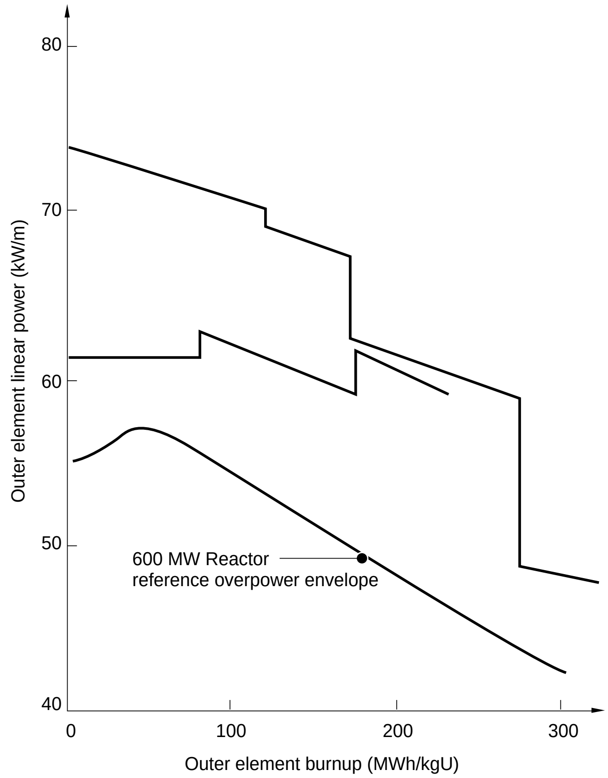
As part of the Canadian reactor test program, two high-power, 37-element prototype bundles were tested to high burnup. Figure 5 illustrates the power histories for these test bundles, and the reference overpower envelope (see sub-section 2.3.3 for explanation of power "envelopes"). The tests were designed to prove fuel performance at constant high power, and to study the effects of fission-gas pressure on sheath strain. The bundles performed very satisfactorily and the tests proved that they have excellent dimensional stability and corrosion performance, and acceptable fission-gas release even at power levels and to burnups which exceed the power-reactor requirements by large margins.

Another prototype reactor fuel bundle was irradiated at a conservative power rating of 35 kW/m (outer element rating) to a burnup of 85 MW.h/kgU, well within the range in which power-ramp defects are possible. The outer elements from this bundle were then power ramped in three groups to different power ratings. The maximum rating was 80 kW/m which was expected (40%

probability for Canlub fuel) to produce some defects. None occurred. The outer elements operate at the highest power, since the flux is depressed toward the centre of each bundle.

Figure 5

Power Histories for Prototype Bundles



2.3.2 Out-reactor Test

For each reactor, the fuel bundle design has been extensively tested in out-reactor full-scale loops located in the Sheridan Park Engineering Development Laboratory in Toronto, and at other test facilities operated by Canadian industrial consultants. The test objectives were to ensure that the bundles can interface with the reactor components under representative conditions of coolant pressure, temperature, and flow. The following tests were done:

- hydraulic head-loss characteristics,
- pressure-tube and inter-element spacer fretting,
- bearing-pad and pressure-tube sliding wear, and
- strength (against re-fuelling loads and seismic events).

Pressure drop tests are done in a full-length test channel and in a short rig to measure the effect of random angular alignments of the bundles, see Figure 6 which shows typical bundle-junction pressure drop signatures. The data from all tests are analyzed to produce the average "design" pressure drop for the 12 bundles in the fuel channel, assuming a random loading position of every bundle.

Endurance tests are done typically for 3000 to 5000 hours. Fretting damage to the bundles (i.e. to the inter-element spacers and bearing pads) and to the pressure tube is assessed. A series of tests uses several bundles to measure sliding wear. A typical surface profile, taken circumferentially across a wear "track" at the bottom of a test pressure tube, is shown in Figure 7.

Strength tests are done to simulate typical fuelling-machine loads. The bundles are tested to loads that give a margin of 1.5 above the design loads. Impact tests simulated the refuelling impacts which can occur as a new bundle is carried into the fuel channel by the coolant flow (CANDU-6 and Pickering reactors). These tests were done at velocities significantly higher than anticipated in the reactor. The results showed no significant deformation of the test bundles. The impact performance of the bundles, therefore, was found to be fully acceptable for the expected in-reactor refuelling conditions

Seismic testing has been done on a complete pressure-tube assembly which contains twelve fuel bundles in cold water to give a margin of 1.5 above the required design basis values. The Electric Power Development Co. Ltd. of Japan also used a full-scale single fuel-channel model and a nine-channel model, Figure 8. This work verified the seismic capability of the CANDU core [5].

In the Canadian nuclear industry all design changes are thoroughly evaluated and tested out-reactor and in experimental reactors before being introduced into an operating power reactor or new reactor. An example of interest was the introduction of Canlub coatings following the startup of the Pickering reactors in 1971/72. In this case, an extensive research effort which supplemented in-reactor testing, resulted in the successful identification and solving of the power-ramp defect problem.

The emphasis of the current development programs is to provide more comprehensive information on fuel behaviour during hypothetical accidents. Future development will focus on the fabrication and performance of advanced fuels which include fuels containing thorium and plutonium.

Figure 6

Typical Bundle-Junction Pressure Drop Signatures

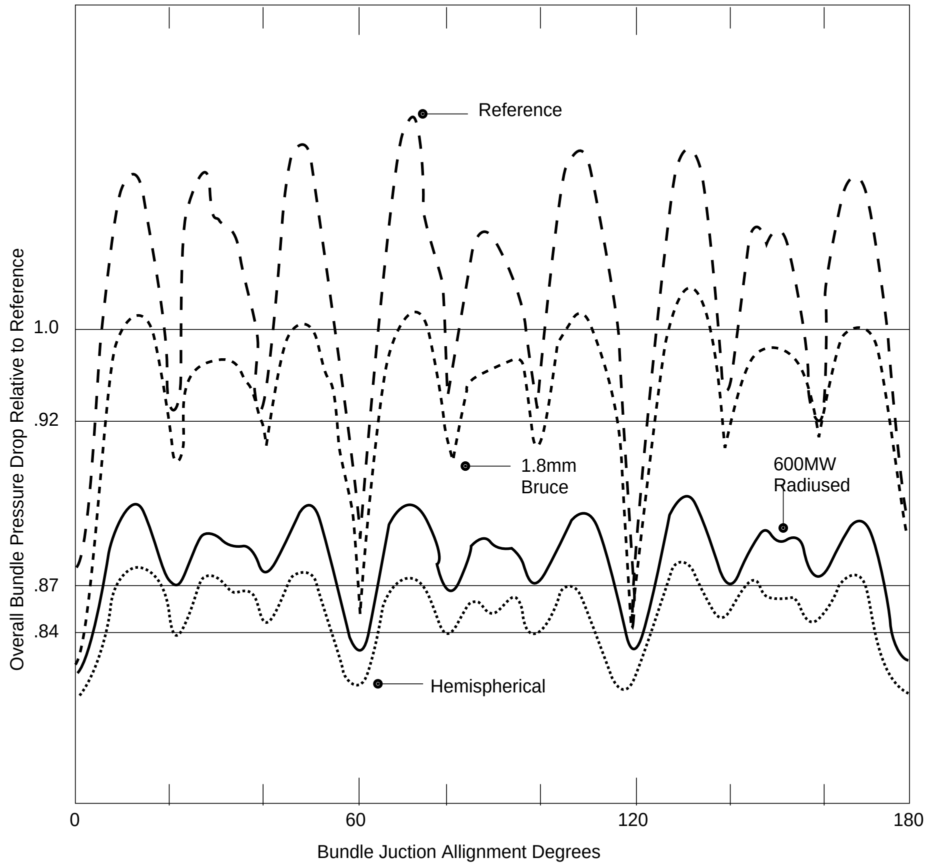


Figure 7

Wear Track from Test Pressure Tube Surface

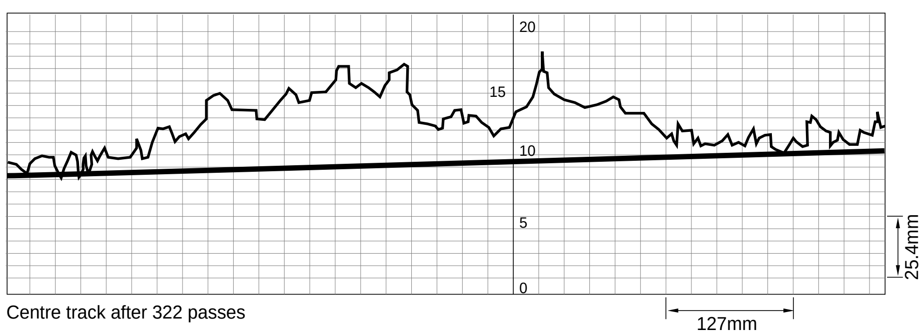
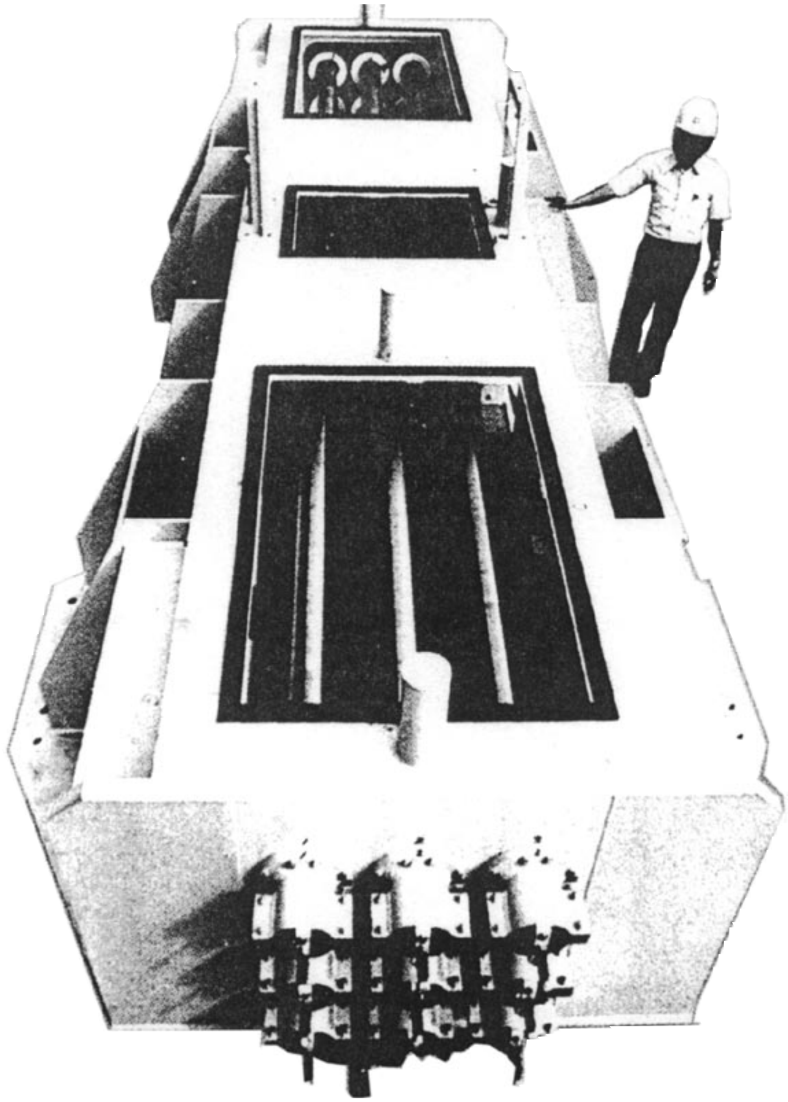


Figure 8
Fuel Channel Assembly Vibration Test Rig - Nive Channel Mode



2.3.3 Computer Codes

Fuel performance and evaluation codes have been developed from theoretical correlations and empirical test data. The codes are design-centre codes and they have been verified over the range of conditions expected for a power reactor. For example, the fast-running fuel performance code ELESTRES and other codes are used to evaluate fuel performance under normal conditions and transient conditions, which include loss-of-coolant accidents.

Fuel operating guidelines for a new CANDU reactor design are set to encompass the normal operating envelopes (including power perturbations caused by on-power refuelling) which are predicted by fuel management simulations. Fuel performance within these envelopes and for the predicted coolant conditions is then evaluated using these codes.

The ELESIM and ELESTRES computer codes are designed to model the behaviour of CANDU fuel elements under normal operating conditions. FUELOGRAMS are used to predict power-ramp performance.

The ELESTRES code was developed from the ELESIM fuel performance code; it models a single fuel element by accounting for the radial and axial variations in stress and displacements. The physical (as opposed to empirical) models in ELESTRES include such phenomena as fuel/sheath heat transfer, temperature and porosity dependence of UO_2 thermal conductivity, burnup dependent neutron flux depression, burnup and microstructure dependent fission product gas release, and stress, dose, and temperature dependent equations for the sheath. The finite element model includes thermal, elastic, plastic, and creep strains for the UO_2 pellet deformation, and the effects of swelling and densification, pellet cracking and the rapid drop of UO_2 yield strength with increasing temperature.

Because of the short length of the CANDU PHWR fuel bundle, the input data are chosen for the centre of the reactor for the maximum power region of the core. Actual power histories are used for assessing the performance of test fuels. To provide a basis for fuel design and analysis, however, curves of bundle power versus burnup have been derived from physics simulations, and compared with actual operation. A "nominal design" bundle power envelope encompasses the entire "time-average" bundle-power distribution in the reactor core.

Higher individual bundle powers may be encountered in "young" channels, i.e. those that have been refuelled recently or those that are going through their plutonium peak. These effects also can produce higher powers in adjacent channels. Local power variations also can be caused by the zone controllers. Therefore, a "reference overpower" envelope was generated to account for these additional effects. The two envelopes are illustrated in Figure 5.

In practice a small percentage of the bundles in the core operate above the nominal design power envelope, and less than 1% of the bundles operate above the reference overpower envelope, and then only briefly.

To be conservative when assessing power-reactor fuel designs, the assumed fuel history follows the nominal design power envelope initially and then increases to the reference overpower envelope for a limited time to a conservative maximum burnup of 360 MW.h/kgU. Typical fuel-element linear power histories are shown by curves 3, 4 and 5 in Figure 9. Figure 10 shows the variation of fission gas pressure with burnup for the five power histories, as predicted by ELESIM. Note that the gas pressure is below coolant pressure for all cases.

In both these codes, fuel temperature is the dominant variable, since many of the physical processes, for example, fission gas release and fuel expansion, are strongly dependent on temperature. In turn, the fuel temperature can be significantly affected by the fuel-to-sheath heat-transfer coefficient (interface conductance). In the case of CANDU fuel, initial clearances are small and the sheath is designed to contact the pellet, so at power the "gap" equals the surface roughness. This means that the thermal conditions can be modeled with great certainty for a CANDU fuel element; the temperature-dependent subroutines (for example, fission-product gas release) and the routines controlling temperature (for example, interface or gap conductance) can be checked rigorously against CANDU fuel data.

Figure 11 shows a comparison of predicted versus measured per cent fission gas release for ELESIM. The calculated values, making allowance for a ± 5% uncertainty in power, compare well with experimental values. Figure 12 shows ELESTRES predictions of plastic sheath strains. The agreements with measured plastic sheath strains are reasonable.

Figure 9

Outer Element Power Histories for the Nominal Design Power and Reference Over Power Envelopes

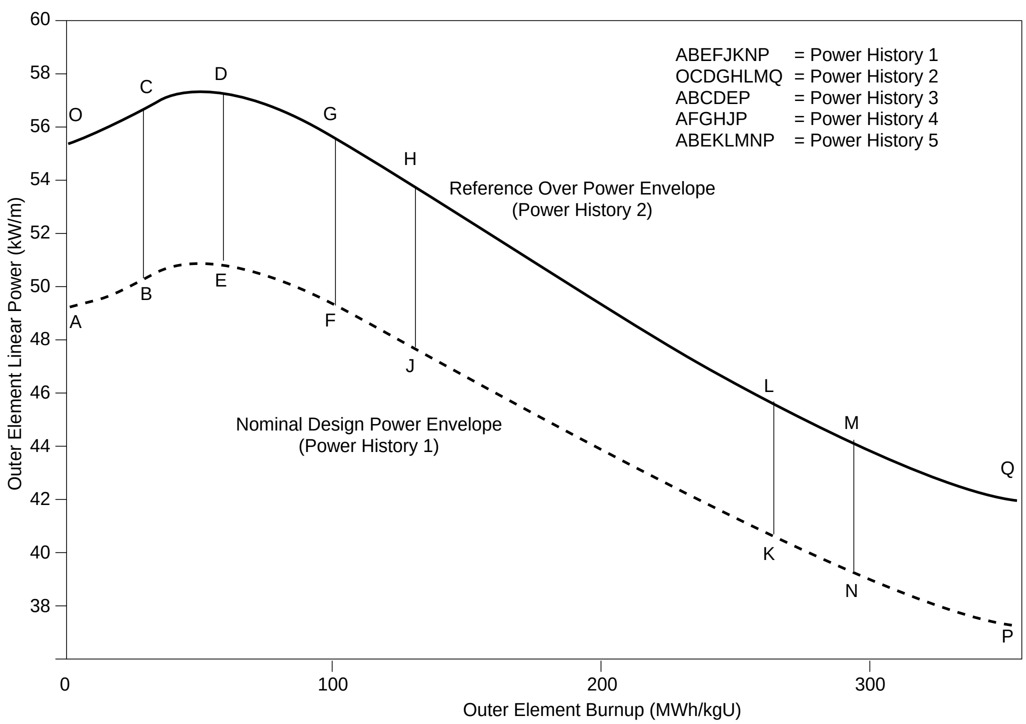


Figure 10

Variation of Calculated Outer Element Fission Gas Pressure with Outer Element Burnup for Various Power Histories Shown in Figure 9

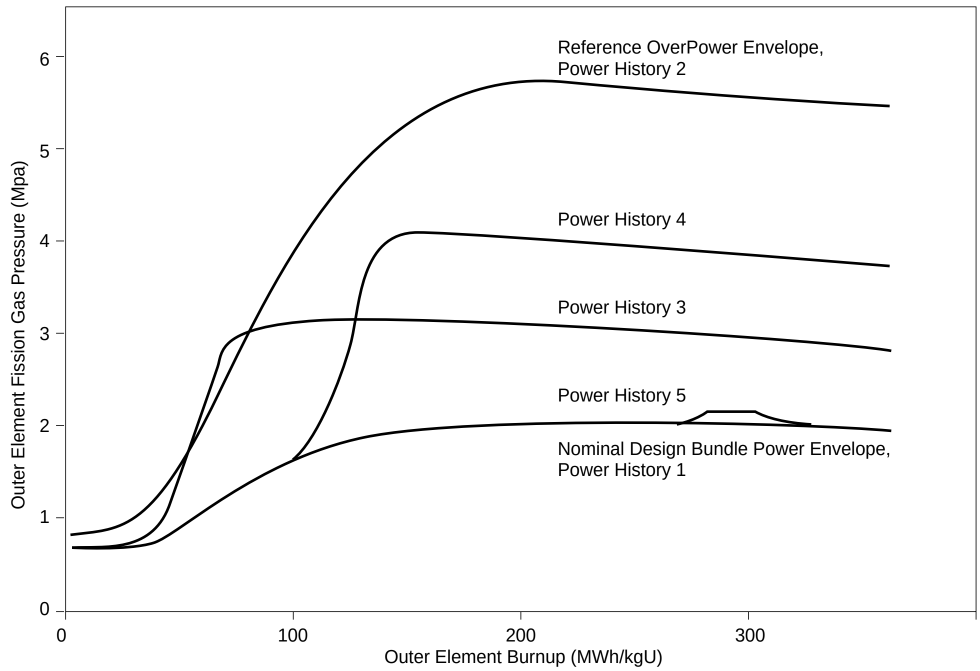


Figure 11

Comparison of Measured Fission Gas Release from Power Reactor and Experimental Fuel Compared with that Calculated by ELESIM

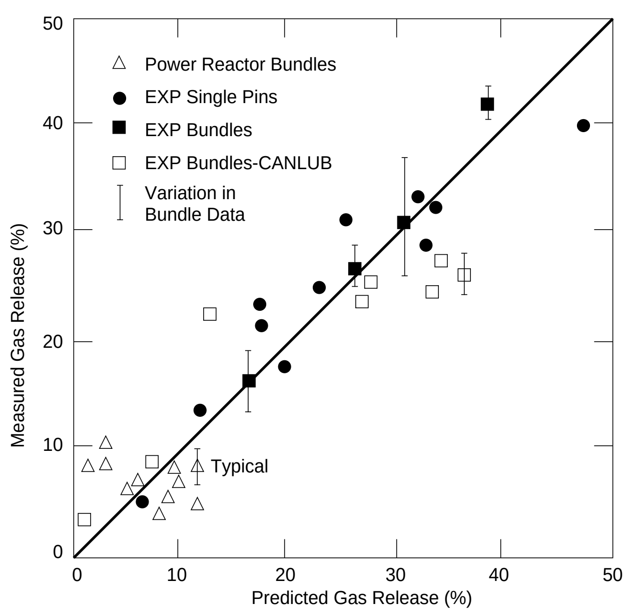
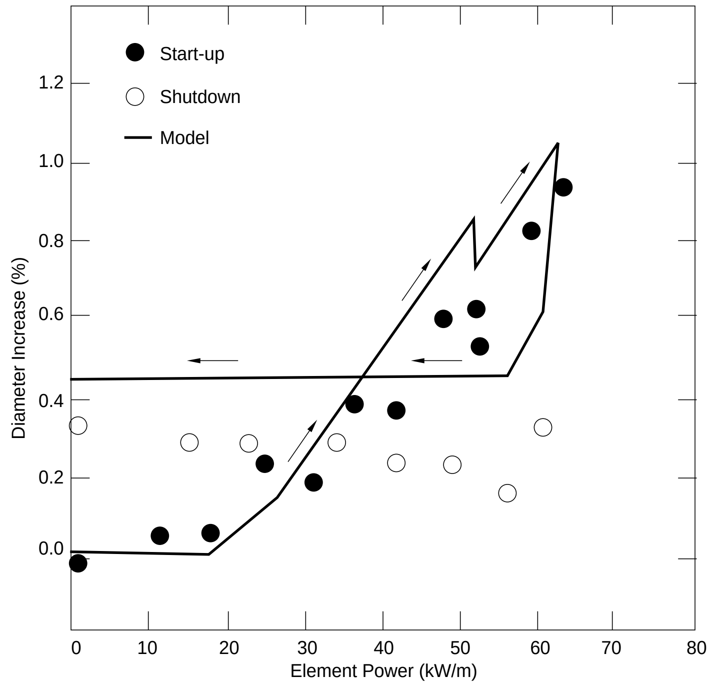


Figure 12
ELESTRES Calculations of Sheath Strain compared with Values Measured in Reactor

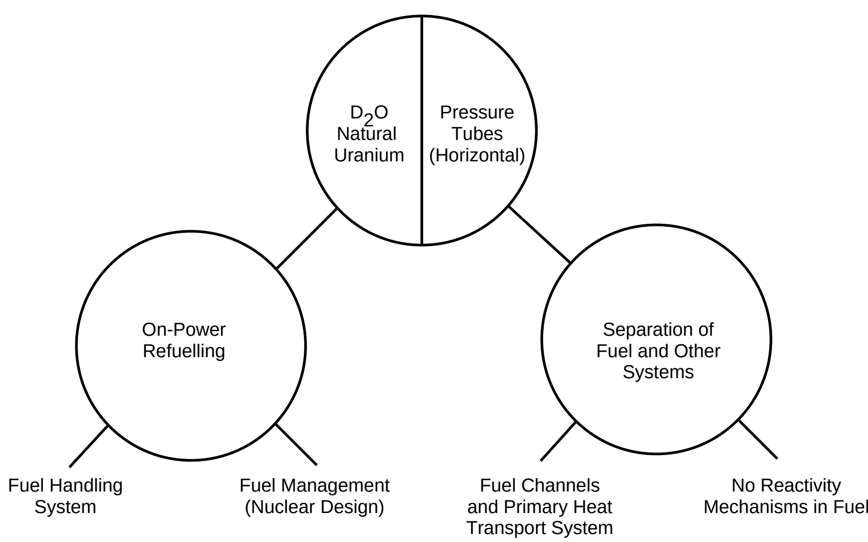


3. Design Requirements

3.1 CANDU Pressurized Heavy Water Reactor - Generic Features

The hierarchic order of the fundamental characteristics of the CANDU reactor, and their influence over the specific reactor features are shown schematically in Figure 13.

Figure 13
CANDU Reactor, Generic Features



Two fundamental characteristics are:
- D_2O moderator; this permits the use of natural uranium in the fuel, and
- pressure tubes rather than a pressure vessel, which are used as the primary pressure boundary.

Two characteristics, which are direct consequences of the above but which have become so intrinsic to the successful operation of these reactors, also qualify as fundamental characteristics. They are:
- on-power re-fuelling, and
- separation of the high-pressure coolant from the low-pressure and low-temperature moderator.

These fundamental characteristics have determined the evolution of the reactor systems with which the fuel interfaces, and which in turn have determined the fuel design requirements.

3.2 Fuel Design Requirements

Structural components are not required in the fuel bundles because they are supported in the core by the pressure tubes. The bundles are small, simple assemblies. This allows them to be moved within the fuel channels by the on-power, fuel-handling system. Because the fuel elements are horizontal; gravitational pellet relocation or "ratcheting" cannot occur, even if pellet densification should occur. The separation of the reactivity regulating system from the fuel means that control rods or other regulating features such as variable enrichments or poisons are not needed in CANDU fuel assemblies. As a result, it is possible to have a single bundle design for the entire core.

Separation of the heat transport system (HTS) from the moderator allows the chemistry of each system to be optimized for that system. In the HTS, the two major requirements for chemistry control are:
- to minimize dissolved oxygen to ensure low corrosion rates of the Zircalloy components, and to avoid pitting of carbon steel and stress-corrosion cracking of austenitic alloys of the end fittings
- to maintain alkaline conditions to achieve acceptably low corrosion rates, and to minimize transport and activation of corrosion products.

3.2.1 Heat-transport System

Figure 14 is a simplified schematic of this system. The coolant mass-flows vary from a nominal rate, in a typical central channel, of 23.9 kg/s to about 10 kg/s in the outer channels, Figure 15.

The fuel bundle design must be compatible with this system with respect to pressure drop and vibration, for example, and must be able to withstand power changes due to refuelling. The five design requirements imposed by the primary heat-transport system are:

- The pressure drop over the fuel bundles in each fuel channel must be compatible with the allowance in the design of the primary heat-transport system.
- Fretting damage caused by the coolant flow must not reduce the inter-element spacer height below an acceptable level. For example, the high flows must not induce excessive vibration of the fuel, particularly at the inlet end of the fuel channel. The HT flow rate at Darlington had to be reduced because of fuel and pressure tube fretting damage during early operation.
- The reduction in the wall thickness of the pressure tube by the bearing pads, due to fretting and other effects, must not jeopardize pressure tube integrity.
- The ability of the fuel elements to contain fission products during normal operation (defined here as including the effects of on-power re-fuelling and adjuster rod movements) must be compatible with the capabilities of the purification system.
- The fuel design must be evaluated to determine its ability to withstand the normal operating pressure, and hydrostatic pressure testing of the primary heat-transport system.
- The fuel bundle design must accommodate element bowing, and fretting and sliding wear without significantly affecting bundle thermal performance, and the bundle design must ensure that the fuel element surfaces remain wet under normal operating conditions.

3.2.2 Fuel Channel and Fuel Handling System

A typical fuel channel is shown in Figure 16. During re-fuelling, the bundles slide along the liner tubes and for short periods, radial coolant flow can induce inter-element vibration.

Separator assemblies in the fuelling machines are used during re-fuelling to detect the gap between pairs of bundles, to prevent movement of the bundles in the channel (due to the coolant flow), and to push the bundles in pairs into the magazine of the fuelling machine.

The fuel-channel design and the fuel-handling system together impose the following geometrical and performance requirements on the fuel bundle design, additional to the five listed in sub-section 3.2.1:

- Bundle bearing pads must allow the bundle to slide inside the fuel channel without jamming on the discontinuities in the channel and the fuel-handling system components. Bearing pads must maintain design clearances between the fuel-element and pressure-tube surfaces during normal operation. Wear of the bearing pads also must not cause interference with fuel-handling operations. Inter-element spacers must not become interlocked due to thermal bow or due to normal axial forces in pressure tubes with and without radial creep. Interlocking is most likely to occur during manual handling. If interlocking occurs the bundle diameter becomes larger than the channel inner diameter resulting in jamming, fuel "gauging" is used to guard against this.

- Bundle ends must be compatible with the operation of the sensors and separators on the fuelling machine (CANDU-6 and Pickering reactors), and the fuel latches and fuelling machine systems (Bruce and Darlington reactors).
- Fuel bundles are designed with allowances for irradiation swelling so they will pass through the worst combination of diameter and misalignment expected in the fuel channel. The maximum force required to push the bundle through the channel (over and above the friction force due to bundle weight) must be estimated.

Figure 14

Heat Transport System

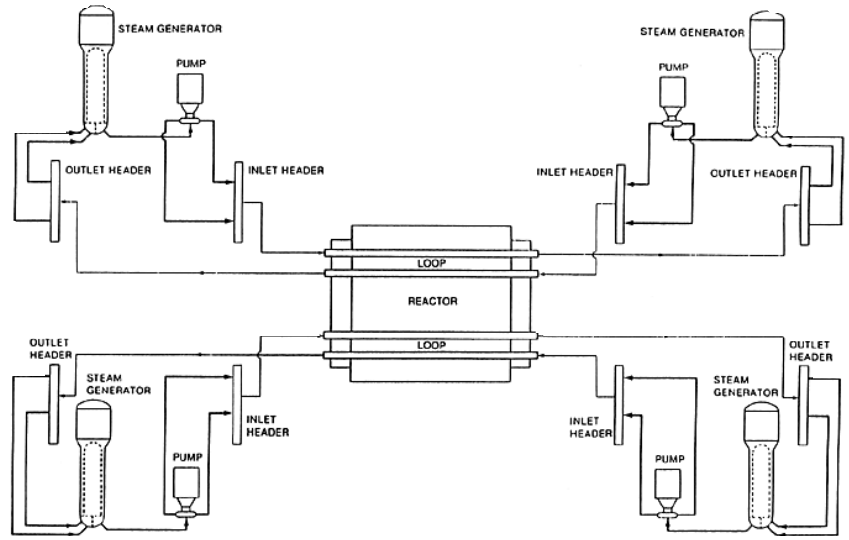


Figure 15

600 MW Reactor Core-Loading of Depleted Fuel and Range of Coolant Mass Flows

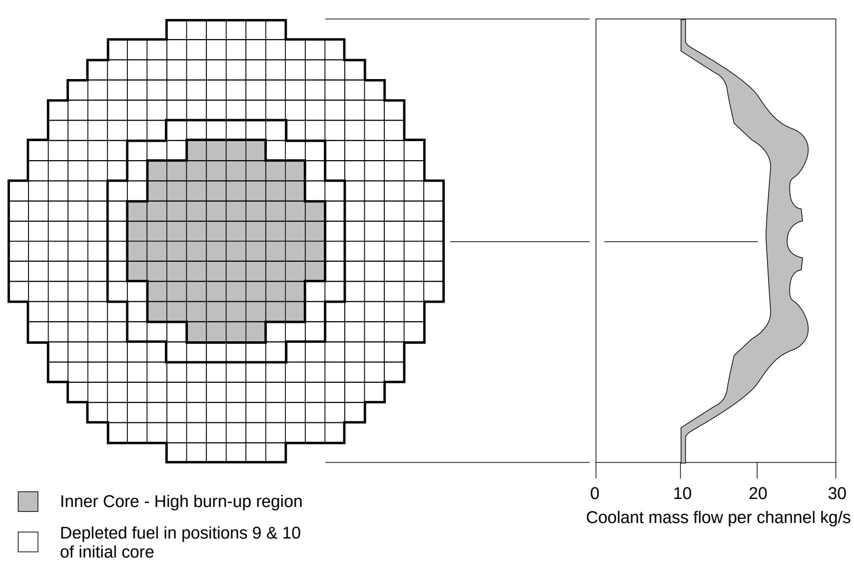
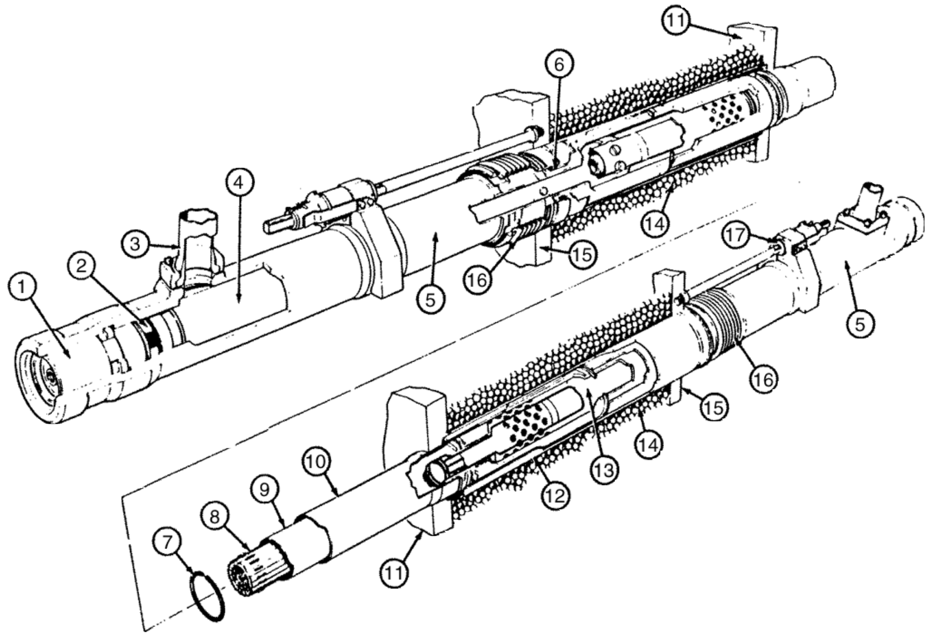


Figure 16
600 MW Reactor Fuel Channel Assembly



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		

- Damage to the pressure tube due to normal sliding wear, fretting and crevice corrosion under the bearing pads of the fuel bundle must be minimal and less than the design allowances.
- A fuel bundle has to have the strength and flexibility to maintain its structural integrity under steady and fluctuating loads applied to it thermally, mechanically, hydraulically, by vibration, and by fuelling operations in fuel channels (including the effects of creep and sag).
- Fuel bundles must be able to withstand (when contained in the pressure tube) the combined axial loads caused by hydraulic drag and the fuelling-machine ram. Bundles must also withstand all normal fuel handling loads applied by sidestops, ram, shield plug and/or pusher, or fuel latches and carrier tubes, as applicable to the different reactor types, without significant dimensional changes and without degradation in performance. A typical allowable ram force is 8,896.5 newtons.
- Fuel bundles have to withstand the maximum flows predicted during fuelling in axial flow, radial flow, and combined axial and radial flow at the outlet feeder, liner tube for representative refuelling dwell times. The crossflow may occasionally be prolonged when fuelling problems arise. (ie) Fuel failure will occur in one to two days under crossflow conditions.

- Fuel design has to provide adequate thermal performance in fuel channels that have crept radially and axially.

3.2.3 Fuel Management

Fuel management in CANDU reactors refers to the on-going or daily replacement of fuel in the reactor core, and its effects on power distributions and fuel burnups.

Because these reactors are re-fuelled with the reactor at full power, in-core fuel management is substantially different from that of batch-loaded reactors which must be re-fuelled during scheduled shutdowns. In CANDU reactors, fuel management is very simple. All fuel bundles are identical, and re-fuelling operations are normally carried out daily so that re-fuelling is essentially continuous. By adjusting the fuelling rate in different regions of the core, the power distribution in the core can be continuously controlled and modified over a period of several weeks, if desired.

Usually, four or eight fresh bundles are added to the channel and four or eight bundles are discharged (each channel contains 12 bundles - or 13 in the Bruce reactors). However, the flexibility of CANDU on-power re-fuelling permits different re-fuelling strategies. For example, four-bundle shifts are used for the inner channels and eight-bundle shifts are used for the outer channels of the core. Other strategies can be used, for example, to maximize fuel usage of the first fuel charge by shuffling fuel removed from one channel into a second channel.

In any re-fuelling scheme, adjacent channels are re-fuelled from opposite ends of the core, so that fuelling is bi-directional. Thus, the global distribution of irradiation in the core is symmetrical, resulting in symmetric axial flux and power distributions.

During re-fuelling the bundle powers will vary. Increases in bundle power, or power boosts, are experienced by some of the bundles which are shifted to higher power positions. These bundles will remain in the reactor for two or more re-fuelling cycles. For example, a bundle initially in position 1 (low power) will be moved to positions 5, 9 (high power) and finally position 13 (Bruce and Darlington reactors). The highest power boost is for the bundle that moves from position 1 to 5. Note that this bundle will have reached a very low burnup when its power is boosted.

During normal operation, adjuster rod withdrawal from the core, or "reactivity shim" addition, will produce power boosts in the fuel bundles adjacent to the adjuster rods that are removed.

Some flux peaking occurs at the ends of all fuel bundles because the Zircalloy material and heavy-water coolant absorb fewer thermal neutrons than UO_2.

Larger peaks can occur during fuelling, when the inlet bundle enters the core as the fuel string is shifted towards the outlet end of the channel (CANDU-6 and Pickering reactors).

These features of on-power refuelling lead to the following fuel management (or nuclear design) requirements:

- The fuel elements must be held together in the required bundle configuration by end plates. (Also, the end plates must be strong enough to ensure the bundle's structural integrity and must be flexible enough to allow differential axial expansion among the elements.)
- The end plates have to be as thin as possible to minimize the neutron absorbing material in them (good neutron economy), and to minimize the separation between the UO_2 fuel in adjacent bundles to minimize the end flux peaking in the fuel elements.
- The fuel bundles must be designed to operate at high power continuously.
- The fuel bundles must withstand power changes caused by re-fuelling, by power ripple and by reactivity shim operations. Either outer element linear ratings must stay below the Canlub defect probabilities for power increases due to refuelling or reactivity shim operation, or the ratings must stay below the Canlub defect probabilities for maximum steady power (see Section 6 Fuel Performance).
- To permit the evaluation of fuel performance under normal operating conditions, it is necessary to derive a high bundle power versus burnup envelope (or equivalent) encompassing 99% of all bundles in the core at any one time. The envelope should include refuelling ripple and the effects of reactivity devices. Fuel is not allowed to operate outside the high power envelope during other power transients, for example reactivity shim operation.

3.2.4 Defective Fuel Location System

Depending upon the reactor design, the system may be supplied with a gaseous fission product (GFP) monitor and a delayed neutron (DN) monitor or a feeder scanner system.

The GFP monitor is a gross radioactivity monitoring system that can detect the presence of defective fuel in the core by monitoring the bulk coolant activity in each of the two reactor loops. In addition, the purpose of the GFP monitor is to measure the ^{131}I activity in the HTS.

Sample lines, typically one from each half of the HTS, are taken past a gamma spectroscope to allow measurement of four isotopes of interest, ^{131}I, ^{133}Xe, ^{135}Xe and ^{88}Kr, as well as gross gamma. The activity levels are transmitted to the station control computers, and can be displayed on operator demand.

The GFP monitor is designed to have a minimum sensitivity of 0.5 MBq/kg for the isotopes of interest in the bulk coolant. Automatic alarms and updates of activity levels are provided.

The DN monitor is used as a failed fuel location system, and is able to locate the particular channel that contains the defect. The location system is a "per channel" system which extracts, on demand, a continuous sample from each feeder and brings it through sample lines into sample rooms. There, the lines are coiled into sample holders arranged in matrix form. A carriage holding the detectors scans the holders for delayed neutrons emitted by I-137 and Br-87 that are released by the fuel to the coolant. When a channel containing a defect bundle is being defuelled the delayed neutrons emitted from the defective fuel are used to identify the defect bundle pair. Under some conditions the bundle pair or the actual bundle with the defect can be identified. The DN monitor system is able to detect a defect that exposes to the coolant only 2×10^{-4} to 5×10^{-4} m^2 of UO_2 surface (including cracks, connected porosity, etc., within the UO^2 pellets) without the need for power cycling (which enhances the sensitivity) or operation at low reactor power during the scan period.

The feeder scanner system is an off-power system to identify deposited fission products in the feeders adjacent to the core. This system is used only when the reactor is shut down as it can not discriminate between readings due to the high background radiation levels during operation. It has never proven to be an effective method of finding failed fuel.

An important feature of CANDU on-power refuelling is that fuel defects can be removed from the core as soon as they are identified (or whenever convenient). It is, in fact, normally recommended that the operator remove a defect as soon as it begins to release uranium. In this way the reactor operator is able to keep the contamination and radiation levels in the HTS very low. For example, radioactive emissions in 1983 from the Pickering A and Bruce A nuclear generating stations were less than 0.01% of the license limit for I-131.
To maintain low levels of background radioactivity, and to improve the sensitivity of the detection systems, the surface contamination on the as-manufactured fuel bundles must be less than 200 micrograms uranium per square metre.

3.2.5 Seismic Qualification of Fuel Bundles

For each reactor site different seismic requirements are specified. Therefore, the components and structures of the reactor are designed for the design basis earthquake (DBE) for the site. This ensures that the reactor is capable of being shut down and maintained in that state indefinitely, the continued removal of heat produced by the fuel, no breaching of the heat-transport system to lead to a loss-of-coolant accident, and radioactivity levels are within licence limits.

The seismic design requirements, in turn the fuel bundle must be able to withstand loads due to a design basis earthquake at the site while maintaining a coolable geometry, and without jeopardizing the heat-transport system, pressure boundary. Consequently, the fuel bundle must be seismically qualified to ensure a coolable geometry and the integrity of the fuel sheaths to contain fission products.

3.3 Design Requirements Imposed by the Fuel on Other Systems

The fuel interfaces with each of the major reactor systems. During design of these systems the interfacing requirements imposed by the fuel must be considered. The following major systems and requirements imposed by the fuel are considered:

- fuel management (nuclear design) requirements;
- the heat transport system;
- the fuel channel;
- the fuel-handling system;
- the defective fuel location systems; and
- transportation and storage systems.

Design requirements imposed by the fuel bundle design on the above systems are discussed in the following sub-sections.

3.3.1 Requirements Imposed by the Fuel on the Nuclear Design

The reactor physics and fuel management schemes must not cause systematic fuel failures. This means that:

- bundle powers and individual element powers must not cause central melting;
- in conjunction with the heat transport system, the reactor physics and fuel management systems must ensure that fuel surfaces are kept wet under all normal operating conditions including refuelling transients;
- either bundle powers or the bundle power increases due to refuelling or the reactivity devices must stay below the fuel defect probabilities for maximum continuous power or power increases;
- Defect probability and flux peaks between adjacent bundles are implicitly allowed for in the curves. The large exposed-bundle-end flux peaks occurring during fuelling are not allowed for. To prevent fuel failures due to central melting or other causes the local element power must be restricted, if required, by using local flux suppressors;
- to permit the evaluation of fuel performance under normal operating conditions, a high bundle power versus burnup envelope encompassing 99% of the bundles in the core at any one time must be derived. It should include refuelling ripple and the effects of reactivity devices;
- the fuel must not normally be allowed to operate outside the above limits during other power transients, for example reactivity shim-mode operation.

3.3.2 Requirements Imposed by the Fuel on the Heat Transport System

The HTS must not cause systematic fuel failures. This means that:

1. the system must provide sufficient flow to keep the fuel surfaces wet under all operating conditions;
2. normal operating pressure should be limited to a certain range;
3. maximum pressure during cold hydrostatic pressure tests must be specified;
4. debris in the coolant system must be minimized during:
   - construction,
   - commissioning,
   - operation,
   - inspection of HTS,
   - maintenance, and
   - modifications;
5. coolant chemistry must be controlled within the following typical ranges:
   - P.H.: 9.5 - 10.5
   - dissolved deuterium: 25×10^{-3} dm^3 D_2/kg D_2O maximum
   - dissolved oxygen: 50 µg/kg maximum
   - crud level: 100 mg/kg D_2O Maximum

3.3.3 Requirements Imposed by the Fuel on the Fuel Channel

The fuel channel design must have an adequate allowance to cover fuel sliding-wear, fretting and crevice corrosion.

The fuel channel design has to allow for axial thermal expansion of fuel for all operating and shutdown conditions.

The fuel channel must be compatible with the geometry of bundles with irradiation swelling.

The fuel channel must provide passage and support for the bundle's bearing pads under all conditions. Vertical step height must not exceed 0.5 mm. Axial gaps in support surfaces should not exceed 20 mm so that bearing pads can bridge these gaps.

Fixed and moving fuel channel components must be designed to minimize fuel vibration during normal conditions and during refuelling.

Components such as the perforated liner tube and feeder nozzle in the outlet end fitting must be designed to minimize fuel vibration and to avoid excessive radial forces on the bundle.

The ram or ram adapter of the fuelling machines must be compatible with the fuel bundle design.

Pushers and shield plugs must apply axial loads to the entire surface of at least the outer two end plate rings of the fuel bundles.

The fuel channel must withstand the effects of fuel bundle impacts during refuelling.

The fuel channel components must provide adequate coolant access the fuel bundle's flow subchannels.

The design of the fuel channel hardware must ensure that the mechanical consequences of a feeder or header break do not aggravate the thermal consequences of the break. This requires that there is:
- no significant flow blockage,
- no structural damage to the fuel, and
- no propagation of the channel or feeder failure.

In addition, for header breaks, the above requires that there is:
- no significant fuel failure, and
- no breaching of the pressure boundary.

3.3.4 Requirements Imposed by the Fuel on the Fuel Handling System

Provision must be made for new fuel to be visually inspected for damage, to be checked for interlocking and to have the serial numbers verified.

The fuelling machine sensors (feelers) and carrier tubes and latches, as applicable, must not damage the fuel bundles.

The fuelling machine sidestops must not contact the bundles when they are driven in or withdrawn. (The load should be applied to the sidestops only when the fuel moves axially). The sidestops should only contact the surfaces of the end caps, and should do so in such a way as to minimize the eccentricity of the axial load on the element. A minimum of six end caps should be fully contacted, three on each side of the bundle.

Fuelling operations must be such that no torque is applied to the fuel bundles at any time during the refuelling cycle.

The coolant temperature in the fuelling machine containing irradiated fuel must not be less than 30˚C during fuelling sequences.

The fuelling machine must be capable of withstanding the effects of fuel bundle impacts against the ram or sidestops.

The normal refuelling operation must not expose any bundle to crossflow at each refuelling for a period longer than has been shown to be acceptable in tests.

New fuel bundles can cause severe gouging of new pressure tubes when they are first loaded. To prevent this damage the bundles must be loaded while resting on metal shimstock which is then removed.

When the fuelling machine contains irradiated fuel to be reloaded into the core, it must not normally be depressurized. (Depressurization causes the sheath to release its grip on the cracked pellets and permits relocation of pellet fragments. This causes a minor reduction in fuel performance which is acceptable in abnormal events but not for routine events).

The irradiated fuel transfer system generally keeps all fuel surfaces wet, however, exposure to air for less than one minute is permitted so that fuel can be transferred out of the heavy water system and into the fuel bay. Emergency water sprays or flooding are needed if the fuel transfer system fails to submerge the bundles in the bay.

3.3.5 Requirements Imposed by the Fuel on the Instrumentation Systems

To minimize station dose, a delayed neutron system is normally installed and a gaseous fission product system permits the prompt on-power location and removal of small defects (before they begin releasing uranium). In some reactors a feeder scanner system permits off-power location of deposited fission products which would indicate a fuel defect.

The delayed neutron system must be designed for routine scans and must also be designed for convenience of single-channel operation during refuelling, so that small defects can be prevented from re-entering the core. (Defects residing in low-power positions can release uranium and yet be difficult to locate even in fairly clean systems).

The purpose of the gaseous fission product system is to warn the operator when defects occur so that they can be located quickly. The system must therefore be designed to provide continuous monitoring with alarms.

3.3.6 Requirements Imposed by the Fuel on the Transport And Storage Systems

The fuel transport and storage systems ensures that the fuel meets all dimensional and any other requirements after receipt at the site.

New bundles are stored in their shipping containers in a clean, dry and secure area. Natural uranium fuel requires no special spacing or amount limits to meet critical mass safeguards, that enriched fuel would require.

Irradiated fuel must be cooled, shielded and contained throughout the complete fuel cycle. cooling capacity of water pool safety is based upon volume of fuel and age of fuel. Figure 17 is a decay heat chart of Pickering reference fuel. For example decay heat drops to about 5 watts/bundle after 5 to 10 years of cooling allowing storage of fuel at this stage in dry canisters where, air cooling is sufficient to avoid bundle overheating or failure.

Gamma radiation levels remain very high for a long time. Figure 18 illustrates dose rate vs time of cooling for a Pickering reference fuel bundle. In practical terms a minimum of 12 feet of water is always maintained in the storage bays above the highest positioned bundle, this depth of water attenuates the 10^6 mrem/hr fuel dose rates to less than 1 mrem/hr at the bay surface. Irradiated fuel retails large quantities of radioactive fission product and actinides for hundreds of years. Figure 19 illustrates the curie content versus cooling period for Pickering reference fuel. From this figure it can be seen that it takes 1000 years to reduce the curie content by a factor of 1000.

Figure 17

Decay Heat - Pickering Reference Fuel

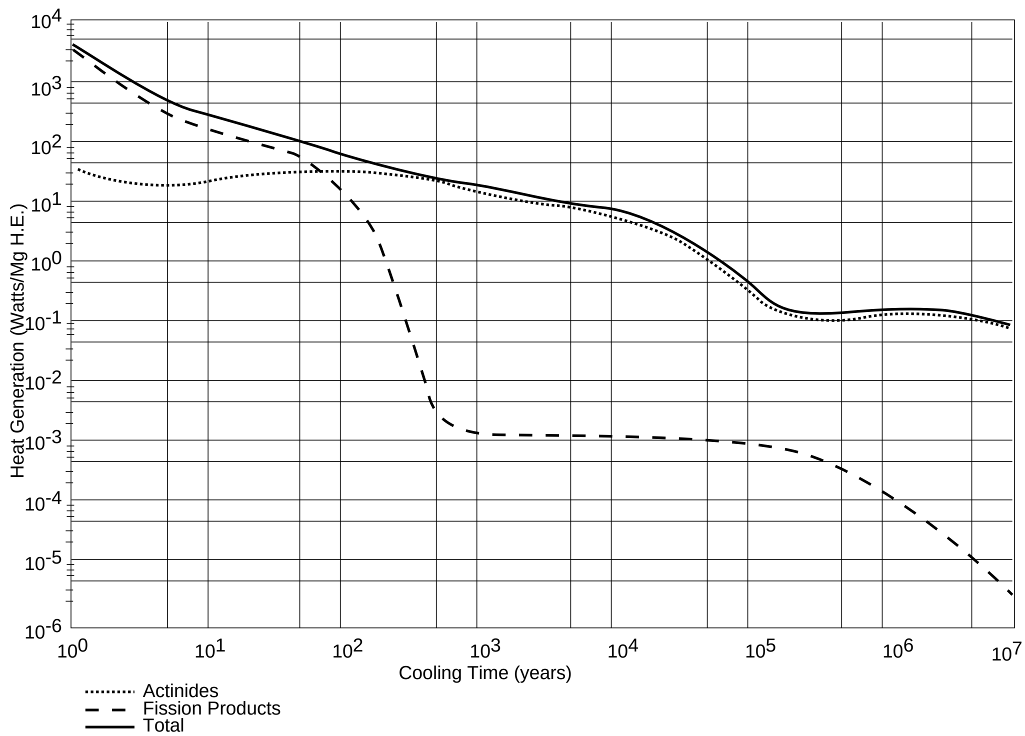


Figure 18
External Radiation Dose Rate One Foot in Air from Fuel Bundle - Pickering Reference Fuel

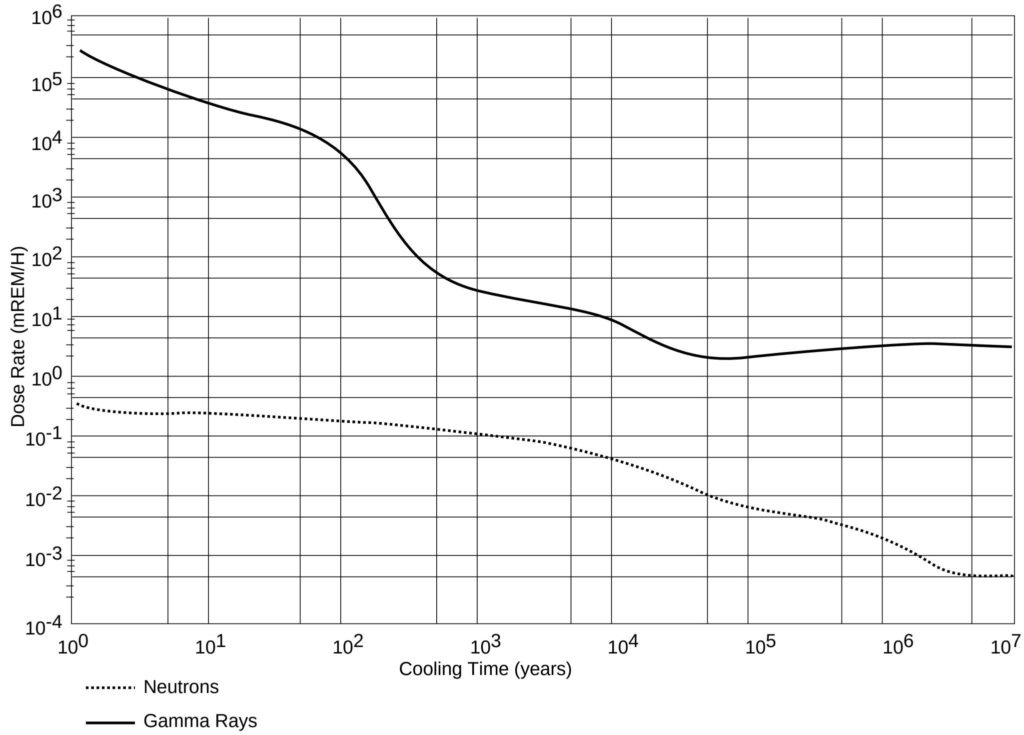
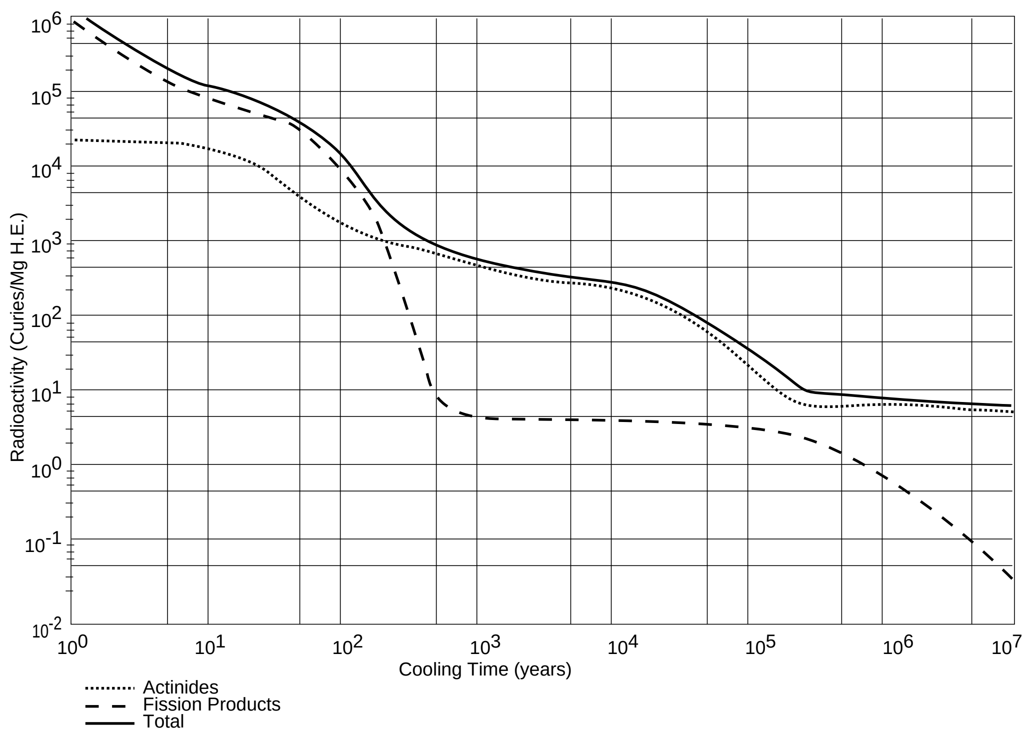


Figure 19
Fuel Radioactivity - Pickering Reference Fuel



4. Fuel Fabrication

4.1 Manufacturing Processes and Their Evolution

Development of manufacturing processes by the fuel fabricators has proceeded in parallel with evolution of the fuel design. The current manufacture of CANDU fuel bundles involves the following major operations:

- test the incoming UO_2 powder and Zircalloy materials;
- make high density UO_2 pellets (typically 97% of theoretical density);
- fabricate Zircalloy structural components (sheath sub-assemblies with appendages, end caps, and end plates);
- load the pellets into the sheath and weld the end caps to the sheath to complete the elements;
- assemble elements into bundles; and
- finally inspect, and package ready for shipping.

Westinghouse and Canadian General Electric are the two Canadian fuel fabricators. They purchase starting materials to conform to relevant "material" specifications. The fuel is then produced to comply with "product" specifications.

The first core load for NPD used inert-gas fusion welding for the end-cap to sheath welds, and for the bundle assembly (elements to end-plate) welds. Subsequently, all CANDU fuel has been manufactured using resistance welding for these joints. This method of welding is fast, inexpensive, and easily automated.

The original NPD bundles used wires wrapped in spirals round each element. These wires were attached by spot welds. The wires promoted coolant mixing between subchannels within the bundles, and helped maintain separation of the elements from each other and from the pressure tube. A short time later the spiral wires were replaced by spacers and bearing pads, brazed to the sheaths.

Today, manufacturing emphasis is on improved productivity and process control. The use of automation and advances in machine control technology, for example using microprocessors, are contributing to improvements in the already high-quality product, and to the low fabrication costs of the fuel.

4.2 Technical Specifications

Comprehensive technical specifications have been developed in parallel with the CANDU fuel development program for all of the basic materials, for intermediate subassemblies and products, for example, the finish-ground UO_2 pellets, for the Canlub coating, brazed joints, and the fuel bundle itself. A simple bundle design drawing specifies overall dimensions and gauging requirements. In the early years, the specifications were based essentially on the fundamental research work; more recent improvements have been made using up-to-date manufacturing and reactor performance data. Because of the accent on neutron economy in CANDU fuel, a key feature of these "material" specifications is the strict control of impurities in the finished product, particularly those impurities with large neutron-capture cross-sections.

The fuel fabricators have developed their own "product" specifications which are used to define the "as-built" details and quality control programs. These have been developed over several years by each fabricator, and they are based on his particular fabrication techniques.

4.3 Processing of Concentrates

In Canada, the conversion of U_3O_8 concentrate into UO_2 powder is not considered a fuel "fabrication" process. UO_2 powder is traditionally produced at a refinery which is separate from the fuel fabrication plants.

Uranium concentrates are obtained from a variety of sources in Canada. The concentrates are refined by solvent extraction, and a nuclear grade, high-purity powder is produced by the ADU process involving precipitation of ammonium diuranate complex, followed by calcination to produce a ceramic grade UO_2 powder. In recent years considerable effort has been made by Cameco and the Canadian fuel fabricators to define the characteristics of the powder, and the means to consistently and economically produce it.

4.4 UO_2 Pellet Production

Uranium dioxide powder is received from the refinery in quantities that are suitable for planned production rates. Prior to using a specific lot, an evaluation sample is processed to pellet form. The parameters of the processes are altered as required, and the pellets are examined for conformance to specification. If the pellets made from the evaluation sample meet the specification, the lot is accepted by the bundle fabricator.

The UO_2 powder is compacted into wafers or slugs. The "slugs" are broken up and passed through a fixed-size sieve. This process increases the bulk density of the powder and improves the flowability.

The granular material is fed to a pill press (mechanical or hydraulic) where it is pressed into "green" pellets to predetermined weight, diameter and height. A statistical sampling plan is used to confirm that the process is producing "green" pellets that conform to the specification.

The "green" pellets are placed in containers or boats which are continuously stoked through a sintering furnace to produce UO_2 densities of better than 96% theoretical density.

Samples of the sintered pellets are checked for diameter, height, density and physical defects. Pellets are also checked for chemical analysis, and a metallographic examination is carried out.

The sintered pellets are ground to the required diameter on a centerless grinder, are washed to remove the grinding coolant and sludge, are dried, and are inspected dimensionally and for surface imperfections using statistical sampling plans.

4.5 Fabrication of Components

When the Zircalloy strips and bars (for spacers, bearing pads, end plates, and end caps) are received they are inspected for dimensions and surface defects, and mill certificates provided by the vendor are checked for compliance with the material specifications. In addition, the end caps are machined, normally on an automatic screw machine. A statistical sampling plan is used to confirm that the end caps are finished to the required dimensions.

4.6 Fabrication of Elements, and Bundle Assembly

The Zircalloy sheaths for the fuel are inspected on a statistical basis. Visual and dimensional inspections, including those for straightness, are done, and the mill certificate from the vendor is checked against the specification.

The sheaths are then ultrasonically tested for transverse and longitudinal defects, sorted by inside-diameter size range, and degreased.

The bearing pads and spacers are tack welded to the sheaths, after which the brazing operation is performed. The brazing process exploits the low melting point (approximately 980˚C) eutectic formed between zirconium and beryllium at 5 weight % Be; the structural brazed joint is formed by local heating of the joint area using a rapid vacuum induction heating method. Joint quality is verified by visual examination and periodic destructive examination of samples. Completion of the sub-assembly requires the introduction of a thin layer of graphite on the internal surface of the fuel sheath; the process involves liquid coating of the sheath interior with a graphite suspension followed by drying and baking to provide a thin, continuous graphite layer.

After sheath coating, the UO_2 fuel pellets are loaded into the fuel sheath and hermetically sealed by welding the pre-machined Zircalloy-4 end caps to the ends of the sheaths. These welds, together with the sheaths, form part of the primary containment of the UO_2 fuel against release of fission products to the reactor coolant, or ingress of coolant to the fuel. The end cap welds are made using a resistance, forge-welding process; precise control of component geometry, surface condition, chemical environment and weld cycle are essential for reproducible, high-integrity welds. Control of the welding process is verified by periodic destructive examination of sample welds and 100% leak testing of all elements; this is done using helium leak detection methods capable of detecting a leak rate in the order of 10^{-8} ml He/second. The sensor gas is introduced at the welding operation.

The final bundle assembly starts with individual elements being placed in a fixture which permits unambiguous positioning of inner, intermediate and outer elements. These are anchored in place by resistance welding the end caps to each end plate. The finished bundle is subjected to a series of dimensional and gauge checks, helium leak testing and is finally washed, packed and shipped to the reactor site.

4.7 Transportation to Reactor Site

The method of packaging and shipping CANDU fuel bundles is to use simple 36-bundle cardboard boxes (subdivided into individual bundle compartments) which are loaded onto a truck for movement within Canada. For a typical overseas shipment, these boxes are placed inside wooden crates and are loaded into regular freight containers, leased from the shipping company.

4.7.1 Packaging Methods

The fuel bundles are sealed individually in air-tight polyethylene bags. They are placed next in styrofoam "egg-crate" modules which form a secondary container and which are used principally as impact/vibration absorbers during transportation. Thirty-six bundles are contained on a single wooden pallet to form a compact unit for handling by fork-lift truck, Figure 20. A typical arrangement for shipping overseas is shown in Figure 21 in which two of the 36-bundle boxes are loaded side by side into a wooden "export" crate.

4.7.2 Transportation Methods

The wooden export crates are loaded into ocean-going ISO containers (eight crates to one 20-foot container) at the fuel fabrication plant. The containers are transported by "air-ride" truck to the port of exit. There the container is loaded onto a container ship for direct shipment to the chosen port of entry.

5. Quality Assurance

5.1 Quality Standards

The quality standards used for fuel procurement in Canada, and the fuel technical specifications, have been developed over a period of 25 years. Initially the quality control requirements and technical requirements were specified in a single, fuel-specific document. As the technical specifications became diversified to include the requirements of the newer reactor designs, the quality standards also evolved. These quality standards were removed from the technical specifications and became non fuel-specific. More recently these standards have been incorporated into the Canadian Standards Association (CSA) Z-299 series of Quality Program Standards.

Figure 20
36 Bundle Shipping Box and Packing Details

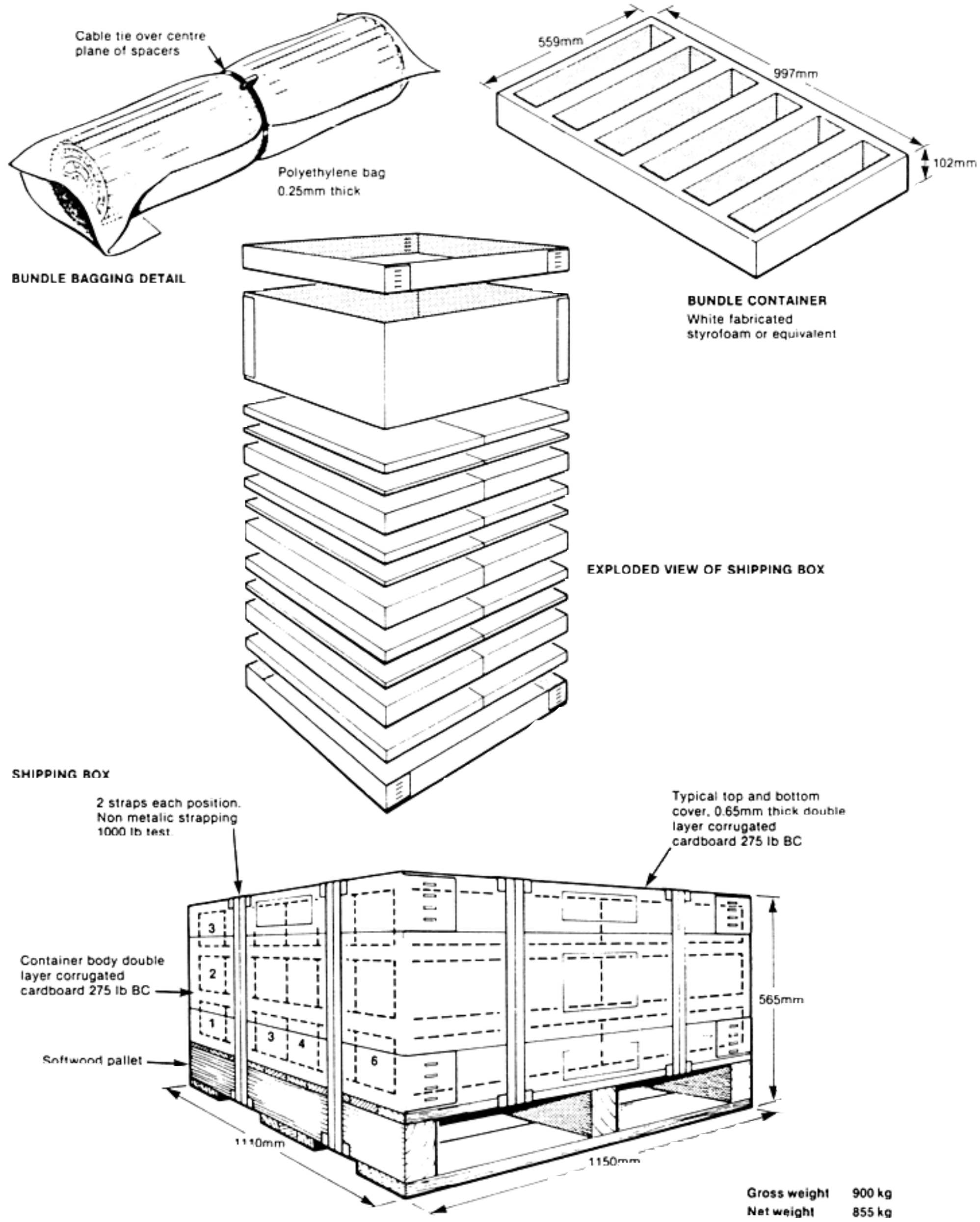
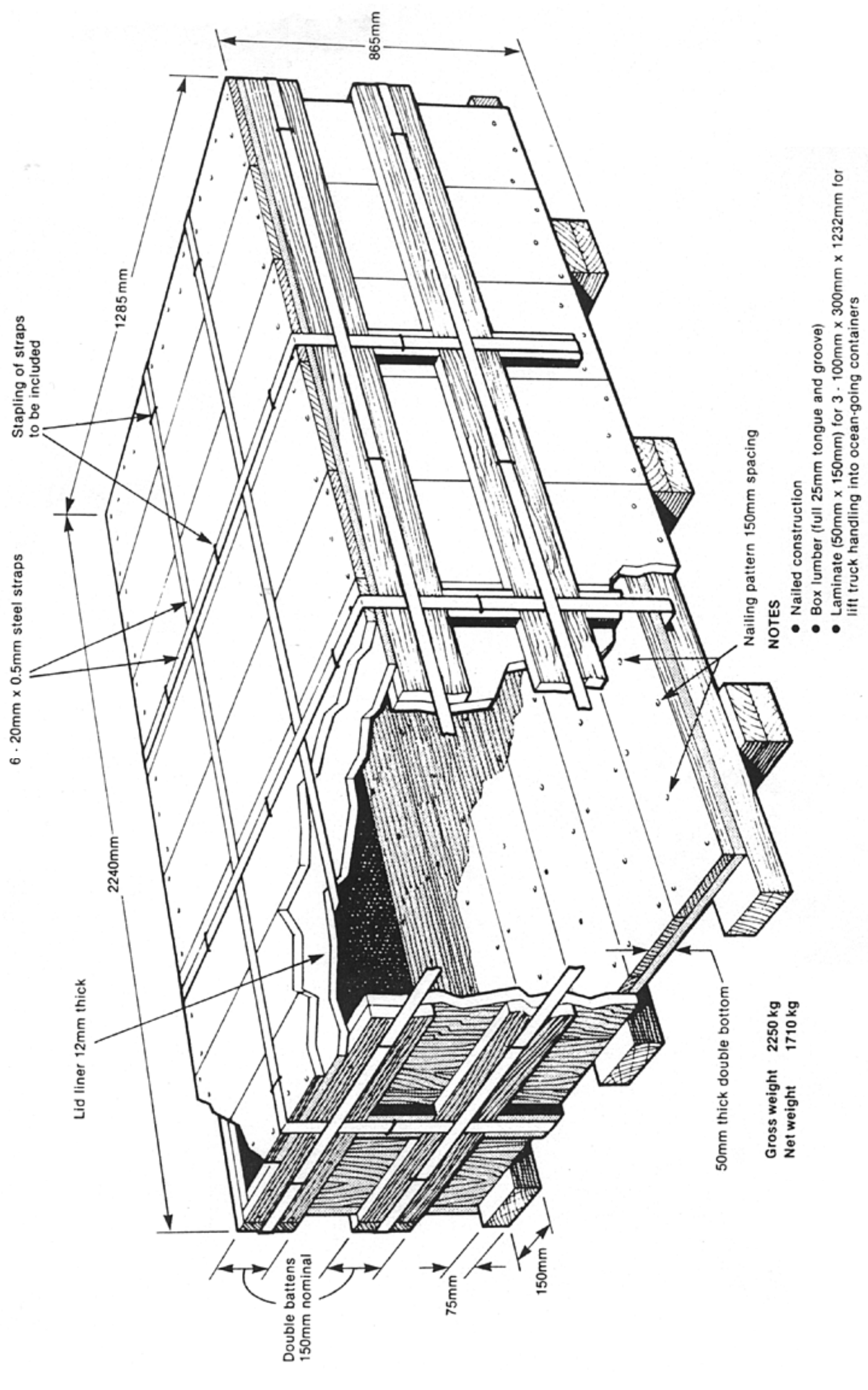


Figure 21
Typical Export Shipping Crate for 72 CANDU Fuel Bundles



5.2 The CSA Z-299 Series of Quality Program Standards

The CSA Z-299 series of quality program standards are built on the philosophy that every good quality program is constructed around the same elements: planning, performance and proof. There are four standards, each of which contains fewer or less stringent requirements than the one preceding it. Specifying a more comprehensive standard does, in general, provide the customer, and the fabricator, with greater assurance backed by documentary evidence, that the quality requirements are met and that the fabricator's quality program is implemented and effective.

The top quality standard is normally selected for the procurement of nuclear fuel. If the fuel is a re-order, however, (and it is usually an identical design) the second level standard may be used.

5.3 Implementation

Four aspects of quality program implementation are of interest to the purchaser:
- Quality Assurance Manual Acceptance and Program Audit,
- Quality Control Procedures Acceptance,
- Qualification of Special Processes, and
- Quality Surveillance and Product Acceptance.

1. Quality Assurance Manual Acceptance and Program Audit

Well before the start of production, the purchaser needs to establish that the supplier has an operating quality program which is also documented. The documentation is presented in a Quality Assurance Manual which normally consists of two parts, a Policy Manual and a Procedure Manual.

The Policy Manual typically describes and defines the inter-departmental relationships of the supplier's organization. This part of the manual defines who does what and when in order to ensure that product quality is maintained.

In the Procedure Manual the supplier is expected to define how each activity is performed. Some of the system functions which are defined are:
- review and approval of drawings
- audit of Quality Programs
- qualification of Quality Programs
- performance of internal audits.

Following a thorough review and acceptance of the Quality Assurance Manual, the purchaser performs an audit on the supplier's plant to verify that the program described in the Manual is being followed. An audit typically involves 5 to 10 clients' personnel and may take up to three days to complete. A report is submitted to the supplier.

2. Quality Control Procedures Acceptance

The supplier is required to supply the detailed inspection procedures, and a complete set of detailed manufacturing procedures for the purchaser's acceptance, and information and comment respectively. Although the manufacturing procedures do not require the purchaser's formal acceptance, they are checked to ensure consistency with the accepted design documents and the detailed inspection documents.

3. Qualification of Special Processes

In the production of CANDU fuel, a number of specialized processes are used which are subject to qualification. The intent is to demonstrate that the process is capable of consistently producing within the required tolerances. Following the acceptance of a qualification proposal, the supplier manufactures a qualification batch, carries out the required inspection and produces a qualification report.

In addition to demonstrating that the process is capable of producing consistent product, the data from the qualification batch is used to select/justify the inspection sampling plans which provide optimum economic control of the process.

4. Quality Surveillance and Product Acceptance

Each purchaser has a representative at the supplier's plant whose responsibility is to verify that all aspects of the quality program are implemented. To guide the representative in carrying out his duties, the purchaser prepares a "Surveillance Plan". In the plan, all of the key points in the supplier's operation are identified, and the verification action for each operation is specified. The representative reports his own and the supplier's activities to the purchaser as verification that the supplier is indeed adhering to his surveillance plan and, in the case of anomalies or deviations, reports these to the supplier and the purchaser. Finally, the purchaser's representative is responsible for reviewing the supplier's inspection reports, and accepting the product.

5. Other Quality Program Features

It is recognized that verification of implementation of the Quality Program requirements described above does not constitute the complete standard. Those activities were highlighted because they have particular relevance to the verification of the implementation of the program from the point of view of the purchaser. All the other features, except design assurance, are of a more general nature and their description and verification of implementation can be adequately inferred from the CSA standards. It should be noted, however, that all the features are mandatory for a fully qualified program.

One system function which has not been described but does merit special mention is "Design Assurance". From earlier discussion, it is clear that CANDU fuel is not designed, developed, produced, and used according to the well established cycle for commercial products. Rather, the designer/developer, the fabricator and the user participate together in the total cycle. Therefore, the "Design Assurance" function carries a significantly different impact for fuel than it does for other products; the overall design (excluding the manufacturing detail) is proven before it is submitted to the supplier for quotation. In fact the supplier is involved in the development of the design, and the specifications.

6. Fuel Performance

6.1 Irradiation Experience

Fuelling systems performance has been excellent with approximately 100,000 channel fuelling operations successfully completed and 99.9% of more than 1,000,000 fuel bundles (irradiated to the end 1992) operating as designed. Figure 22 shows the outer-element linear powers (the highest rating in each bundle) and burnups of 6240 fuel bundles in a Bruce A reactor taken at one instant in time (a "snapshot"). To interpret the graph in terms of performance statistics, multiply each point shown by 18, to give 112,320 data points. The effects of bundle, radial flux depression can be incorporated giving a total of 230,880 fuel-element data points. All elements at significant power experience a constant, well defined axial neutron flux. These data are extracted from reactor physics and re-fuelling calculations once or more per week and are stored on computer files for every irradiated fuel bundle.

This data can be processed (Figure 23) to show a typical outer-element power history for a bundle from a Bruce A high-power channel. The three distinct power levels in its history show that it resided in the "four-bundle shift" region of the core.

For Bruce A, Figure 24 shows the percentage of outer elements as a function of discharge burnup. Although the graph is based on only 257,896 outer elements (13,772 bundles) discharged during 30 reactor-months operation, it is a good representation for all outer elements of the more than 2,000,000 discharged from Bruce A. The peak at approximately 135 MW.h/kg(U) is from lower burn-up fuel discharged from the "eight-bundle shift" region. These are now being assessed for recycle into the four-bundle shift channels. Based on the same data, Figure 25 shows the percentage of outer elements which operated at the maximum power shown, as a function of discharge burnup. Note the large cumulative percentage above a rating of 40 kW/m.

Figure 22

Outer Element Linear Power vs Burnup for all Fuel Bundles in Bruce NGS-A, Unit 1

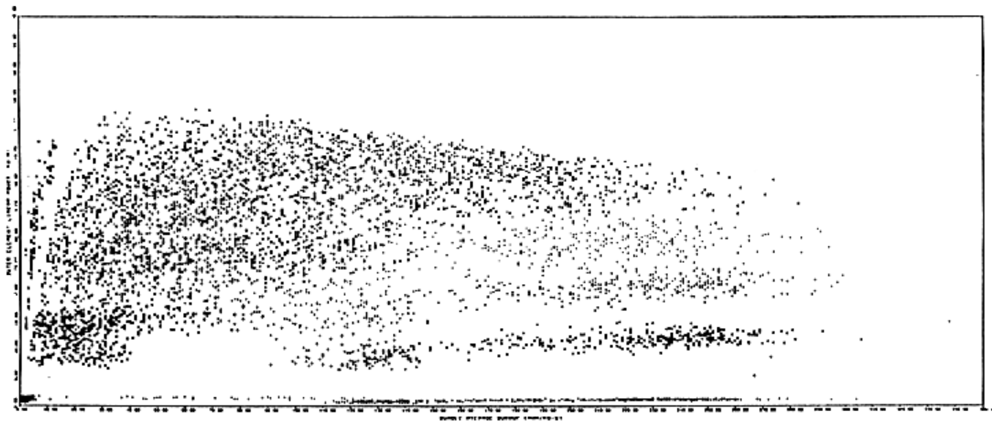


Figure 23

Outer Element Power History for Bruce NGS-A Fuel

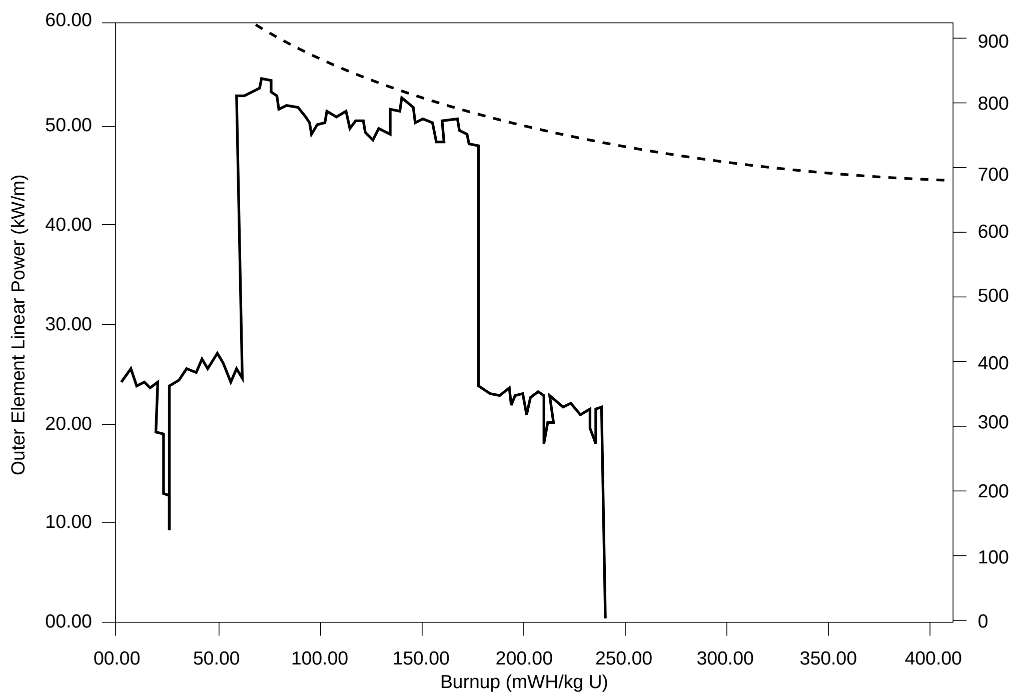


Figure 24

Bruce NGS-A, Outer Element Discharge Burnup Distribution

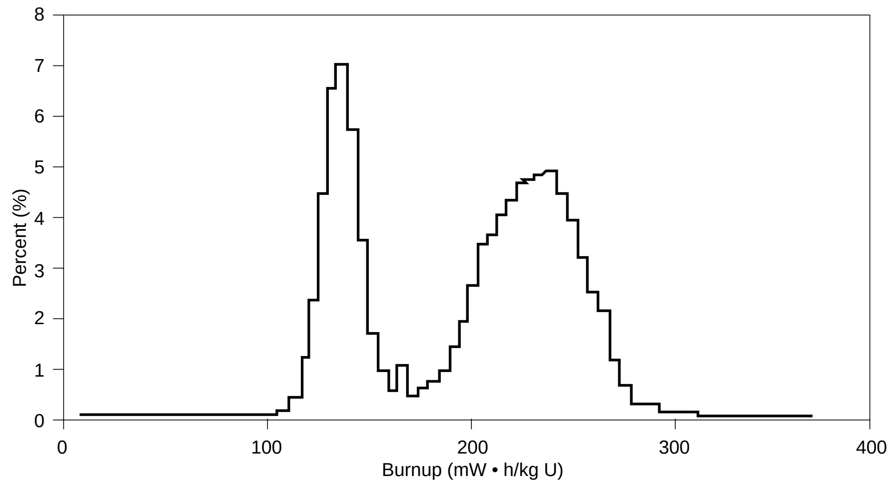
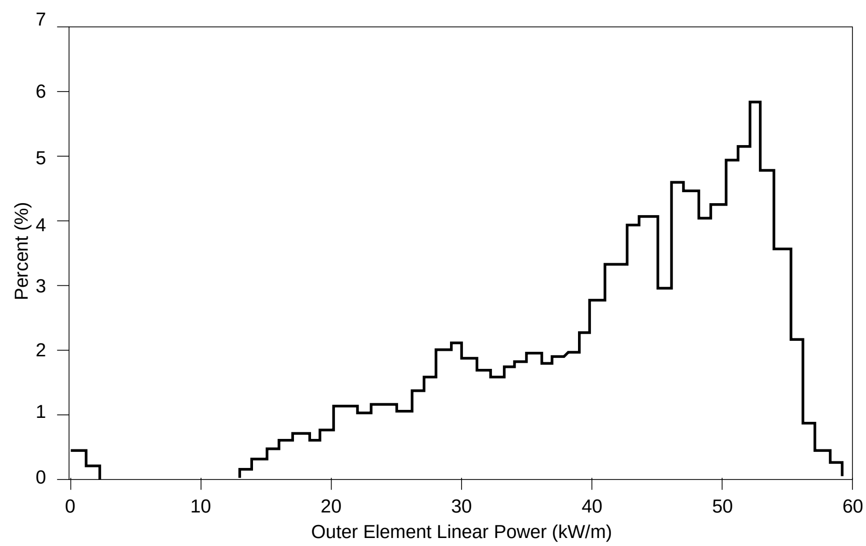


Figure 25

Bruce NGS-A, Outer Element Peak Operating Power Distribution for Discharged Fuel



6.2. Defect Statistics

A very significant benefit of the on-power refuelling feature of the CANDU reactor is that fuel defects have very little impact on reactor operation. When a defect is detected and located, it can, if necessary, be removed from the core without reducing power or significantly perturbing the operation of the reactor. If the rate of activity release for the defect is low enough, the defected fuel can be left in the core to be removed later, perhaps several weeks or longer, when there would be no disturbance of the normal refuelling schedule. Even the premature discharge of the defected bundle has negligible economic impact because the cost of the short, natural UO_2 CANDU fuel bundle is so low. Despite the fact that the defects have little impact on the operation of the CANDU reactor, CANDU fuel has been designed, built, and operated so that its defect rate is low.

Fuel bundles which leak fission products have occasionally operated in CANDU reactors for long periods of time. Small leaks or leaks in low-power positions may be below the threshold sensitivity of the defective fuel monitors. Also, some small leaks may become plugged as a result of oxidation of the Zircalloy. These types of leaks are considered inconsequential to reactor operation.

In some fuel defects, deterioration of the defect has resulted in fission product gases, UO_2, and zirconium alloy being released to the primary coolant. The UO_2 and radioactive fission products that are lost from the defective element do not create a hazard to either the public or the station operators. However, they do contribute to radiation levels in the plant and to maintenance costs of cleanup systems. Therefore, it is desirable to avoid gross deterioration of defective fuel elements.

When a defect is detected and located by the monitoring system, the fuel bundle can be discharged using the on-power refuelling system. Removal might not be immediate, for example due to other fuelling machine operational requirements of local reactor flux conditions.

There is a relationship between the rate of deterioration of defective fuel and the surface heat flux of bundle power. Generally, the higher the power of the defective bundle, the sooner the bundle should be discharged. (See Figures 26 and 27.)

There was a significant number of fuel defects experienced in the Pickering A reactors in the early 1970s following initial startup. Pickering reactors do not have individual channel defect location systems, however, iodine in the coolant indicated the presence of a number of defects. Following this incident, Canlub coatings were introduced. Since that time the defect rate has been insignificantly small.

An estimate of the current fuel defect rates was obtained from the Bruce A reactors which incorporate delayed neutron (DN) monitors. A total of 219 bundles had been designated as being defective, to give a cumulative defect rate of 0.12% . The current bundle defect rate is approximately 0.04% (or less than 0.002% on a single element or rod basis).

Figure 26

Power Increase (ΔP) vs Burnup Defect Probability Curves for Graphite CANLUB Power Reactor Fuel

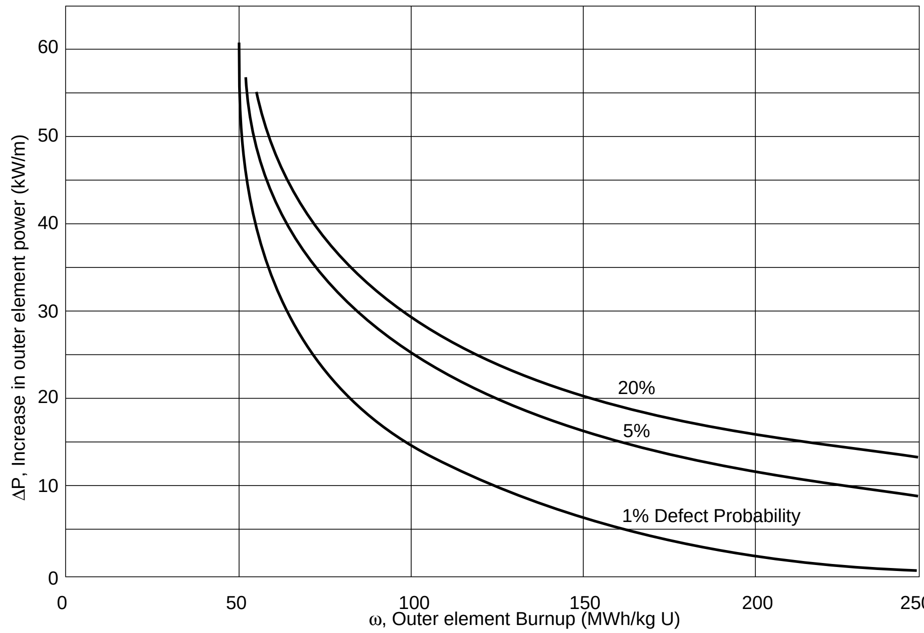
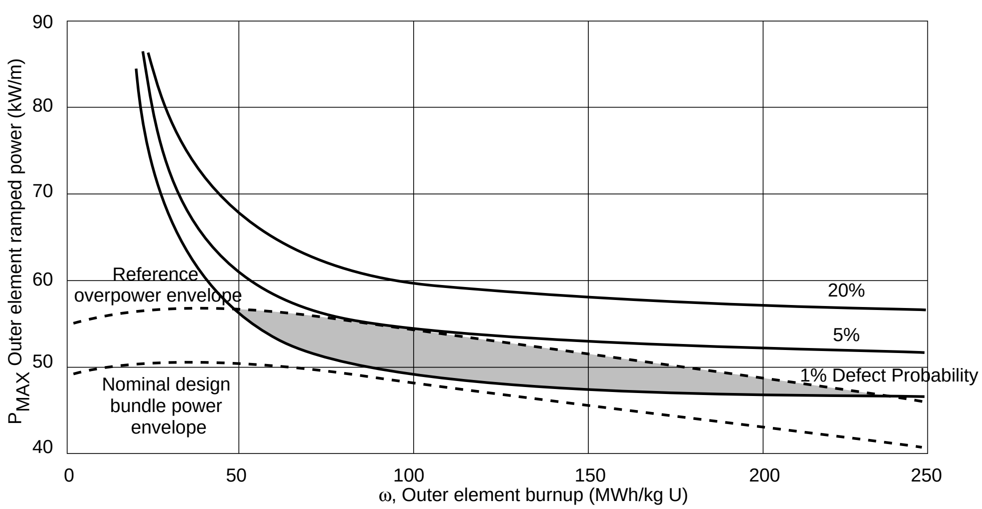


Figure 27

Ramped Power I(P_{MAX}) vs Burnup Defect Probability Curves for Graphite CANLUB Power Reactor Fuel



For the CANDU-6 reactors, which represent the latest designs and incorporate the latest DN and gaseous fission product monitoring systems, the cumulative defect rate was 0.07% (or less than 0.002% on a fuel element basis). The defect rate remains similarly low to the end of 1992.

6.2.1 Fuel Defect Causes

Fuel defects can be caused by any of the following effects:

- stress corrosion cracking (SCC) of the Zircalloy sheathing as a result of high stresses experienced during re-fuelling power ramps (early Pickering experience),
- fabrication flaws such as a small "piping" pathway in the end-cap weld or in the end cap itself, which allows ingress of primary, heat-transport fluid into the fuel element (early Bruce experience), and
- a few sheath penetrations caused by fretting by a small amount of debris in the heat-transport circuit.

(i) Stress-Corrosion Cracking (SCC) Defects

In the Pickering reactors, most of the identified fuel defects occurred in 1971 and 1972 (Unit 1 went critical in February 1971). The defects were usually small cracks in one or two of the 28 elements making up each fuel bundle (Figure 28). During the first year of operation, most defects resulted from power increases (boosts) caused by out-of-sequence movement of the reactor control rods. The sequence was modified and therefore control rod movement was eliminated as a defect cause. Iodine levels continued to increase, however, indicating that the reference eight-bundle re-fuelling scheme with non-Canlub fuel was causing additional fuel defects. Hot cell examinations at CRL showed the appearance of cracks from the Pickering fuel sheaths to be similar to that from laboratory tests on stressed sheathing in iodine (Figure 29). It was apparent that eight-bundle fuelling in power-peaked channels could have caused more defects, particularly in the central region of the reactor. Due to the increased fuelling in this region to remove fuel defects caused by control rod movement, the final power of the bundles after the re-fuelling shift were higher than normal. Changes in fuel management were made, and these virtually eliminated fuel defects in Pickering, but the need remained to understand why defects occurred, and to develop a fuel more tolerant to power increases. This would provide wider operating margins and increase operating flexibility. Fuel with the graphite inter-layer-designated Canlub - became the reference design. Since the adoption of Canlub, there have been no fuel defects which can be attributed to power increases.

Figure 28
Crack in Element of Pickering NGS-A Element Fuel Bundle Caused by Power Ramping

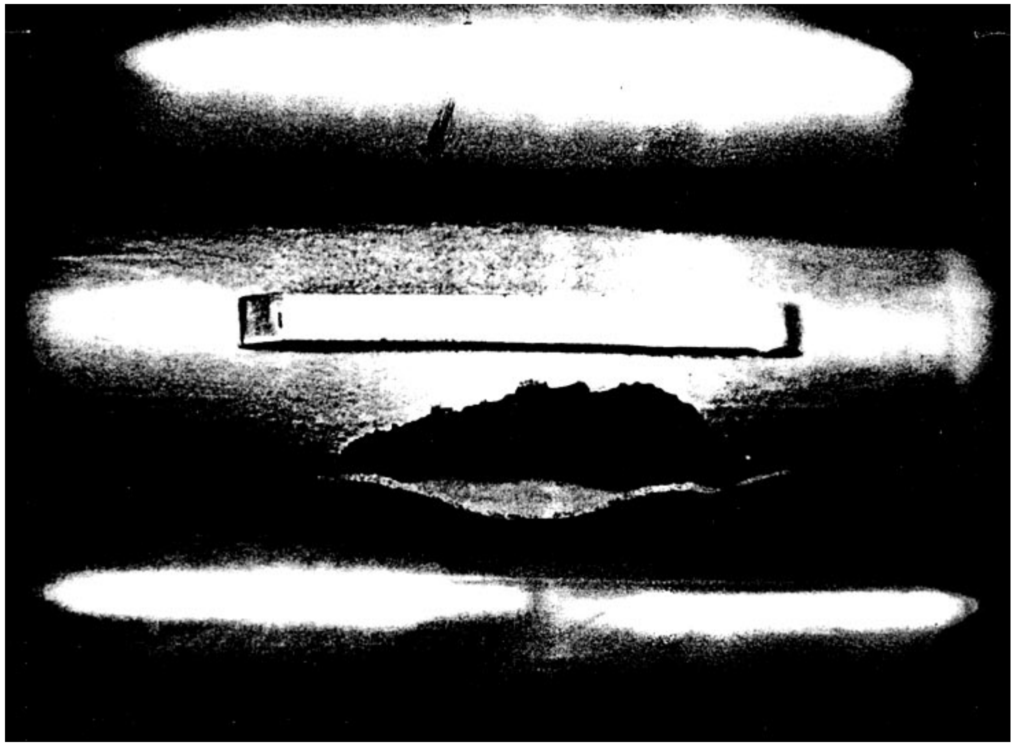
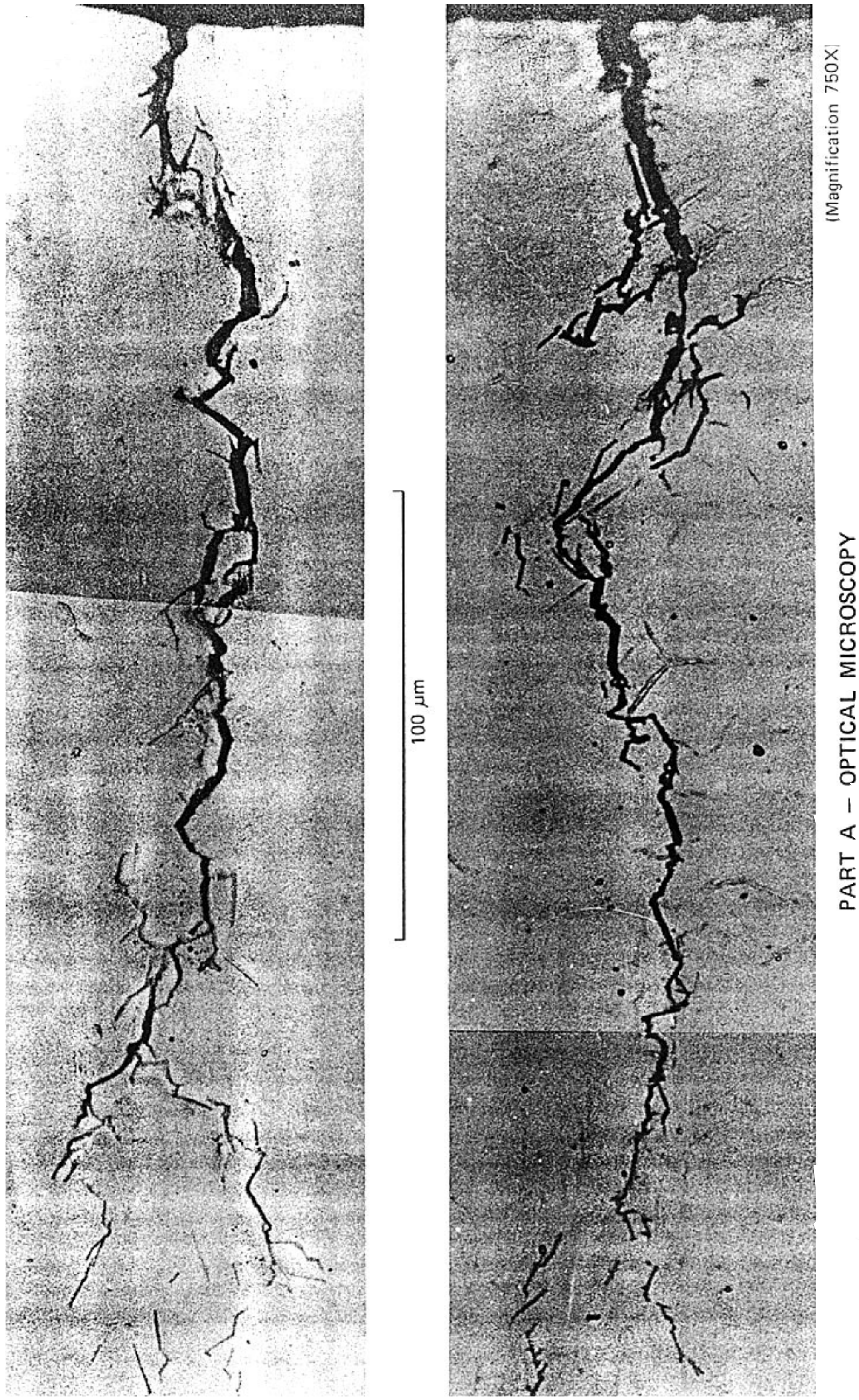


Figure 29

Left: *Iodine Induced Stress Corrosion Crack*

Right: *Crack Found in the Sheath of a Defective Fuel Element*



(ii) Fuel Fabrication Defects

During the first year of service of Units 1 and 2 at Bruce in 1977-78, defective fuel bundles were detected by the DN system. Through detailed visual examinations, neutron radiography, gas puncture analysis, leak testing and metallography, the primary defects were shown to be fabrication flaws. Of the defective elements which were examined, 80% had incomplete sheath to end-cap welds. Another 10% had stringers through the end-cap itself. As a result of being able to identify individual defective bundles, the researchers identified the specific defect causes and the fuel manufacturer has been able to identify and implement improved process controls. This has significantly reduced the number of defects from such fabrication flaws.

(iii) Mechanical Damage by Debris Fretting

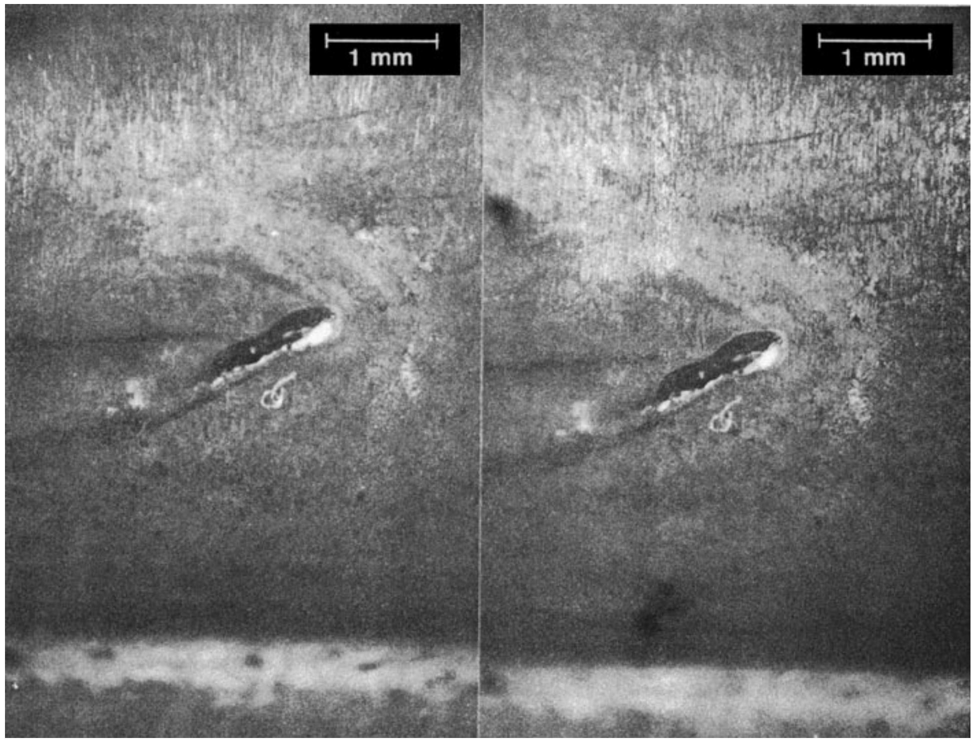
The remaining 10% of early defective fuel was caused by debris; typically this frets against the fuel sheathing and wears a hole through this protective barrier (Figure 30). Because of the size of the heat transport system in a power reactor, a small amount of debris is to be expected. Most of the damage to date has been from strainer wire breaking away from strainers used during commissioning.

6.3 High Burnup Experience

Fuel utilization experience continues to be excellent with current average discharge burnups of 195 MW.h/kg(U) at Pickering A and 196 MW.h/kg(U) at Bruce A. Fuel bundles routinely operate at burnups up to 280 MW.h/kg(U). It is interesting to note that these average burnups are significantly higher than the original design values of 162 and 188 respectively.

Occasionally fuel handling system abnormalities will prevent fuelling a particular channel for an extended period of time. In this event, the fuel burnups increase beyond the nominal range. More than 3000 bundles have now experienced average burnups in excess of 280 MW.h/kg(U), with a maximum value of approximately 630 MW.h/kg(U). This corresponds to an outer element burnup of 700 MW.h/kg(U). We have not identified any performance limits resulting from high burnup operation; all these bundles were discharged normally.

Figure 30
Hole Fretted Through Sheath by Debris



The power reactor experience shows that the fuel can operate in our current reactors to well over the design average burnup without problems. There have been no defects directly or indirectly attributed to high burnup. This is confirmed by a number of loop irradiations at CRL in the NRU reactor where fuel bundles have usually operated at higher power than the power reactor fuel. Again, there have been no loop fuel defects attributed to high burnup, and even when internal gas pressure were above coolant pressure there were no unacceptable dimensional changes.

6.4 Improving Fuel Usage

The use of natural uranium in the CANDU reactor has resulted in the best uranium utilization of any commercially viable system yet developed by a margin of at least 20 percent. The excellent performance of the more than five hundred thousand bundles (37-element design) irradiated in the Bruce and the CANDU-6 reactors in Canada, Argentina and Korea to the end of 1992 confirms the design, nevertheless, improvements are still possible. The flexibility of on-power fuelling and the compact design of the CANDU fuel bundle combine to allow a well-defined, statistically-significant demonstration of design modifications. In addition, proven modifications and fuelling strategies can be implemented very quickly in CANDU reactor operation. Two examples are:

- to demonstrate lower-cost fuel and/or higher-performance fuel, and
- increased burnup.

6.4.1 Advanced Fuel Designs

Bruce A has been undertaking demonstration irradiations of 1000 fuel bundles with a water-based graphite coating and 200 fuel bundles with siloxane coating on the inside of the fuel sheathing. Both coatings show potential for improved fuel performance and economics, and all bundles have operated or are operating satisfactorily. The current objective is to prove good performance at high burnups for advanced fuel designs - see also Section 8.

6.4.2 Increased Fuel Burnup

The nominal fuel discharge burnup is set by the reactor and system designs. Maintaining and improving this value is possible. Following startup this becomes the responsibility of the owner/operator. Significant improvements in the average discharge burnup have been achieved by Ontario Hydro, with corresponding fuel cost savings. Three methods have been followed at Pickering A and Bruce A nuclear generating stations for the past several years to achieve fuel cost savings of over 10% per annum. These are:

- increasing the purity of the moderator,
- increasing the uranium content in the bundle, and
- improving fuelling management strategy.

The normal discharge burnups for the Pickering and Bruce reactors as originally designed were 162 and 188 MW.h/kg(U). Average burnups which are being achieved today are 195 and 196 MW.h/kg(U) respectively. These improvements which have been achieved by three programs are illustrated diagrammatically in Figure 31 for Bruce A. A burnup improvement of 10% relative to that which was being reached in 1980 has been achieved. It is expected that similar schemes can give similar optimization of fuelling in the other CANDU power reactors.

6.5 Closing the Design Cycle

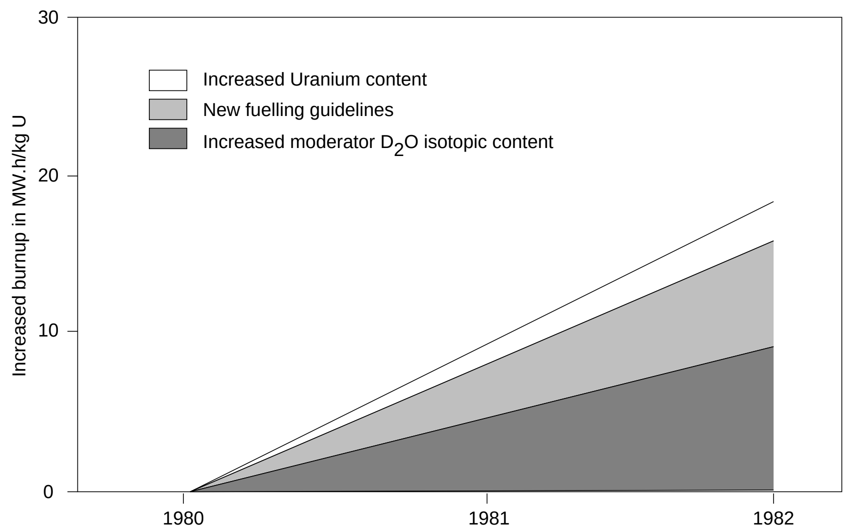
In Canada, the fuel designer's first task has been completed with the successful operation of mass-produced fuel in the Pickering, Bruce and CANDU-6 stations. The complete development program has included:

- running type tests on prototype bundles - to demonstrate that the fuel designs would satisfy the design requirements,
- preparing technical specifications which contain design knowledge and design requirements; these were made available to the fuel fabricators,
- developing and verifying computer codes which are used in the design and performance assessment of the power-reactor fuel designs,
- investigating the causes of fuel defects and introduced design changes,
- monitoring the operation and performance of the fuel in the power reactors, including that of demonstration irradiations, and used performance information to modify the design requirements and technical specifications, and
- improving operational techniques of fuel usage.

CANDU fuel is a mature proven product.

Figure 31

Bruce NGS-A Fuel Burnup Performance Relative to 1980



7. Fuel Cost Trends

7.1 Fuelling Costs

The early perception that economic nuclear power depends on low fuelling cost and hence strict neutron economy, set the stage for the development of efficient processes for the fabrication of CANDU fuel. For example, in 1964, when fabrication processes were first being automated for large-scale production, the first fuel charge for the Douglas Point reactor cost (Cdn) $73.20/kg(U). This resulted in a fuelling cost of m$1.1/kW.h. A few years earlier, it had been predicted that the "1 mill/kW.h barrier" would be broken, and this occurred shortly after when the cost of producing Pickering reactor fuel was m$0.95/kW.h in 1967. The cost remained below m$1/kW.h up to 1975, when the rapid increase in the cost of uranium began.

The estimated fuelling cost for the current CANDU-6 reactors is $2/kW.h based on the current cost of concentrates (spot market cost of (U.S.) below $10/lb U_3O_8) and current fabrication and processing costs, and on overseas shipping costs from Canada. It is above the 1967 cost due to inflation and an increase in the cost of concentrates. CANDU fuel cost trends are shown in Figure 32.

The m$2/kW.h figure appears, at first glance, to contradict Ontario Hydro's cost figures of m$4.52/kW.h and m$4.13/kW.h for their 1984 costs in the Pickering B and Bruce B reactors respectively. However, Ontario Hydro include M$0.6-1.3/kW.h for future decommissioning and irradiated fuel transportation, storage and disposal. They also have long-term contracts for the supply of concentrates.

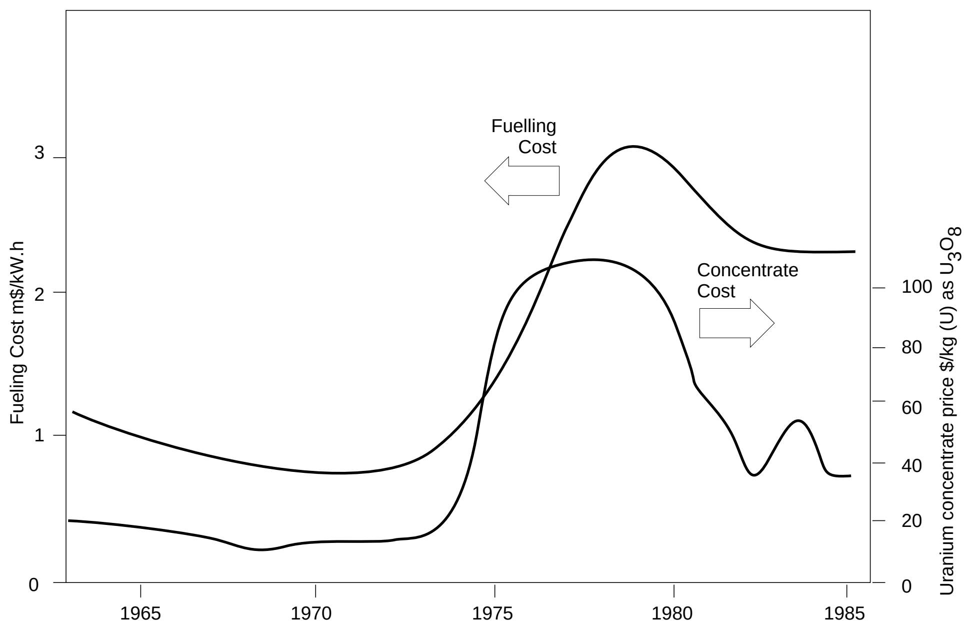
7.2 Fabrication Costs

The fabrication cost (in dollars of the year) of CANDU fuel has steadily declined and is now typically (Cdn) $85/kg(U). There are no conversion to UF_6 nor enrichment costs in the fabrication of HWR "natural" fuel; the cost of "conversion" of the U_3O_8 concentrates to sinterable grade UO_2 powder is included in the above fabrication cost.

Fuelling costs are affected by the lead times (time period prior to fuel being loaded into the reactor) for the procurement of natural uranium concentrates, for refining/conversion to UO_2, for fabrication itself and for shipping to the reactor site. With the comparatively short lead times required for CANDU fuel (due to the missing enrichment step and the very short fabrication step), the indirect fuel costs are a small fraction of the direct costs.

Figure 32

CANDU Fuelling Cost Trends



Traditionally the fuel is paid for upon delivery, and for a single reactor, such as may be operated by a small utility, the fuel would be conveniently delivered in approximately equal batches three or four times a year.

With a lead time of six months (the batch is delivered six months before it is used for re-fuelling the reactor), an inventory of new fuel would always be available at the reactor site. This would be a minimum of six month's inventory (from previous batches), and it would be increased to 9 months worth of re-fuelling following delivery of the new batch. Such an inventory would provide a "cushion" for upsets or delays in fuel deliveries.

Today, a quarter core batch of fuel can be made in one month by a fabricator with the technology. Processing of the yellowcake to UO_2 powder is done during the previous three months. Hence the uranium concentrates are supplied to the refinery 10 to 12 months before the completed fuel is delivered to the reactor site. For a fabricator and utility just beginning to make and procure CANDU fuel, longer lead times would be expected.

8. Future Developments

There are still opportunities for improvements in CANDU fuel, and these are being explored. The capability of on-power re-fuelling of the CANDU reactor allows simple and gradual introduction of new fuel designs, for example, bundles with thin-wall sheathing (to improve burnups), or bundles with graded element sizes. Both these features will also maintain good UO_2/Zircalloy ratios.

It is possible to obtain significant reductions in fuelling costs by taking advantage of the reducing costs of enrichment that are occurring on world markets today. The AVLIS (atomic vapour laser isotope separation) process which may be available in the future, promises low separation costs. Recovered uranium (RU) from irradiated LWR fuel promises to be equally inexpensive. How can advantage be taken of this in the CANDU reactor?

The versatile CANDU reactor can accept RU or slightly enriched uranium (SEU) fuel with no changes being needed in reactor hardware. The initial fuel cost, which for SEU fuel might be higher than that for natural uranium fuel, would be more than offset by the higher achievable burnups of the SEU fuel. Other costs, such as for irradiated fuel storage and disposal, would also be reduced due to the smaller quantities of fuel that are needed. Also there would be a decrease in fuelling machine usage due to the reduced fuelling rate.

Typically a reactor would use natural uranium fuel at initial startup. Following this a transition to the SEU cycle could be carried out. An enrichment of 1.2 wt % ^{235}U will produce a fuel burnup of 530 MW.h/kg (U). An enrichment of 1.5 wt.% ^{235}U will produce a burnup of 650 MW.h/kg (U). This is a four-fold increase in the exit fuel burnup of the natural uranium cycle.

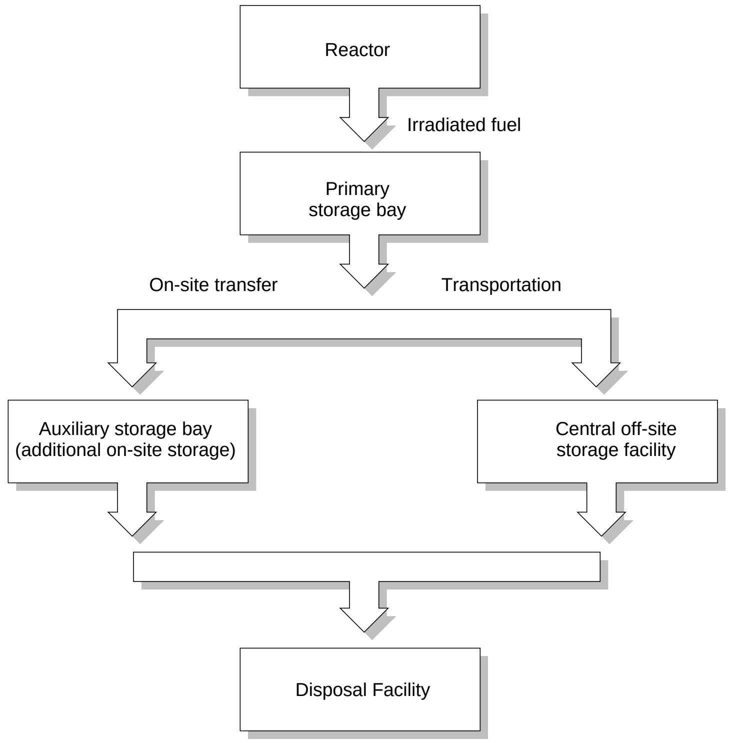
Studies have determined the fuelling schemes which are most suitable to use with the RU and SEU cycles. Several schemes are possible, in which the transition from the natural to SEU cycle, for example, is predicted to take 900 Full Power Days (FPD) of reactor operation. Fuel and channel powers are within the limits specified for the natural uranium cycle.

The SEU fuel cycle offers some flexibility in the design of the fuel itself (for example, the optimization of individual element heat fluxes within the bundle) and in the reactor, which can lead to reduced capital costs for future reactors by as much as 10%. This results in a reduction to the capital cost component of the total unit energy cost. Current research programs at the Chalk River Laboratories and the development programs at AECL-CANDU in Mississauga are geared towards optimizing these and other cycles for CANDU. Many irradiations of SEU fuel have been completed, and many of the results were used for the current fuels in use today. Similar programs are continuing, and are being supplemented by development of fabrication processes and manufacturing technology. By focusing on the SEU cycle, many of the techniques for the other advanced fuel cycles, such as the recycle enriched uranium or the uranium/plutonium or tandem cycles, are being developed. If economic conditions should favour these cycles, advantage can be taken to move into development of these cycles quickly.

9. Irradiated Fuel Storage and Disposal

In closing the fuel cycle, it is necessary to store the irradiated fuel, and ultimately to dispose of it in a safe and responsible manner [19]. The sequences for the movement of irradiated fuel from the reactor to storage, and to disposal, are shown in Figure 33. The primary storage bay, in the reactor building, is used to first store the fuel. The bay is built to store up to ten years-worth of fuel from full-power operation plus an allowance for a full core-load. When needed, an auxiliary bay can be built, also at the reactor site. An alternative is to ship the fuel, away from the reactor site, to a central storage facility. Final disposal of the fuel, and any fuel wastes, will be deep underground in a stable geologic formation either at the central storage facility, or at a dedicated disposal facility.

Figure 33
Irradiated Fuel Management Sequence



9.1 Storage

9.1.1 Primary Storage Bay

When first discharged from the reactor, the fuel is stored under water in a primary storage bay. This is a large water pool, Figure 34, in which fuel bundles are stacked in racks or modules. The bundles can be stored at a high density, to efficiently use the storage volume, because there is no criticality hazard with the fuel.

9.1.2 Auxiliary Storage Bay

Very similar to the primary bay, an auxiliary bay can be built later in the life of the reactor plant. The bay is usually built adjacent to the reactor buildings, in which case the fuel is transferred through an underground tunnel. For example, auxiliary bays have been built by Ontario Hydro for their Pickering and Bruce nuclear generating stations (Figure 35).

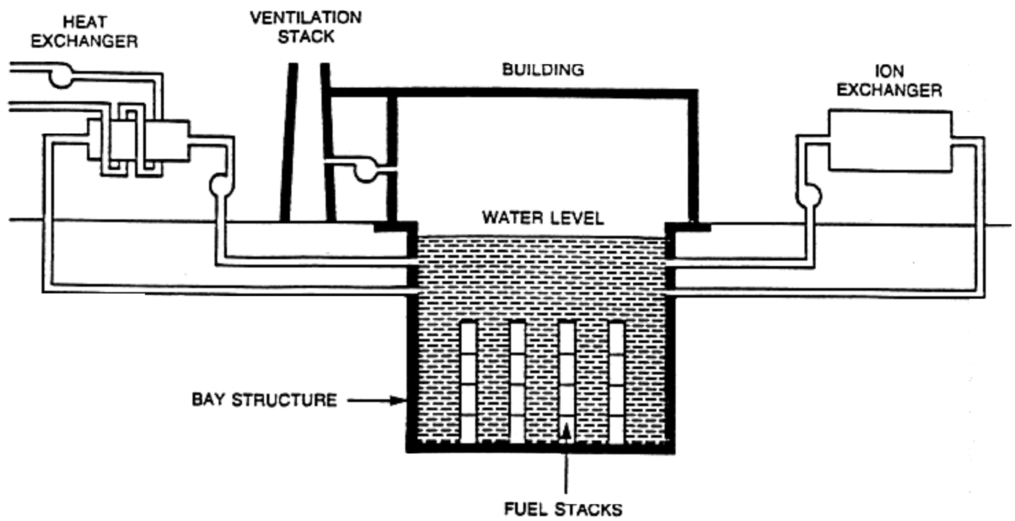
9.1.3 Central Off-site Storage

A second concept for the long-term storage of irradiated fuel is to ship the fuel away from the reactor site. Although this concept has not been adopted in Canada, AECL and Ontario Hydro continue to conduct joint programs to investigate different storage methods.

Under Water Storage.
Major objectives of the test programs in Canada are to determine how long irradiated fuel can be stored in water pools, and more specifically to study the stability of the Zircalloy-clad UO_2 at 30-50°C and the corrosion of the Zircalloy clad itself.

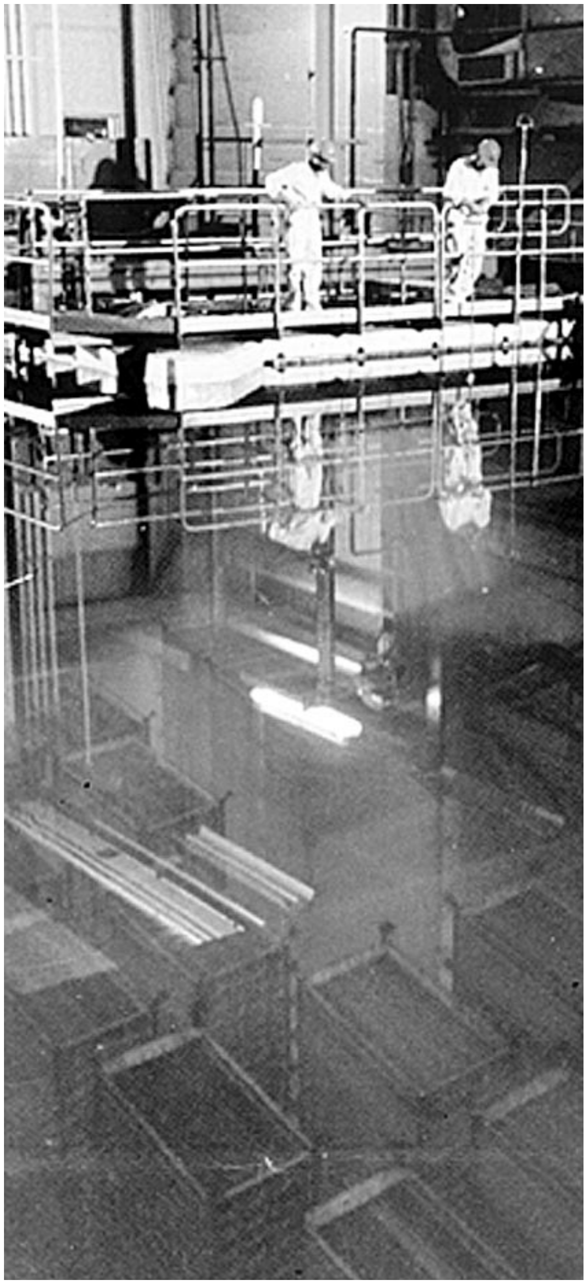
Eighteen years of experience, data from detailed examinations and careful projections of Zircalloy deterioration show that the fuel can be stored for more than 50 years under water.

Figure 34
Irradiated Fuel Storage Bay



Dry Storage in Concrete Canisters.
The long-term behaviour of irradiated CANDU fuel is also being thoroughly investigated for storage in concrete canisters in dry air. The fuel is being tested now at 150°C in dry and in moist air. First inspections show that dry storage in canisters is a viable and acceptable concept.

Figure 35
Auxiliary Fuel Storage Bay



9.2 Disposal

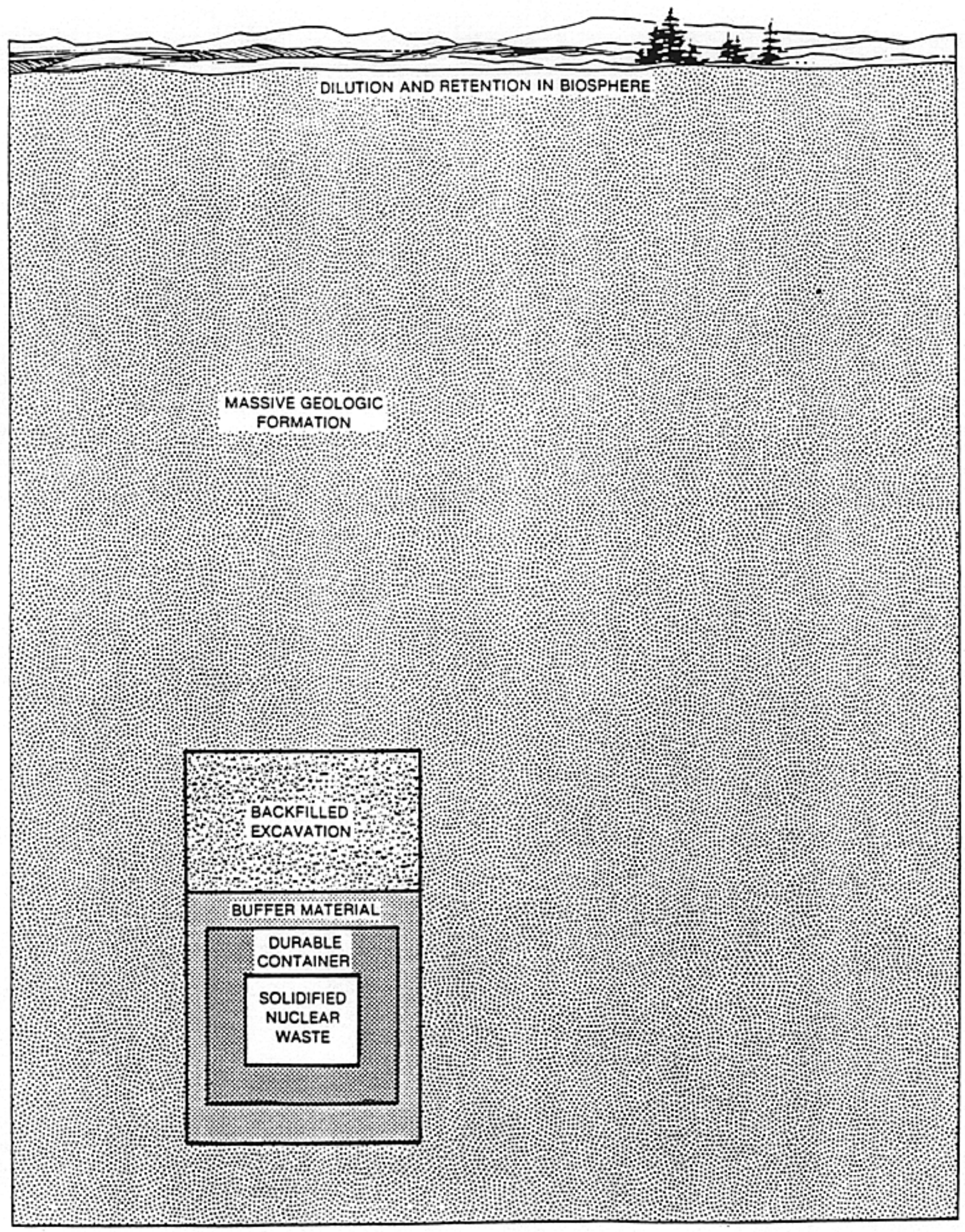
Although irradiated CANDU fuel is likely to be stored for many decades,the techniques for the safe, ultimate disposal of nuclear fuel waste are being developed now. The overall objective of the Canadian nuclear fuel waste management program is to ensure that there will be no significant effect on man or the environment at any time. To achieve this objective AECL proposes to dispose of suitably immobilized material deep underground in a stable geologic formation. The consensus of world scientific and engineering opinion considers this the most suitable way of ensuring isolation of these materials from man and the environment.

The Canadian nuclear fuel waste management program is concentrating on disposal in the stable, hard-rock formations of the Canadian Shield which underlies much of the country. Work on other formations, such as bedded salt, clays or shales, is limited to the identification of potentially suitable deposits. However, by contributing to and participating in the world-wide exchange of information on waste management programs, in international studies such as the Seabed Working Group, and by keeping abreast of work in other countries, Canada maintains the option of pursuing disposal in other geologic formations. Canada has bilateral exchange agreements with the United States of America, the United Kingdom, the Commission of the European Communities, and Sweden. These agreements not only provide for the exchange of data and other information, they also encourage co-operative programs in areas of mutual interest.

9.2.1 Geologic Disposal

The basic concept of geologic disposal uses the principle of defense-in-depth, as shown in Figure 36. It is widely accepted that once the radioactive waste has been emplaced deep underground, the material could only escape to the biosphere, and so to man, if transported by water. Isolation of the radioactive materials can therefore be ensured by a number of natural and engineered barriers which act together to prevent their escape. In the first place, the waste is made as insoluble in groundwater as possible and clad in a corrosion resistant container; this step is called immobilization. Secondly, by surrounding the immobilized waste with buffer and backfill materials to retard the movement of any radioactive material that may be dissolved, along with the natural adsorption characteristics of the rock formation, a further degree of safety is added. Thirdly, burial of the immobilized waste some 500 to 1000 m deep in a stable rock formation provides a natural barrier; further, choosing a rock formation with a very low groundwater flow adds to the effectiveness of that barrier. Finally, should any radioactive material escape to the biosphere, dilution in the surface environment will reduce the effect on man.

Figure 36
Features Protecting Man from Nuclear Waste



9.2.2 Research Programs

The initial research phase of the program is aimed at assessing the basic safety and environmental features of this concept. The emphasis is being placed on disposal in crystalline hard-rock formations known as plutons. These cover a wide spectrum of rock types and fracture patterns. Laboratory and field studies are being performed to determine their properties and hydro-geological characteristics. The data are being used to assess the effectiveness of the various natural and engineered barriers.

To ensure that expertise is available in all the scientific and engineering disciplines essential to the program, AECL has actively encouraged participation of Canada's scientific community. Private industry and consultants are also extensively involved. The research and development program is now well-established and significant progress is being made. As it is not practicable to describe in detail the work being done, selected highlights are presented.

Canada has not yet taken a decision on the future recovery and recycle of plutonium. Therefore techniques are being developed for the immobilization of both irradiated fuel and reprocessing wastes. The characteristics of used fuel and reprocessing wastes dictate that a different approach to immobilization must be taken in each case.

Work on used fuel immobilization is focused on the design of high-integrity containers with a life of 300 to 500 years. If this can be achieved, isolation of the vast majority of radioactive fission products for the duration of their hazardous lives would be guaranteed. The used fuel itself is very resistant to dissolution under certain conditions, and experiments to date indicate that it may be a viable waste form.

Work also is being done on the incorporation of the fission products, actinides, and other wastes into a low-solubility matrix. This would deal with the waste associated with the reprocessing of used fuel from all potential CANDU fuel cycles. AECL pioneered this work thirty years ago. Separated wastes have been incorporated into glass matrices in several countries. Conventional borosilicate glasses have been used but attention is being given to other materials, including aluminosilicate glasses and crystalline solids.

The dissolution rate of the potential host materials and the UO_2 fuel are comparable to those of granite, under the conditions of the tests, and give encouragement that waste forms will have durability comparable to that of granite in the underground environment.

Geotechnical research is being performed at three established research areas, at Forsberg Lake, near Atikokan in Ontario, and at AECL's laboratories, CRL and WL.

Information obtained from the boreholes at these sites indicates that, although many fractures exist in granitic plutons, there are significant regions of rock, at depth, with few "open" fractures and very low hydraulic conductivities.

9.2.3 Underground Research Laboratory

As the program has developed, it has become evident that to complement information obtained by laboratory and borehole testing, it is essential to conduct geotechnical experiments on a scale and in an environment which can only be achieved underground. An underground research laboratory (URL), at a depth of about 300 metres, in the Lac du Bonnet batholith, is now being built. Interest in such a laboratory is being shown by several other countries. The project will provide information on the effects of excavation on the rock mass and the hydrogeological behaviour in addition to a wide range of geological, hydrogeological, and geochemical information (Figure 37).

9.3 Closing The CANDU Fuel Cycle

Work is well-advanced towards the development of the techniques needed to safely dispose of the nuclear fuel waste, whether this be in the form of immobilized used fuel or immobilized wastes from the recovery and recycle of fissile material. We are evaluating the concept of deep underground disposal in the hard-rock formations of the Canadian Shield. Preliminary assessments indicate that nuclear fuel wastes can be effectively isolated from man and the environment. Much of the technology generated by this waste management program will be relevant to disposal in other types of geologic formation and to other countries. It will also be directly useful in the disposal of other toxic materials.

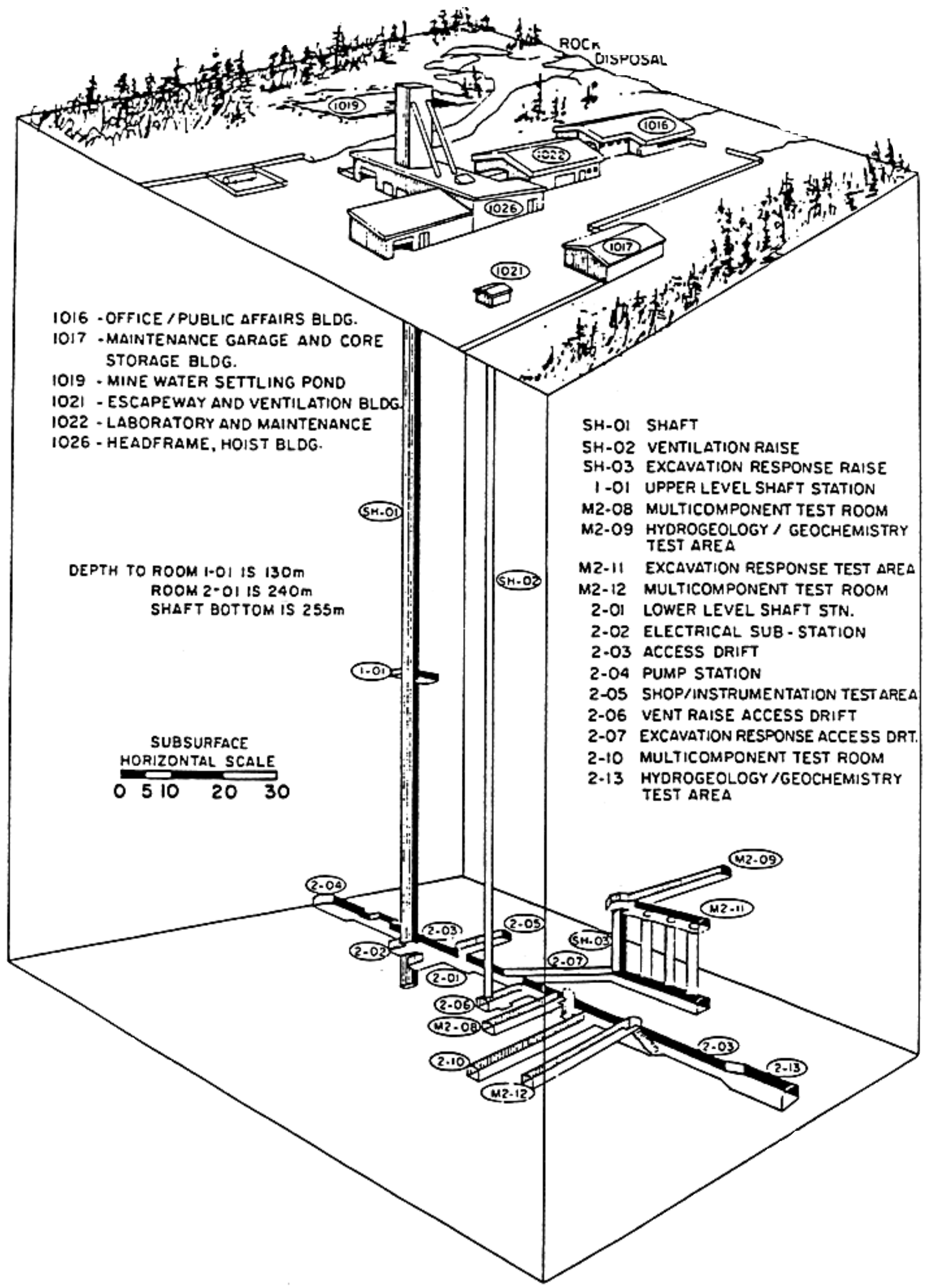
10. Closure

This report has reviewed the 35-year evolution of the current CANDU fuel design, and the extensive irradiation testing program that has been the basis for the design. The out-reactor testing programs have been equally vigorous and have proceeded in parallel with the evolution of manufacturing processes.

The fuel interacts with a number of reactor systems, and these have been described. The development of technical specifications for fuel performance and fuel design provides the link between fundamental research data and the requirements of the power reactors. The designers' choices have been verified and proven using type tests and established models. Continuing analysis of the data from the operating commercial reactors confirms the design.

The performance of the CANDU-PHWR type of nuclear generating station, which has exceeded the performance of any other type of nuclear station in the world, depends in part on the in-depth development program. This has involved, and continues to involve, all the scientific and engineering disciplines; the establishment of objectives, measurement of results in the extensive test programs, identification and solving of problems and continuous analysis of operating experience from the commercial power reactors support this performance.

Figure 37
Underground Research Laboratory at WNRE, Manitoba



11. Acknowledgements

This lesson is based on an IAEA Nuclear Power Course lecture on Technology of Water Cooled Power Reactors, Lecture CA1.7.7, "CANDU Fuel Design, Manufacture, and Performance", by I.E. Oldaker and M. Gacesa, 1984 October/November. H. Underwood working for Canadian Nuclear Services provided operating information and data.

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Fuel Channels

Training Objectives

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

1. Identify the major components of a fuel channel assembly.
2. Trace the design evolution of fuel channels and recognize the differences between the two basic types of fuel channels.
3. Identify the design requirements of fuel channels by way of:
   - 3a) Function
   - 3b) Performance
   - 3c) Materials
   - 3d) Safety codes and standards
   - 3e) Installations
4. State the characteristics of fuel channel components with respect to:
   - 4a) Pressure tube design and fabrication
   - 4b) End fitting design and fabrication
   - 4c) Calandria tube design and fabrication
   - 4d) Feeder pipe connections
   - 4e) Channel closures and shield plugs
5. Quantify the operating parameters of fuel channels and variance between stations.
6. Trace the historical performance of pressure tubes and the impact on design and assessment, with respect to:
   - 6a) Delayed hydride cracking (DHC)
   - 6b) Leaking pressure tube experience
   - 6c) Pressure tube rupture experience
7. State the basic need for and requirements to establish fitness for service of pressure tubes throughout the reactor life.

Table of Contents

1. Introduction

An important feature of the CANDU (CANadian Deuterium Uranium) PHWR is the use of heavy water both as the moderator and as the heat transport coolant, which allows a critical chain reaction to be sustained with natural uranium fuel. This fuel can be sufficiently separated from the moderator so removal of the heat generated by the fuel can occur in a high temperature primary coolant that is separate from the moderator. Development of zirconium alloys which have a sufficiently low neutron capture cross-section that they do not impose an excessive neutron penalty when placed between the fuel and the moderator allows the use of a pressure tube reactor design to separate the primary coolant from the moderator.

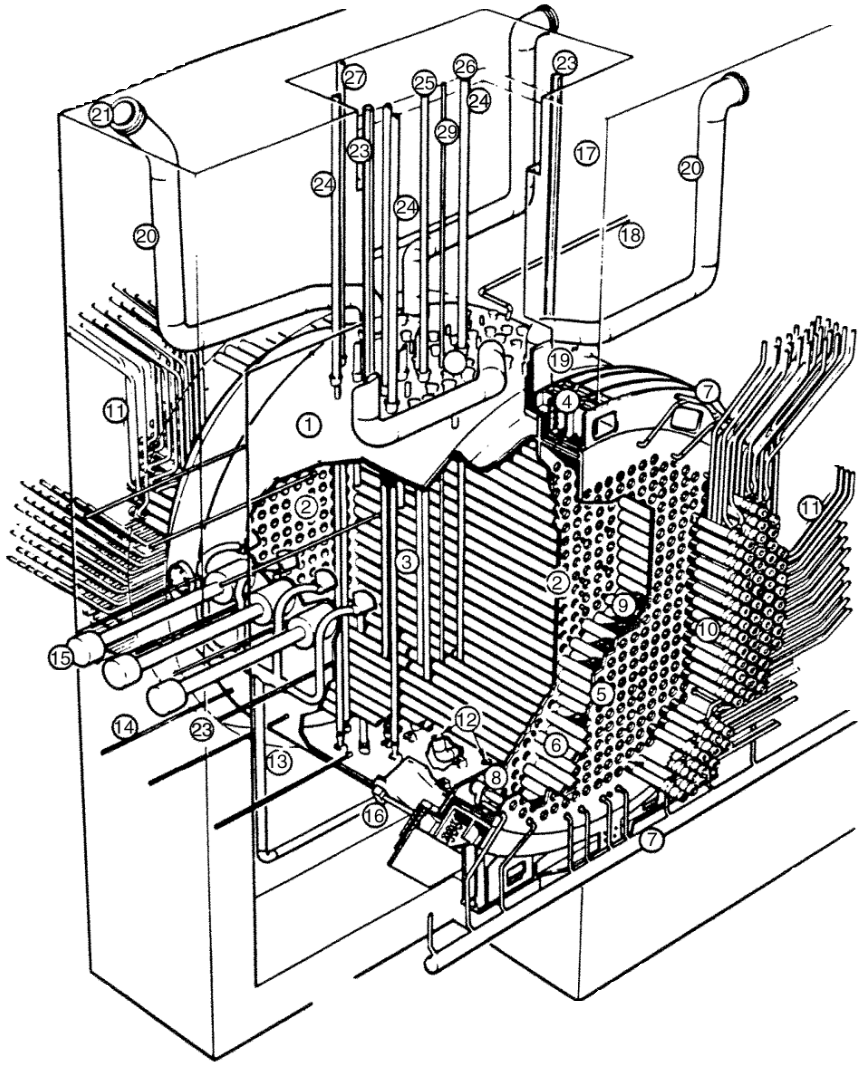
As is illustrated in Figure 1, the core of a CANDU reactor consists of a large, thin wall, low pressure cylindrical tank (the calandria) whose axis is horizontal. Its vertical ends (the end shields) are joined by a few hundred horizontal calandria tubes having a lattice pitch of 28.5 cm. A pressure tube is located inside each calandria tube and is separated from it by spacers in the annular gap, as is illustrated in Figure 2.

Since the fuel bundles reside inside the pressure tubes, heat generation in a CANDU reactor takes place in a few hundred high pressure fuel channels, rather than in a single large pressure vessel as occurs in a PWR, whose core is contained within a large thick walled reactor pressure vessel. Inside this vessel, there is no distinction between the moderator and primary coolant. However, in the core of a CANDU reactor, the cool, low pressure, heavy water moderator contained in the calandria vessel is separated from the hot pressurized, heavy water primary coolant contained in the fuel channels.

A schematic flow diagram for a typical CANDU heat transport system and moderator system is shown in Figure 3.

Use of a low temperature moderator provides a large heat sink capable of absorbing any energy that might be released during postulated accidents. In addition, separation of the primary coolant and the moderator permits their chemistries to be independently controlled and separately optimized. It also permits the use of dissolved neutron absorbers in the moderator to control or shutdown the reactor.

Figure 1
CANDU Reactor Assembly



1	Calandria	16	Earthquake Restraint
2	Calandria - Side Tubesheet	17	Calandria Vault Wall
3	Calandria Tubes	18	Moderator Expansion to Heat Tank
4	Embedment Ring	19	Curtain Shielding Slabs
5	Fueling Machine- Side Tubesheet	20	Pressure Relief Pipes
6	End Shield Lattice Tubes	21	Rupture Disc
7	End Shield Cooling Pipes	22	Reactivity Control Units Nozzles
8	Inlet-Outlet Strainer	23	Viewing Port
9	Steel Ball Shielding	24	Shutoff Unit
10	End Fittings	25	Adjuster Unit
11	Feeder Pipes	26	Control Absorber Unit
12	Moderator Outlet	27	Zone Control Unit
13	Moderator Inlet	28	Vertical Flux Detector Unit
14	Horizontal Flux Detector Unit	29	Liquid Injection Shutdown Nozzle
15	Ion Chamber	30	Ball Filling Pipe

Figure 2

Illustration of the Portion of a CANDU Fuel Channel that Passes Through the Reactor Core

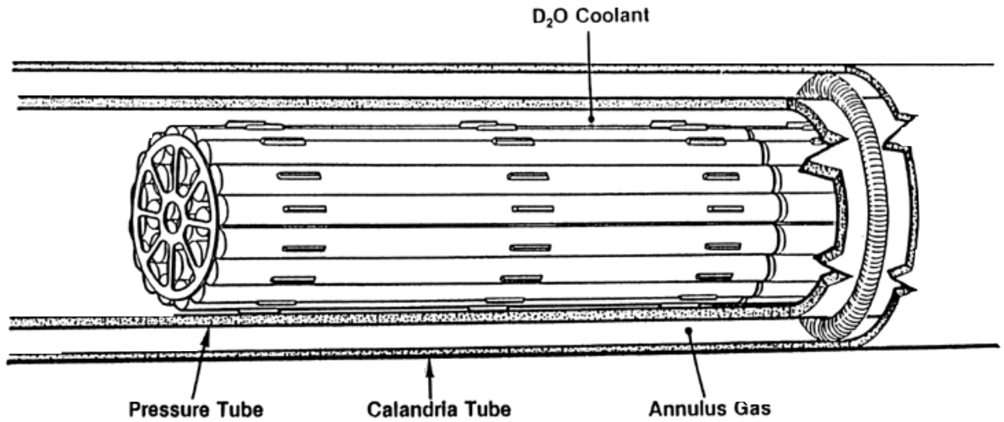
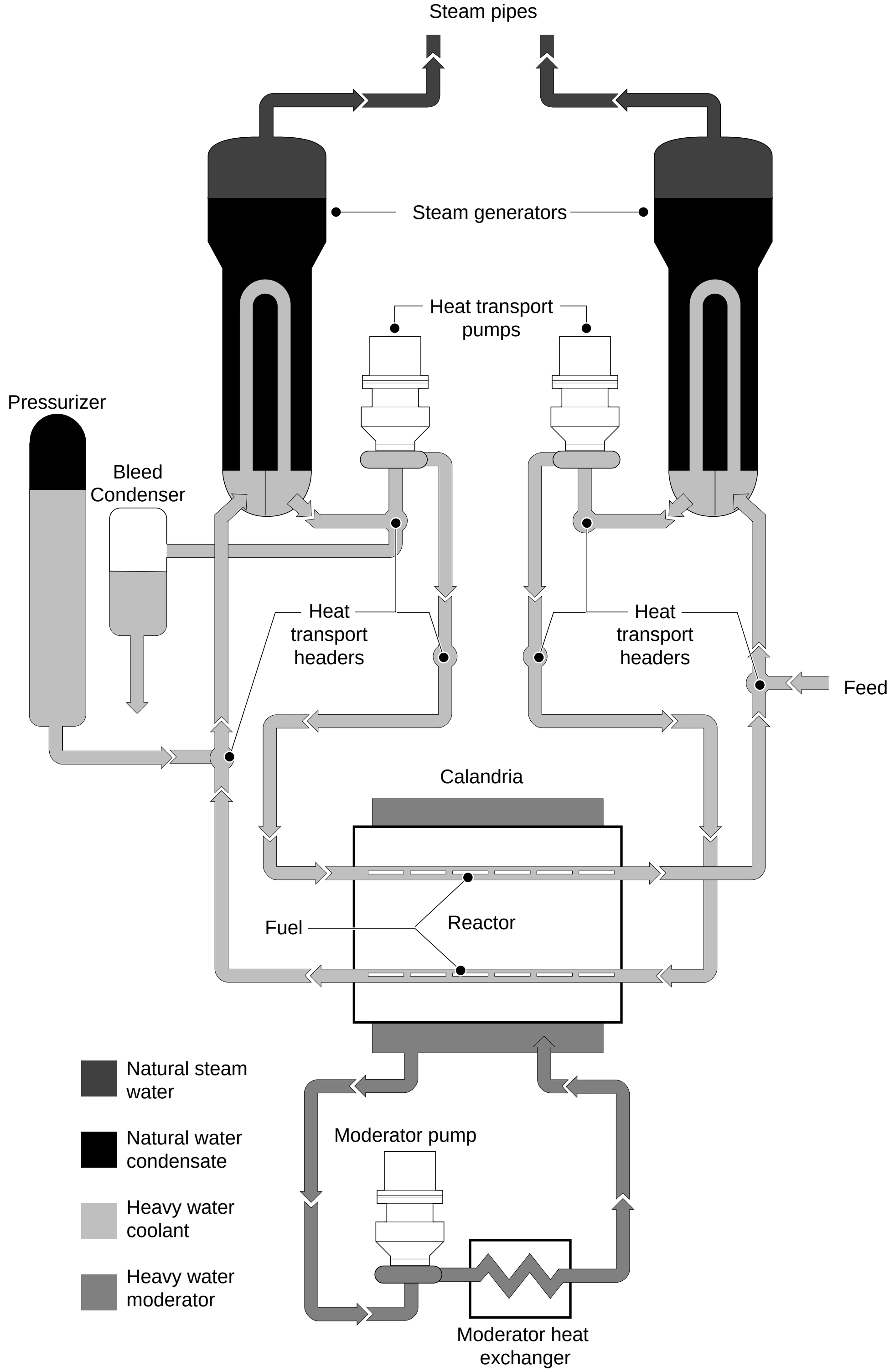


Figure 3
CANDU Reactor Simplified Flow Diagram



1.1 On Power Fuelling

Use of natural uranium as fuel, and heavy water as the moderator, requires that CANDU fuel must be removed relatively frequently and for efficient fuel utilization the fuel needs to be shuffled within the fuel channel. This necessitated the development of on-power fuelling. The use of fuel channels in the CANDU reactor core allows refuelling of the reactor while it is operating at full power. On-power refuelling, by fuelling machines which connect to the ends of a fuel channel, contributes to the high net capacity factor of CANDU plants by eliminating the need for reactor shutdowns to change fuel.

1.2 Fuel Channel Replacement

Because fuel channels pass through the centre of the CANDU reactor core, the fuel channel components experience significant radiation induced changes in mechanical properties and dimensions. Although the knowledge available to predict these changes was limited for the early designs, the expectation was always that the life of channels was finite, since fuel channels are the CANDU reactor component seeing the most severe duty. Thus the channel design has always allowed for relatively easy replacement. Replaceability of single fuel channels allows:

- surveillance removal of channels, and
- correction of any individual channel problem that may occur.

The ability to remove fuel channel components, and then test them in hot cells, has provided valuable feedback for development and predictive purposes.

In addition, complete retubing of units permits life extension of a reactor core, if its fuel channels reach their end of life before most other reactor components. This is possible since fuel channel components have more severe operating conditions than any other CANDU reactor components. They operate at high pressure and temperature while passing through the centre of the reactor core, so they also experience a high level of irradiation damage.

1.3 General Description of a Fuel Channel

Fuel channels are one of the major distinguishing features of a CANDU reactor, and their reliability is crucial to the performance of the reactor. The following gives an outline of the key characteristics for CANDU fuel channels.

Fuel channel assemblies are part of the heat transport system (HTS) in a CANDU reactor. The primary purpose of each channel is to locate and support the fuel bundles in the reactor core so that they can generate heat, which will be taken to the HTS steam generators by the coolant flow.

Figure 4 is a schematic illustration showing the key features for one of the few hundred channels in a typical CANDU reactor core. Each fuel channel consists of four major components: the pressure tube, the calandria tube, the annulus spacers and the end fittings.

The most important component of each fuel channel assembly is its 6 m long, 10 cm diameter zirconium alloy pressure tube that has a wall thickness of about 4 mm. As these tubes contain the high pressure and high temperature primary coolant, they must be adequately resistant to corrosion/erosion as well as to creep/growth under neutron bombardment.

Each pressure tube is located inside a calandria tube with the gas filled annulus between these two tubes insulating the high temperature primary coolant inside the pressure tube from the low temperature moderator outside the calandria tube that surrounds each pressure tube.

Four annulus spacers (spaced about a meter apart) keep each pressure tube separated from the calandria tube which surrounds it, while also allowing the calandria tube to provide sag support for the pressure tube. The annular space around the fuel channel, which is filled with dry CO_2 gas, is connected to an annulus gas system that incorporates moisture detecting instrumentation to warn of leaks from either the pressure tube or the calandria tube.

The ends of each pressure tube are rolled into stainless steel end fittings to form a pressure tight, high strength joint. These end fittings provide a flow path for primary coolant between the fuel channel pressure tube and the rest of the primary heat transport system by having a feeder pipe attached to each end fitting, as is illustrated in Figure 5.

Each fuel channel end fitting passes through a lattice tube in one of the reactor end shields. Contacting surfaces of the end fitting and the lattice tube which supports it are made of hardened tool steel to form sleeve bearings that allow an axial sliding motion of the end fitting. The ends of the fuel channels are sealed by removable channel closure plugs which provide access for the fuelling machines to remove irradiated fuel from the channel and insert fresh fuel into it, while the reactor is operating at full power.

Figure 4

Schematic Illustration of a Typical CANDU Fuel Channel

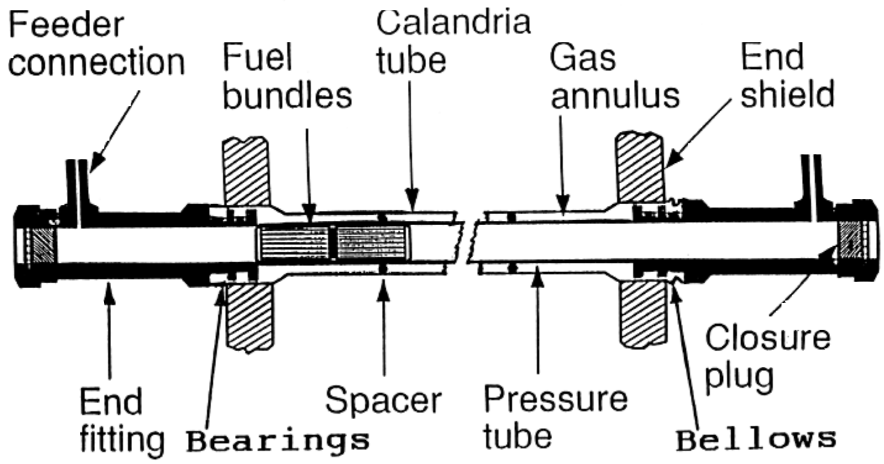
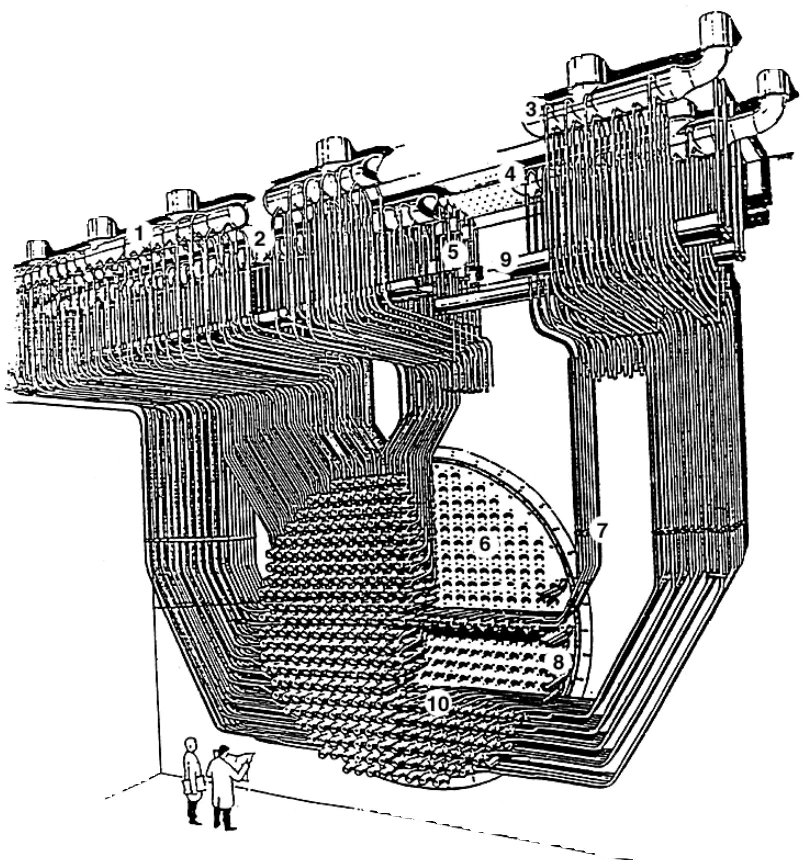


Figure 5

Schematic Illustration of the Feeder Pipe Arrangement for a Typical CANDU Reactor Core



1	*Reactor Outlet Header*	*2*	*Reactor Inlet Header*
3	*Reactor Outlet Header*	*4*	*Reactor Inlet Header*
5	*Feeder Tuber Upper Supports*	*6*	*Calandria End Shield Face*
7	*Tube Spacers*	*8*	*Support Brackets*
9	*Walkway*	*10*	*End Fittings*

2. Fuel Channel Design Evolution

CANDU reactor design is based on experience derived from preceding CANDU reactors and virtually every design feature of the latest CANDU reactor is identical to, or is an evolutionary improvement of, an earlier proven design.

NPD (20 MW), which went into service in 1962, was the first Canadian reactor that had fuel channels. The next reactor of this type was Douglas Point (200 MW) that went into service in 1967. The first commercial (≥ 500 MW) CANDU reactors were the four Pickering A units that went into service between 1971 and 1973. The four Bruce A units then went into service between 1977 and 1979.

This evolution of the fuel channel design is illustrated by Figure 6, which shows the essential features of the NPD, Douglas Point, Pickering, and Bruce designs.

Table 2 and Figure 7 indicate that there are two basic types of CANDU fuel channels. This is because there are two types of CANDU fuel handling systems. Douglas Point, Pickering and CANDU 6 reactors use the AECL fuel handling system for which there are 12 fuel bundles in each channel (only ten for Douglas Point) with the fuel being supported against the end of a shield plug during operation. New fuel is inserted from the inlet end of the channel. In contrast, Bruce and Darlington use the CGE fuel handling system for which there are 13 fuel bundles in each channel with the fuel supported against a latch in the outlet EF during operation. New fuel is inserted from the outlet end of the channel. This fuel handling system requires a larger diameter end fitting because of the use of a fuel carrier for fuelling operations, but minimizes fuel bundle movements in the pressure tube, as indicated in Table 2.

Table 2

Fuel Channel Characteristics

	Pickering and CANDU 6 Type	Bruce and Darlington Type
Fuel Separation	In Fueling Machine	In Fuel Carrier
No. of Fuel Bundles per Channel	12	13
Fuel Bundle Passes (4 bundle shifts)		
- inlet end	4	4
- outlet end	16	4
Closure Plug Seal	Expanding jaws	Rotary motion

N.P.D. had tight fitting "garter" springs while all of the units built later in Canada had loose fitting "garter" springs, unfortunately the loose fitting design allowed displacement from the design position to occur during commissioning and early operation periods before any appreciable sag of the pressure tube had occurred to pinch and trap the "garter" spring against the calandria tube. The fact that the garter springs would not remain in their design location went unnoticed until 1983 with the failure and subsequent investigation of a pressure tube in Pickering (to be discussed later). The consequence of "garter" springs out of design location proved to be greatly detrimental and led to the complete retubing of Pickering units 1 and 2 after only 12 years of service. During the period up to 1983, 17 other Candu units were committed to the loose fitting garter springs. The pressure tube failure in 1983 launched probably the largest research program Ontario Hydro had ever undertaken, because they were suddenly faced with the probability of more unpredictable major failures and the huge economic impact of premature retubing of all reactors at less than half their predicted design life of 30 years.

Also to be noted is that designers allowed for 30 years of axial pressure tube creep in N.P.D. which proved to adequate based on in-service results, however Pickering A and Bruce A show as only 13 years allowance. The reduced allowance to only 13 years at Pickering and Bruce was due to the unexpected growth rate in larger power reactors operating at much higher neutron flux levels than the test reactors from which creep data was originally obtained by the designer.

While the premature retubing of Pickering A was driven by the failure event above, retubing or major reconfiguration would have been necessary at 13 years service anyway.

Updated creep data allowed the designers to build in sufficient creep allowance in all units subsequent to Pickering A and Bruce A.

Table 2

Basic Type	NPD	Douglas Point	KANUPP	Pickering A						Bruce A					
CANDU Units	NPD	Douglas Point RAPP-1, 2	KANUPP	Pick. 1, 2 (Original)	Pick. 1, 2 (Retubed)	Pick. 3, 4	Pick. 5, 6, 7, 8	Gent.2 Embalse Pt.Lepreau Wolsong 1	Cernavoda 1-5	Bruce 1, 2	Bruce 3	Bruce 4	Bruce 5, 6, 7	Bruce 8	Darl'ton 1-4
Number of Channels	132	306	208	390	390	390	380	380	380	480	480	480	480	480	480
Pressure Tube -Material -In'd dia.(mm) -Wall thick'ss mm	cwZirc.2 82.6 4.20	cwZirc.2 82.6 3.94	htZr2.5%N b 82.6 4.09	cwZirc2 103.4 4.95	cwZr2.5%Nb 103.4 4.06	cwZr2.5%Nb 103.4 4.06	cwZr2.5%Nb 103.4 4.01	cwZr2.5%Nb 103.4 4.19	cwZr2.5%Nb 103.4 4.19	cwZr2.5%Nb 103.4 4.06	cwZr2.5%Nb 103.4 4.06	cwZr2.5%N b 103.4 4.06	cwZr2.5%Nb 103.4 4.11	cwZr2.5%Nb 103.4 4.11	cwZr2.5%Nb 103.4 4.19
Calandria Tube -Material - In'd dia.(mm) -Wall thick'ss mm	Alum. 101.9 1.27	Zirc.2 107.7 1.24	Zirc.2 101.0 1.27	Zirc2 130.8 1.55	Zirc.2 130.8 1.56	Zirc.2 130.8 1.55	Zirc.2 129.0 1.37	Zirc.2 129.0 1.37	Zirc.2 129.0 1.37	Zirc.2 129.0 1.37	Zirc.2 129.0 1.37	Zirc.2 129.0 1.37	Zirc.2 129.0 1.37	Zirc.2 129.0 1.37	Zirc.2 129.0 1.37
Garter Springs -Material -Type -Quantity per FC -Coil dia. (mm)	IncX750 Tight 1 4.75	htZrNbC u Loose 2 7.52	Incon.X750 Tight 2 4.32	htZrNbC u Loose 2 6.81	Incon.X750 Tight 4 4.83	htZrNbCu Loose 2 6.81	htZrNbCu Loose 4 5.59	htZrNbCu Loose 4 5.50	Incon.X750 Tight 4 4.83	htZrNbCu Loose 2 6.81	htZrNbCu Loose 4 6.81	ht ZrNbCu Loose 4 5.59	ht ZrNbCu Loose 4 5.59	Incon.X750 Tight 4 4.83	Inc.X750 Tight 4 4.83
Minimum Axial Creep Allowance	30 yrs	20 years	30 years	13 years	25 years	13 years	35 years	30 years	30 years	14 years	14 years	30 years	30 years	30 years	30 years

Figure 6
The Change in CANDU Fuel Channel Design from Prototype Reactor to Power Reactor

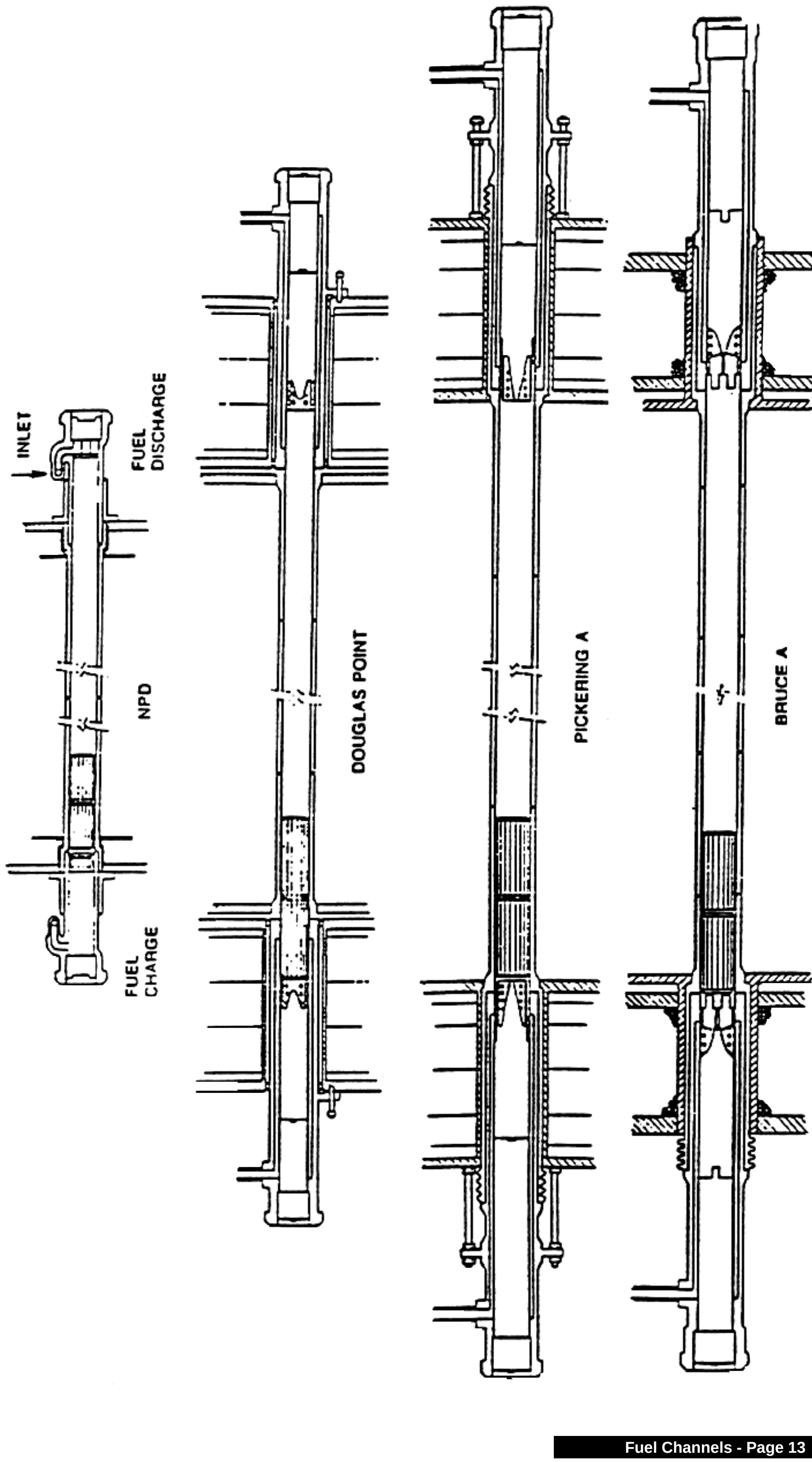
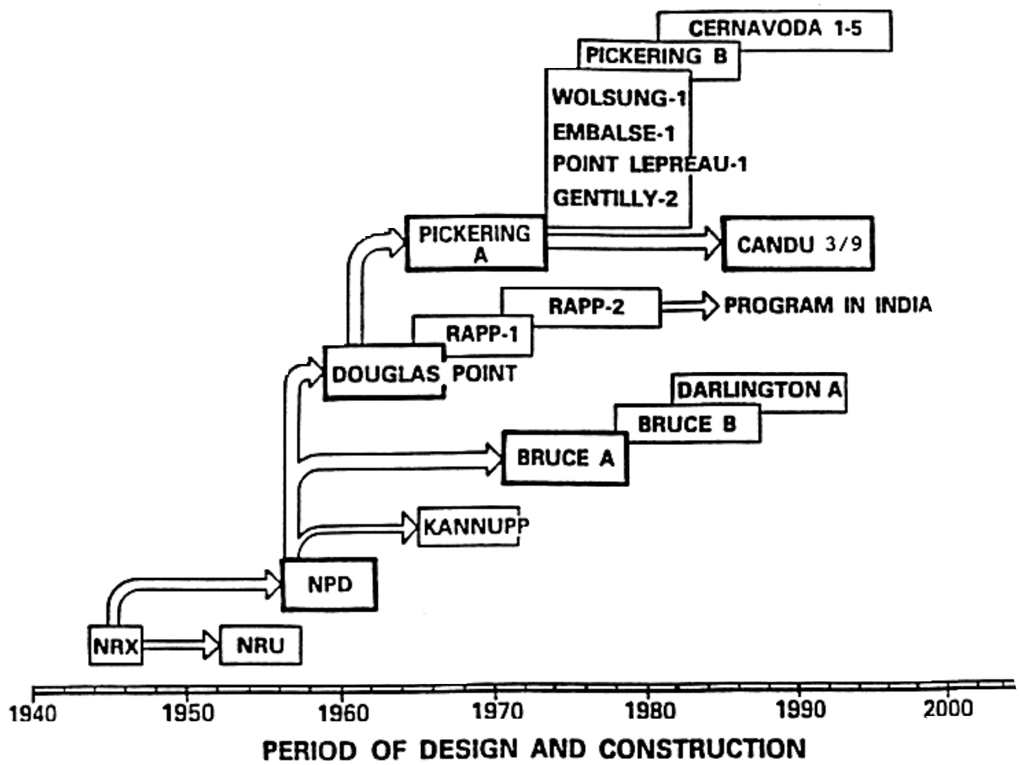


Figure 7
Evolution of CANDU Fuel Channel Design



3. Fuel Channel Design Requirements

The primary purpose of the fuel channels in a CANDU reactor is to support and locate the fuel within the reactor core. They are designed to withstand the coolant flow, temperatures, pressure and transient service conditions imposed by the heat transport system (HTS) throughout the channel design life for all existing units this is 30 years at 80% capacity, giving an effective full power life of 24 years.

Calandria tube assemblies must similarly withstand the conditions imposed on them by the moderator system.

The materials of all in-core channel components must have acceptable corrosion resistance and change in properties, during their design life. In addition, the fuel channel must satisfy the following functional, performance requirements:

3.1 Functional Requirements

The functional requirements of the fuel channel are to:

- support and locate fuel in the reactor core
- permit the HTS coolant flow to efficiently remove fuel heat with low pressure drop and low fuel vibration
- permit free passage of fuel through the reactor core during refuelling

- provide for low leakage fuelling machine connections onto the fuel channel ends at full primary coolant pressure and temperature for refuelling at full power
- form part of the HTS pressure boundary
- provide connections to the HTS feeders
- accommodate thermal, as well as creep and growth dimensional changes
- provide for detection of leakage from the pressure tube or the calandria tube
- provide thermal insulation so the transfer of primary coolant heat to the moderator and end shield coolant is minimized in normal operation
- minimize neutron absorption
- provide shielding to attenuate radiation where they pass through the end shields
- retain shield plugs and channel closures
- allow for fuel removal with one fuelling machine disabled
- be easily replaceable.

3.2 Performance Requirements

Fuel channel performance requirements corresponding to the preceding functional requirements are:

Leakage Requirements

All joints in the fuel channel are designed and built to minimize leakage. Rolled joints are used to connect the pressure tube to the end fittings as well as the calandria tube to the calandria tubesheet. D_2O leakage through these joints which interface with the annulus gas system must be consistent with the background moisture limits for the annulus gas moisture detection system. Sample joints are tested to confirm that target equilibrium leak rates are not exceeded. For the pressure tube rolled joints, test joints must have an equilibrium leak rate of less than 2×10^{-5} atm.cc/sec of helium. For the calandria tube rolled joints, the target leak rate is 6×10^{-7} atm.cc/sec helium after 7.5 min and 2×10^{-6} atm..cc/sec after 15 minutes. Target leak rates for the feeder to end fitting connections are 4.5×10^{-2} g/h of D_2O per connection and for the channel closures are 0.42 g/h of D_2O per closure. During a seismic event the resulting interaction loads between the fuelling machine and channel may result in some leakage. This must be limited to 10 kg per event in total. All welded joints (e.g. the lattice tube to end shield joint) must have no detectable leakage of any kind.

During reactor operation the annulus gas moisture detection system is maintained dry and monitored by dew point trend. Sensitivity in the range of a few grams/hour of D_2O pick-up is practical, so annulus gas dew point trend is used to detect the onset of a leak and the leak rate of the pressure tube. While detecting a leak in the reactor core is relatively easy, locating the specific channel is not, since all the annulus gas tube connections lead into common headers.

Shielding Requirements
Fuel channels must incorporate radiation shielding where they pass through the calandria end shields, so that maintenance and inspection can be carried out in low radiation fields during reactor shutdowns. A maximum dose rate at the feeder cabinet face of 1.0 mSv/h, 24 hours after shutdown of the reactor is the design objective for the shielding.

Leak Detection Requirements
The gas annuli between the outside of the pressure tubes and the inside of the calandria tubes must be capable of being monitored to detect a leak from a pressure tube crack in time to shutdown the reactor before the crack reaches an unstable length, and also to detect any leakage from the moderator or end shields.

3.3 Material Requirements

Material specifications for fuel channel components are listed in CSA Standard N285.6. These define the requirements for impurities, heat treatment, mechanical properties, cleanliness, and testing.

3.3.1 Corrosion and Wear Allowances

Corrosion and wear allowances are established for pressure tubes, calandria tubes and end fittings. These must be taken into account when calculating dimensions for stress analysis of these components. Pressure tube wall thickness must include an allowance for both internal and external corrosion, for thinning due to creep/growth and for wear due to fuel movement. By example, fretting and wear by fuel bearing pads as deep as .46 mm. has rendered the pressure tube unfit for service as determined by stress analysis.

3.3.2 Allowance for the Effect of Environment

The effect of environmental conditions on fuel channel material properties must be taken into account. The effects of stress, irradiation, temperature, hydrogen absorption and any other significant environmental factors on material properties must be accounted for. It must be demonstrated that no significant reduction in the initial stress margins will occur in-service, and that protection against non-ductile failure is provided in accordance with ASME Section III paragraph NB-3211(d).

The combination of stress, temperature and fast neutron flux results in irradiation induced creep and growth of the pressure tube. Pressure tube axial and diametral dimensional changes, as well as fuel channel creep sag must be accommodated.

3.4 Seismic Requirements

The Canadian seismic requirements are documented in the CAN3-N289 series of standards.

Seismic qualification of the fuel channel is performed by analysis as required by ASME, Section III, using the Design Basis Earthquake (DBE) as a level C condition. An analysis is done for the fuel channels as part of the whole reactor assembly to determine the seismic loading on the fuel channels. The magnitude of seismic loads and number of cycles is then provided as input to the fuel channel stress analysis.

Seismic qualification is done to "Category A", which means that the fuel channel must allow HTS coolant to continue to flow through it after a DBE. The fuel channel is analyzed to confirm the integrity of its pressure boundary as part of the seismic analysis.

3.5 Safety Requirements

The pressure boundary integrity of the fuel channel assembly has to be maintained during all postulated emergency cases, as well as during all normal and upset operating conditions, as postulated in the station specific safety reports.

The rupture of any one fuel channel must not lead to rupture of any other channel. Also, under certain postulated accident conditions, the design must provide for effective heat transfer from the fuel channel to the moderator.

There have been two historical cases of channel rupture to date. One on-power event at Pickering in which a pressure tube ruptured and its moderator tube remained intact. A second event at Bruce was the off-power rupture of one pressure tube plus its calandria tube.

3.6 Reliability and Maintainability Requirements

The original design objective for fuel channel components was to achieve high reliability and no channel failures during their design life. It was anticipated, however, that replacement of pressure tubes would be required in a reactor after 30 years operation. Fuel channel design is therefore intended to minimize the cost and radiation exposure associated with this retubing. Downtime associated with retubing is not expected to reduce unit lifetime capacity factor below its designed value. Retubing of Pickering A has taken one to two years per unit.

3.7 Inspection and Testing Requirements

In-service inspection of the fuel channel is performed in accordance with the requirements of the Periodic Inspection of CANDU Nuclear Power Plant Components, CSA Standard N285.4.

Besides mandatory inspections, utilities may institute additional inspections for reasons of maintenance, operational reliability, etc. These may include sampling of pressure tube material using scrapings from the inside of tubes or periodic single channel replacement.

3.8 Codes and Standards

Fuel channel components are manufactured to the CSA Z299 Quality Assurance Standards. Design, analysis, and testing of fuel channel primary coolant pressure boundary components is performed to the requirements of CSA N285.2 for Class 1 components. This standard applies the intent of Section III of the ASME Code, which was written with BWR and PWR pressure vessel reactors in mind. It provides rules for some of the unique features of pressure tube reactors with on-power refuelling. CSA codes provide rules that are supplementary to ASME, Section III, in areas like the design, materials and joining techniques for fuel channels. Fuel channels are registered and controlled under the authority of the local Ministry of Consumer and Corporate Relations (M.C.C.R.).

3.8.1 Design

The design approach for the Class 1 fuel channel components uses the design by analysis process of ASME, Section III, Subsection NB3200. The ASME Code rules are complemented by additional requirements of the Canadian Standard, CAN/CSA-N285.2, in particular to address the possibility of delayed hydride cracking in zirconium alloys.

To prevent failure due to hydride cracking, CSA N285.2 requires various measures. It imposes limits on maximum tensile stress under design level A and level B service conditions, plus the initial residual tensile stress. In addition, it is required that pressure tubes be supported such that they do not contact the calandria tube, as this could produce thermal gradients which may lead to hydride accumulation and the potential for unstable cracking.

The pressure tube material is also required to be capable of sustaining a detectable leak-before-breaking, and a system capable of detecting pressure tube leaks before they grow to unstable length is required.

3.8.2 Materials Pressure Tubes

For pressure tubes, a Zr-2.5% Nb alloy has been developed. The condition of the material is cold-worked and this condition is not presently listed in the ASME Code, Section III, Appendix 1 as a material for Class 1 application. Requirements have therefore been specified in Canadian Standard CAN/CSA-N285.6.1-88, "Seamless Zirconium Alloy Tubing for Fuel Channels". Some early CANDUs used Zircaloy 2 for pressure tubes, which is also covered by this standard. The design data for these alloys are provided in CAN/CSA-N285.6.7-88, "Zirconium Alloy Design Data".
For inspection and non destructive examination of zirconium alloy components,

the requirements of the ASME Code, Section V, are supplemented by the additional requirements of CAN/CSA-N285.6.6-88, "Inspection Criteria for Zirconium Alloys".

Calandria Tubes

Calandria tubes are part of the moderator pressure boundary. The calandria is a Class 3 system that has a design pressure of 70 to 85 Kpa(g) and a design temperature of 100°C. Normal operating temperature is less than 85°C. Calandria tubes are made from rolled Zircaloy 2 strip, seam welded, drawn and annealed in accordance with CAN/CSA-N285.6.4-88, "Thin Walled, Large Diameter Zirconium Alloy Tubing".

End Fittings

A high strength end fitting material is needed in the region of the pressure tube rolled joint to withstand the rolling forces and maintain the residual stress needed to retain strength and leak tightness. Strength, toughness and corrosion requirements are met by a modified type 403 stainless steel. The modifications are in the chemical composition limits and include:

- the range of the main alloying elements such as carbon and chromium, which are restricted to reduce variability in tensile and impact properties;
- the allowable limits of residual elements such as phosphorus, sulphur and copper are lower to improve corrosion resistance and to reduce the shift in RT-NDT with irradiation. Cobalt is restricted for activity reasons. The requirements for this material, its inspection and design data are specified in CAN/CSA N285.6.8-88.

3.8.3 Joining Zirconium to Steel

Zirconium alloys and steels cannot be welded satisfactorily due to the formation of brittle intermetallic compounds. Proprietary mechanical joint designs have therefore been developed for pressure tubes and calandria tubes to join them to their fittings, tubesheets or housings.

Development programs have defined rolled joint designs that involve residual compressive stress for sealing. In addition, the pressure tube material is extruded into grooves in the end fittings to provide pull out strength, such that the joints will not separate under service loading.

The rules which these rolled joints shall meet are provided in Canadian Standard CAN/CSA N285.2-N89. Prototype joints are required to be qualified by both test and analysis for Class 1, and by test or analysis for other classes. Production joints are then to be made using the same procedures and tooling design as used in the qualification of the prototypes.

Each production joint is tested for leakage and its structural integrity is verified by ensuring that parameters determined from the prototype joints have been achieved.

Feeder Joints
Choice of type 403 stainless steel for end fittings complicates the use of welded joints for feeder connections without post weld heat treatment. Therefore, a mechanical joint has been used. Feeder joints have generally been developed from the Grayloc joint (a proprietary bolted joint design employing a solid metal gasket), but other proprietary designs are not excluded. The joint design is governed by the requirements of CSA N285.2-N89.

4. Fuel Channel Components

In the following, the key characteristics of each of the fuel channel components is described.

4.1 Pressure Tubes
Pressure tubes are the most important part of the fuel channel as they pass through the calandria and contain the fuel bundles. They are zirconium alloy tubes that are about 6 metres long, about 10 cm in diameter and have a wall thickness of about 4 mm. Because CANDU reactors use natural uranium fuel (without enrichment), it is important to use low neutron absorption material in the reactor core. Therefore, all in-core structural materials are zirconium alloys while the moderator and primary coolant are heavy water. Special attention is also paid to the design and manufacture of in-core structures, like the pressure tubes, to ensure that no material is used beyond that necessary to satisfy the appropriate design code.

The first two commercial CANDU reactors initially used cold-worked Zircaloy 2 pressure tubes, but all of the later reactors and the retubings of the early units have used cold-worked Zr-2.5% Nb tubes. This newer alloy, which was developed by AECL, is about 20% stronger than Zircaloy 2, so tubes having a thinner wall thickness could be used, which improves fuel burnup. An extensive test program has been carried out to obtain the information needed to design and qualify Zr-2.5% Nb pressure tubes for reactor use.

4.1.1 Pressure Tube Design
A pressure tube acts as a horizontal beam supporting itself, as well as the fuel and coolant which it contains. The pressure tube is supported at each end, by its attachment to end fittings, and at intermediate points by the surrounding calandria tube via garter spring spacers.

The design of the pressure tube consists primarily of the determination of the length, the inside diameter and the wall thickness of a simple thin-walled cylinder. The length of the pressure tube is primarily determined from the core length obtained from reactor physics considerations. The inside diameter of the pressure tube is derived from fuel passage and thermohydraulic considerations. The wall thickness of the pressure tube is determined by the stress analysis of a pressure retaining component subjected to the appropriate internal pressure and mechanical loads. This analysis complies with the Class I vessel design-by-analysis rules of Section III of the ASME Boiler and Pressure Vessel Code. The large diameter to wall thickness ratio (26:1), allows the evaluation of the pressure tube stresses using standard classical elasticity equations for thin-walled cylinders and permits the thermal stresses to be neglected.

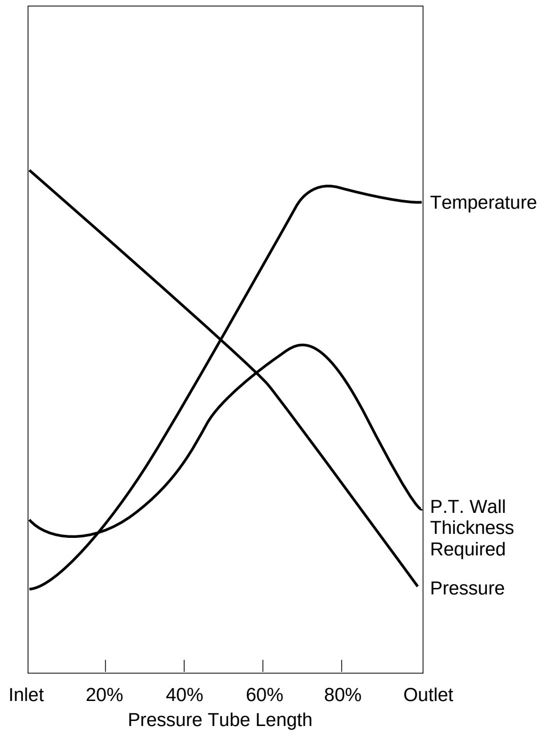
One of the main requirements of the pressure tube design is to optimize wall thickness to minimize neutron absorption. Since pressure and temperature vary along the length of the pressure tube when a reactor is operating, as is illustrated in Figure 8, the design condition is established by evaluating stresses at 20 locations along the length of the tube to determine which location requires the largest wall thickness. In these calculations, the design pressure is determined by the set point of the primary heat transport system relief valves. The design temperature is that of a central fuel channel running at 110% of normal fuel channel rating, plus an additional margin of about 10% to cover a possible instrument measuring error.

Allowances for corrosion and wear are added to the calculated minimum wall thickness to determine the minimum allowable wall thickness for pressure tube fabrication.

In addition to coolant temperature and pressure, the following are considered during the calculation of stresses and assessment of the fatigue life for the pressure tube:

- weight of fuel channel components, fuel and coolant
- feeder pipe loads and torques
- fuelling machine loads
- axial loads due to bellows, fuel movement and end fitting bearing friction
- annulus spacer loads
- effects of tube initial bow, misalignment and end of design life sag and elongation
- loads imposed during a seismic event
- HTS pressure relief valves
- temperature of central fuel channel at 110% ± 10%
- allowance for corrosion and wear
- stress load

Figure 8
Typical Pressure and Temperature Variation Along the Length of a Fuel Channel and Their Influence on Required Pressure Tube Wall Thickness



Pressure tube design analysis take account of the effects of creep and growth, which causes the pressure tubes to increase in length and diameter, and deflect by sag from the weight of the fuel and coolant, as is illustrated in Figure 9. Pressure tube stresses are evaluated for both beginning of life and end of design life conditions to account for dimensional changes that occur over the life of the plant. At the beginning of life, the initial wall thickness and unirradiated material properties apply. At the end of life, pressure tube diameter and length have increased, while wall thickness has decreased and material strength has increased. For the end of design life condition, credit is taken for only a small fraction of the strength increase obtained from irradiation. While strength increases with radiation, ductility is decreased.

Although zirconium alloys are not included in the ASME Boiler and Pressure Vessel Code, the pressure tube allowable design stress was established by an extensive test program on the same basis as that for ASME Class 1 materials. That is, the allowable design stress is the lowest of either one third of the minimum tensile strength or two-thirds of the minimum yield strength, both taken at the design temperature. For cold-worked zirconium - 2.5% niobium, one third of the minimum tensile strength is the governing condition. On this basis the design stress is about 160 MPa at 300°C.

4.1.2 Manufacture of Pressure Tubes

Cold-worked Zr-2.5% Nb pressure tubes for CANDU reactors are made by hot extruding hollow billets into tubes that are then cold-worked to produce the final dimensions. These fabrication techniques have been developed so pressure tubes can be consistently produced within close dimensional tolerances and with uniform physical and mechanical properties. All manufacturing steps are closely controlled to ensure that the tubes meet very stringent specifications. For example, to ensure that pressure tubes are free of unacceptable defects, the ingots, billets, and finished tubes are each ultrasonically inspected.

The fabrication process for cold-worked Zr-2.5% Nb pressure tubes, which is illustrated in Figure 10, is summarized in the next few pages:

Figure 9 PT
Creep and Growth

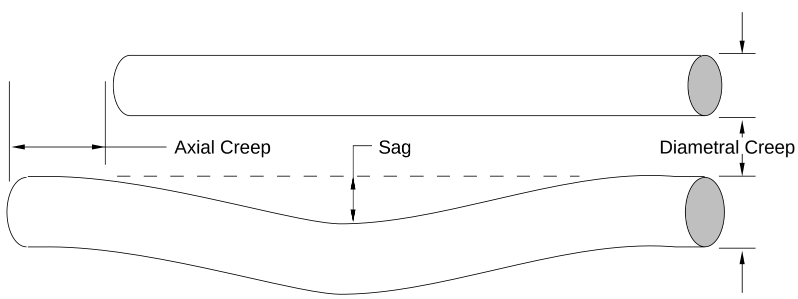
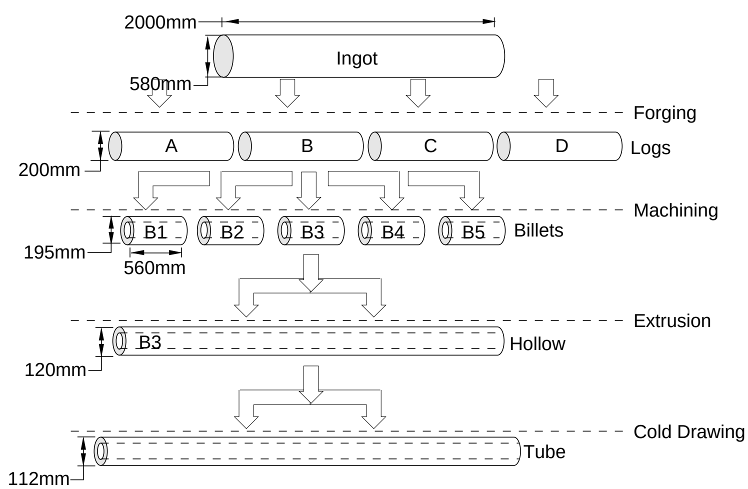


Figure 10
Schematic of Pressure Tube Production Sequence



Ingot

Zirconium sponge, which is produced from zirconium silicate ore, is compacted into briquettes along with a master alloy of zirconium and niobium. These briquettes are then electron beam welded together to form a long rod that is melted by passing a current through it in a consumable electrode arc furnace to form a Zr-2.5% Nb ingot about 0.6 m in diameter. This melting process, which is done at least twice, produces material almost free from inclusions, with a very low impurity content. The ingots, which can each produce about 30 tubes, are machined to remove surface defects and are then ultrasonically inspected for piping defects using a normal beam technique on the machined cylindrical surface. The machined ingot is also checked for conformance with the specification requirements for the two major alloying elements (niobium and oxygen) and 23 residual elements.

Billets

The ingots are preheated to almost 1000˚C and forged into round bars about 0.25 m in diameter using flat and rotary forges. A second chemical analysis is done, then the bars are machined into about 30 hollow billets. To refine the grain size and produce a more uniform microstructure, the machined billets are beta quenched from 1290°K (1017°C). The machined billets are then ultrasonically inspected in a spiral scan using the pulse-echo immersion technique. Two angled shear wave beams, travelling in opposite directions around the circumference, and one compression wave beam propagating in a radial direction, are used.

Extrusion

The short hollow billets are preheated to about 850˚C for extrusion into tube hollows, which is the most critical step in determining the uniformity in the final tube wall thickness.

Cold Drawing

The tube hollows are then cold drawn in two passes, each 12 to 15% reduction in area. The cold drawn tubes are first sand blasted on the inside and are then honed with about 0.064 mm being removed from the inside surface to ensure that any small defects and embedded sand particles from the sand blasting operation are removed. The outside of the tubes are centreless ground to the required wall thickness.

After the honing and centreless grinding operation, an ultrasonic immersion inspection is conducted that includes angled shear wave examination with two beams travelling in opposite directions around the circumference and two beams travelling in opposite directions parallel to the axis of the tube.

The product specification requires that the following tests and examinations also be carried out on each pressure tube before it can be accepted:

- Hydrostatic pressure test to about twice the operating stress.
- Chemical analysis of an off-cut to ensure conformance of the four major alloying elements (Niobium, Oxygen, Nitrogen, Hydrogen).
- Tensile testing of an off-cut at 300˚C to ensure conformance with the specified strength requirements.

- Corrosion testing of a small sample to confirm that the metallurgical condition is as required.

"Off cut" archive samples are saved for future needs.

Finishing Operations
To minimize dimensional changes during service, the tubes are stress relieved in an autoclave for 24 hrs at 400°C. This treatment produces a hard adherent oxide layer on the tubes which acts as a hydrogen barrier and provides some wear resistance during operation. The pressure tubes finishing operations also involve:
- A cleaning operation.
- A visual examination inside and out.
- A duplicate ultrasonic examination.
- An eddy current examination for internal surface flaws.
- A dimensional inspection (corrections of ovality and bow are allowed to meet the dimension requirements).

4.2 End Fittings
In channel designs, the need to make and break high temperature, high pressure connections at power for refuelling, and the decision to use carbon steel piping in the out-reactor circuit were satisfied by having a fitting between the pressure tubes in the core and the carbon steel piping. This end fitting is an out of core extension of the pressure tube that provides the intermittent connection for on-power fuelling, the connection to the carbon steel feeders and the connection with the pressure tube.

As illustrated in Figure 11, each fuel channel end fitting passes through a lattice tube in one of the reactor end shields. Contacting surfaces of the end fitting and the lattice tube which supports it are made of hardened tool steel to form sleeve bearings that allow an axial sliding motion of the end fittings caused by both pressure tube permanent elongation and thermal expansion. The inboard end of each end fitting is connected to one end of a pressure tube by a rolled joint. The outboard end contains a removable closure and provides facilities on which a fuelling machine can clamp and make a high pressure seal to allow on-power refuelling. Near the outer end of each end fitting is a side port for connection of a feeder pipe by a bolted connection.

A shrunk-on ring provides an attachment for welding to the bellows which join the end fitting and the lattice tube to seal the gas annulus between the pressure tube and the calandria tube.

Figure 12 illustrates that inside each end fitting there is a steel liner tube which is the same diameter as the pressure tube so that fuel bundles can pass through the end fitting on their way into or out of the pressure tube. During reactor operation, a shield plug is latched into the liner tube to provide radiation shielding. Coolant flow is not blocked by the shield plug as this flow can pass through holes in the liner tubes and then move along the outside of the liner tube.

4.2.1 End Fitting Design

End fittings are designed as Class 1 vessels with finite element stress analysis being performed to show that they satisfy the stress limits in Section III of the ASME Code.

Figure 11
Schematic of a CANDU Reactor

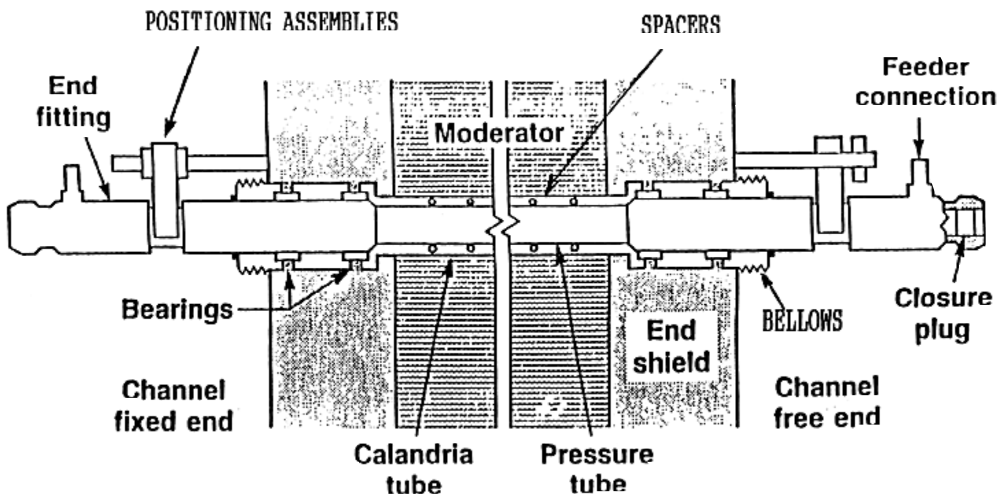


Figure 12
Schematic of a Fuel Channel Assembly

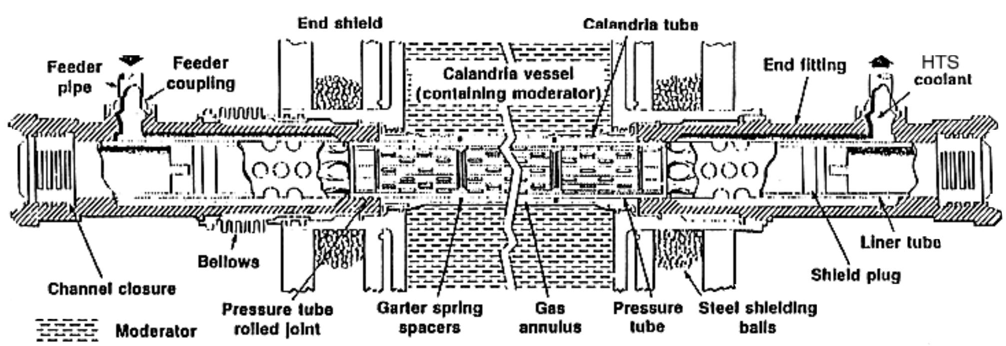
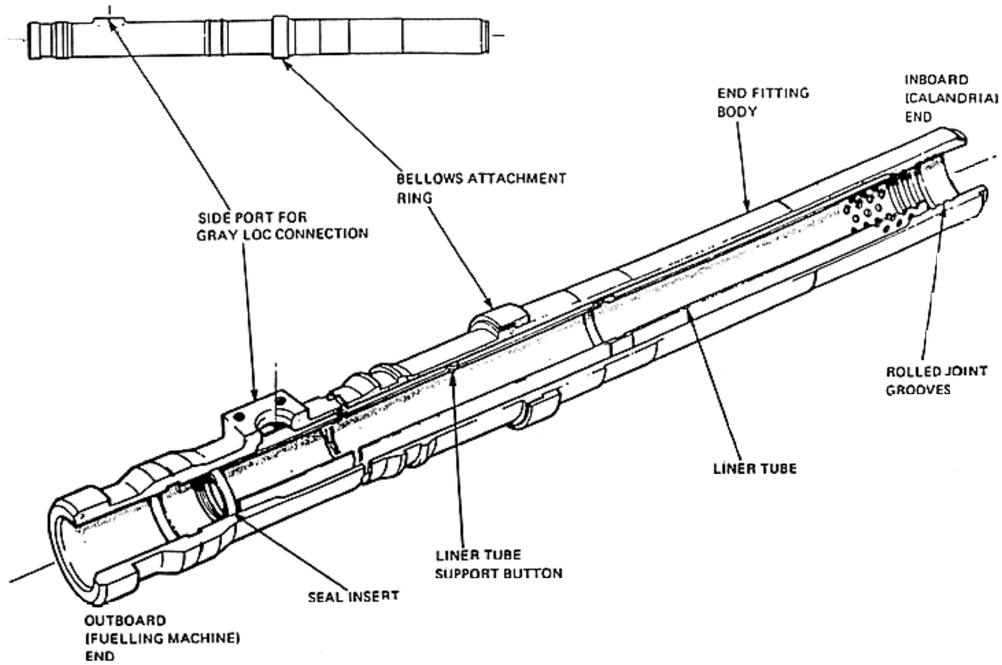


Figure 13
End Fitting Assembly



4.2.2 Manufacture of End Fittings

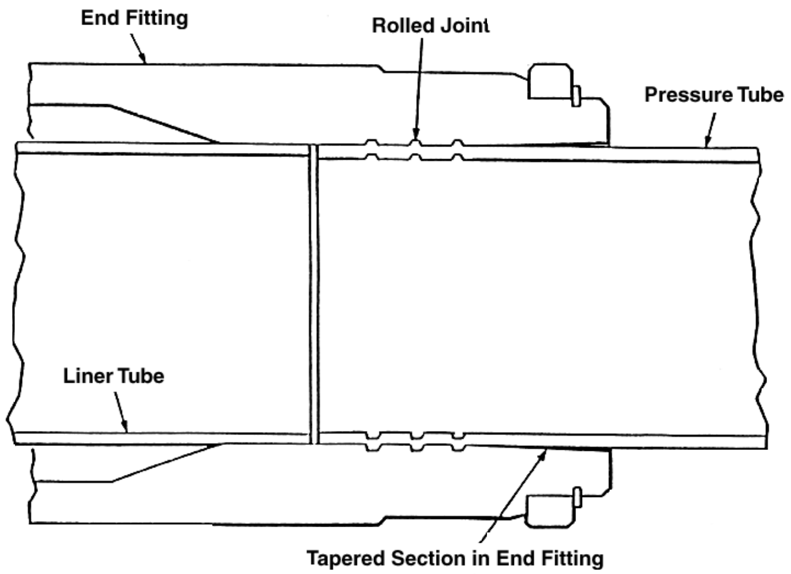
To meet the combination of high strength and corrosion resistance needed by the pressure retaining portion of the end fittings, they are made from a modified AISI type 403 stainless steel. End fittings are forged in one piece, heat treated and then machined to the finished configuration, which is illustrated in Figure 13. They are subjected to rigourous quality controls during manufacture.

4.3 Pressure Tube Rolled Joint

Each end of a pressure tube is roll expanded into an end fitting. As can be seen in Figure 14, the pressure tube rolled joint design is very simple, involving only three grooves in the end fitting hub. Pressure tubes are assembled into their end fitting hub with a shrink fit prior to rolling. This is known as a "zero clearance" joint. The pressure tube is roll expanded into its hub, with the rolling tool illustrated in Figure 15 creating about a 13% reduction in wall thickness. Although no rules are given in the ASME code for the use of roll expanded joints in Class I vessels, their use has been justified and approved for this application by extensive development testing and analysis.

Sample pressure tube rolled joints have been subjected to an extensive test program to demonstrate their acceptability for use in fuel channels. A finite element stress analysis of the rolled joint is conducted to confirm that the stresses in the rolled joint region meet the allowable limits specified by Section III of the ASME Code. During fuel channel installation in the reactor core, each rolled joint receives a careful dimensional inspection to verify that rolling has been done correctly, and the leak tightness of each joint is checked using highly sensitive helium leak detection techniques.

Figure 14
Pressure Tube Rolled Joint



4.4 Calandria Tubes

A calandria tube surrounds each pressure tube. Calandria tubes have an internal diameter of about 129 mm and span the calandria vessel between the two end shields. These tubes, which are arranged on a square pitch to form a circular lattice array, provide access through the calandria for the pressure tube/end fitting assemblies. The calandria tubes help to support the fuel channel pressure tubes by means of four spacers per channel.

Calandria tubes also act as axial stays resisting the internal pressure of the moderator in the calandria vessel. The ends of the calandria tubes are rolled with stainless steel inserts into grooves in the calandria tubesheets, forming high-integrity sandwich joints, as is illustrated in Figure 16.

Calandria tubes are made of seam-welded, annealed and stress relieved zirconium alloy (Zircaloy 2). This material was chosen for its low neutron absorption cross-section and corrosion resistance. The tubes are expanded (belled) at each end to allow clearance for their rolled joint insert. A thick insert is required for roll joint strength, since there would be too much "spring back" of the relatively thin walled calandria tube if rolled by itself into the rigid tube sheet.

Figure 15
Diagram of Rolling Process

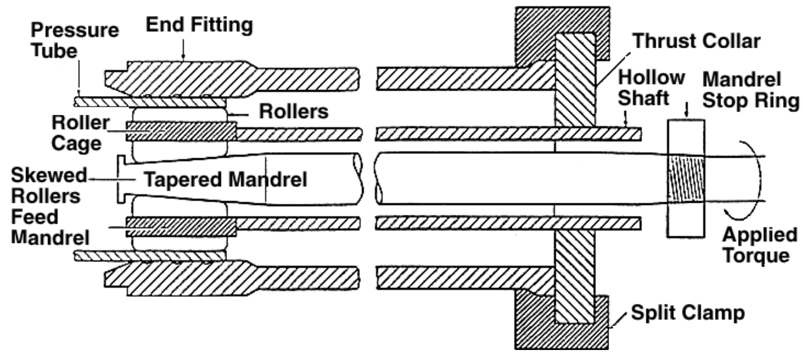
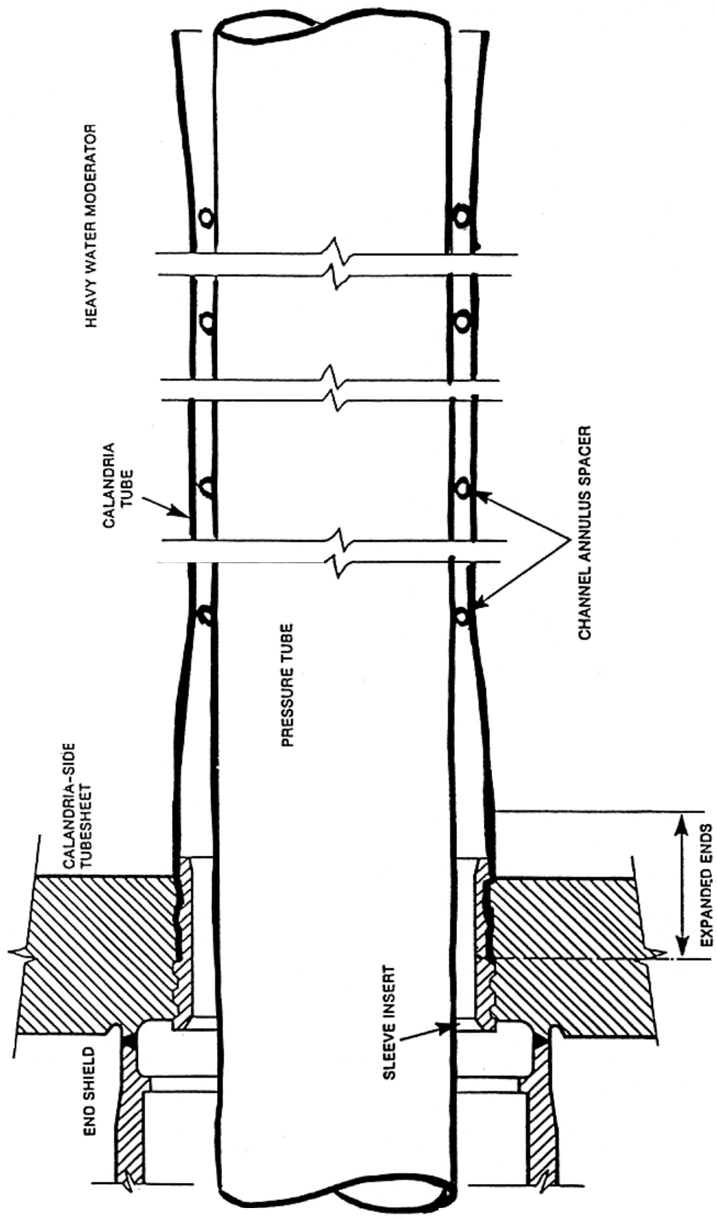


Figure 16
Schematic of Installed Calandria Tube



The calandria tubes serve as the boundary between the relatively cool moderator and the gas-filled annular gap surrounding the hot pressure tubes. This gap minimizes heat loss to the moderator from the fuel channels.

The calandria tube design ensures that in the event of abnormal external pressurization of the calandria tubes, they will elastically collapse onto the pressure tubes and spring back when the pressure is reduced.

4.5 Annulus Spacers

Each pressure tube is separated from, and supported by, a calandria tube by means of four spacers, as is illustrated in Figure 17. These spacers are positioned about a meter apart so that pressure tube sag will not allow a pressure tube to contact the calandria tube surrounding it for at least 30 years.

The spacers are made by forming Inconel wire into a close coiled helical spring. The current spacer design has hooks on the two ends of these springs so that they can be stretched around a pressure tube and then hooked together. Therefore, when installed, the annulus spacers conform to the outside diameter of a pressure tube with their tight fit on the pressure tube preventing them from moving away from their installed positions. Stretching of these springs, as well as the existence of a diametral gap between each spacer and the calandria tube surrounding it, allows for diametral growth of the pressure tubes. Axial movement of the pressure tubes is allowed by a rolling motion of the annulus spacers, which results in almost no wear on the pressure and calandria tubes where they contract the spacers.

Spacers have been qualified for in-reactor use by an extensive test program to demonstrate their acceptability.

Early CANDU units used a spacer design that was a loose fit on the pressure tube rather than the current spacer design that is a tight fit on the pressure tube, as is indicated in Figure 18.

4.6 Feeder Pipe Connection

The feeder pipe connection located on the side of each end fitting is a bolted connection having a metallic seal, as illustrated in Figure 19. Four bolts pass thorough a flange into holes tapped into the end fitting body to tighten this connection. The flange holds a hub welded to each feeder pipe tightly against the metal seal ring. This connection is shown to satisfy the stress limits of ASME, Section III using a finite element model.

Figure 17
Annulus Spacer

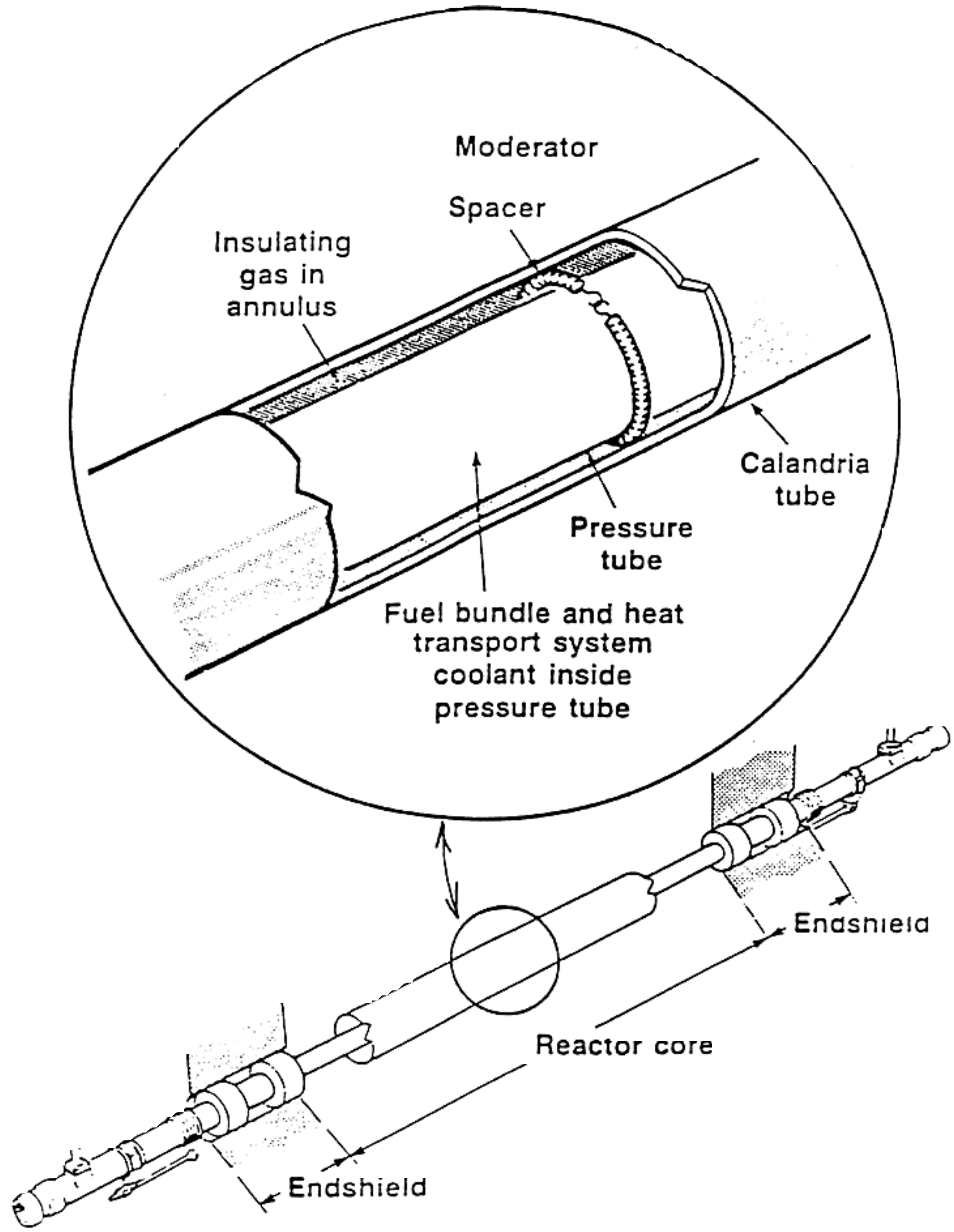


Figure 18
Comparison of the Two Types of Pressure Tube to Calandria Tube Spacers

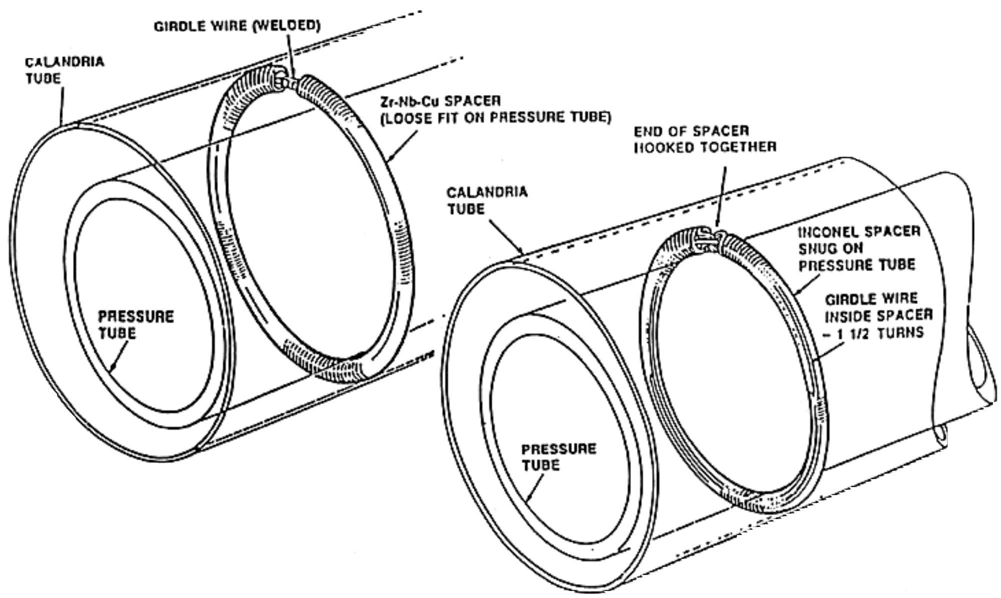
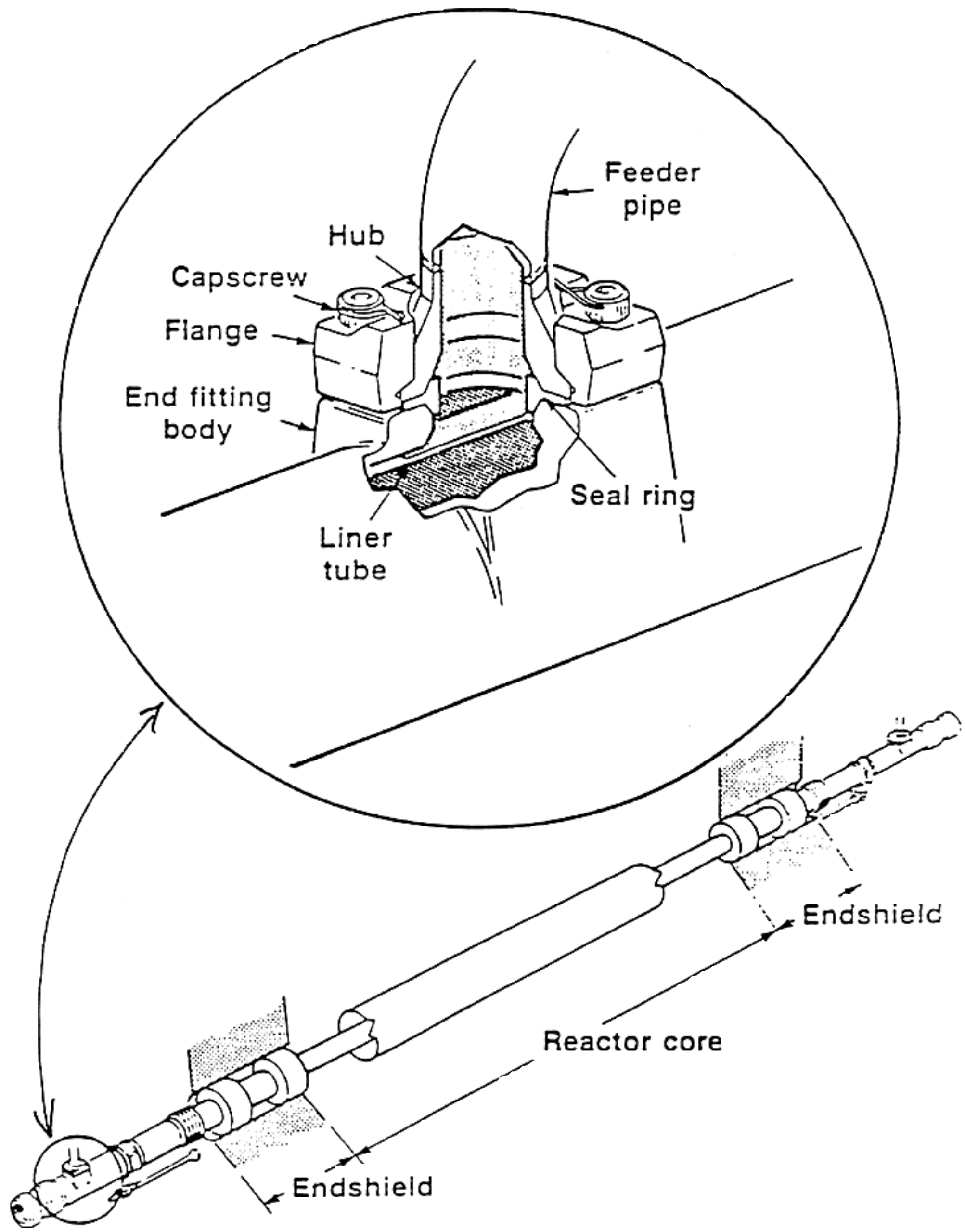


Figure 19
Feeder Coupling



4.7 Annulus Bellows

Each end of the annulus between a calandria tube and a pressure tube is sealed by an Inconel bellows, as is illustrated in Figure 20. The bellows, which connects between an end fitting and the reactor end shield, allows axial motion of the channels and also limits the torque imparted to the end fitting by the feeder piping. Each end of the bellows is welded to an end ring. One end ring is attached to the lattice tube/calandria tubesheet by welding and another is a shrink-fit onto the end fitting. A small diameter tube welded to the bellows connects the sealed annulus that surrounds each fuel channel to the annulus gas system which circulates dry CO_2 gas and monitors the moisture content of this gas.

Proof tests are performed on prototype bellows.

4.8 Positioning Assembly

Each fuel channel is located axially within the reactor by a positioning assembly (Figure 20) which is connected to one end shield. A second positioning assembly is installed at the other end of the fuel channel but it is not attached to the end shield so axial motion resulting from thermal expansion, as well as from pressure tube permanent elongation, is permitted. This second positioning assembly exists so the point of channel restraint can be readily changed once the elongation allowance is used up at one end of the reactor.

4.9 Channel Closures

Channel closures (Figure 21), which can be remotely removed by a fuelling machine, are located in each end of a fuel channel to seal the primary coolant and to permit on-power access to the fuel channel by the fuelling machines. Operationally there has never been a major loss of channel integrity due to a closure failure, however there have been many cases of excessive leakage due to debris getting trapped between the seal plate and seal ring during removal and replacement.

Figure 20
Positioning Assembly and Bellows

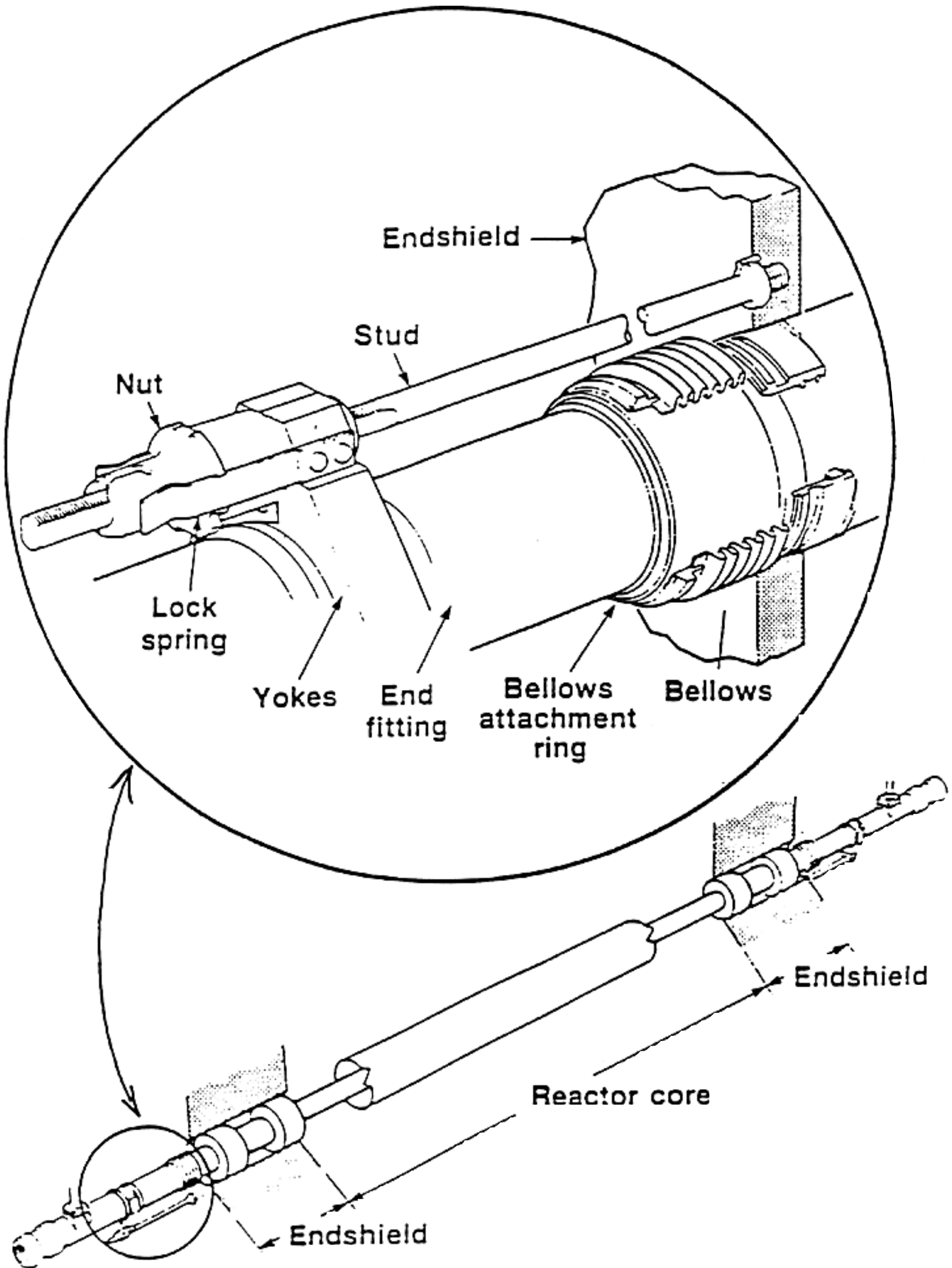
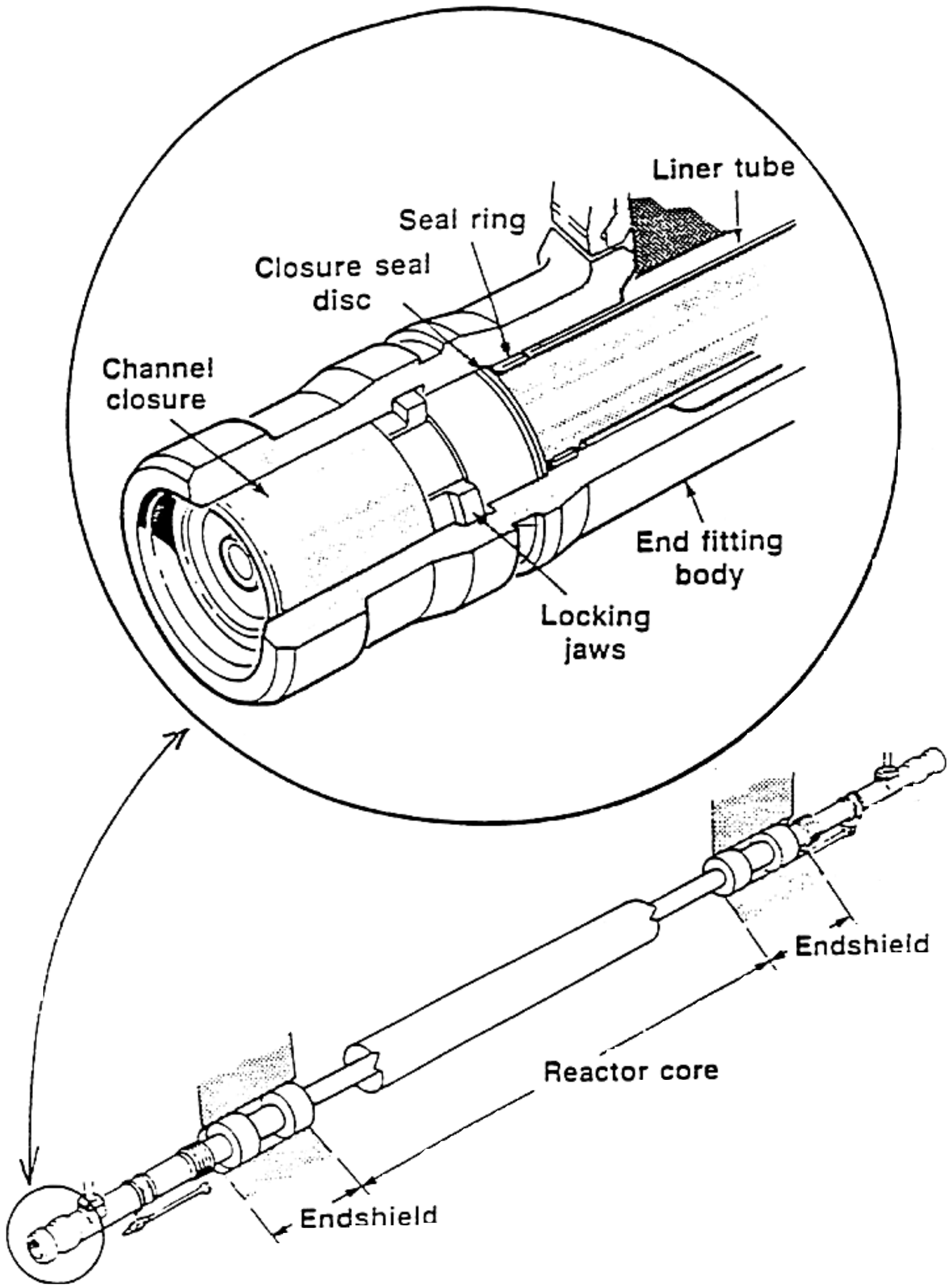


Figure 21
Channel Closure



4.10 Shield Plugs

The shield plugs, which provide shielding where the fuel channels pass through the reactor end shield, are latched into the end fitting (see Figure 12). They are also removed by the fuelling machine before the refuelling of a channel can occur.

5. Fuel Channel Analysis

5.1 Stress Analysis

The pressure boundary stresses associated with the operation of a complete fuel channel assembly are calculated and documented in a Stress Report that is registered with the appropriate regulatory group before fuel channels can be installed into a reactor core.

This fuel channel analysis considers the following mechanical loads;

Axial Loads	**ASME Classification**
Fuelling Machine Clamping	Primary
Fuelling Machine Ram	Primary
Hydraulic Drag	Primary
Fuel Friction	Primary
Bellows (Extension)	Primary
Feeder Pipe	Primary
Bearing Friction	Secondary
Earthquake	Primary (Emergency)
Vertical Loads and Moments	
Due to Dead Weight	Primary
e.g. Fuel, D_2O, Components)	
Bellows attachment shrink fit loads	Secondary
Interactions:	
- Between CT and PT (at GS)	Primary
- At Bearings	Primary
- Due to Skew and Misalignment	Secondary
Earthquake	Primary (Emergency)
Torsional Moments	
Feeder Pipe	Primary
(Shared between PT and Bellows)	

5.2 Deformation Analysis

Pressure tube deformations due to creep and growth are also calculated. These deformations are obtained using formulae developed by AECL based on measurements of tube deformations in operating reactors and extensive research reactor testing.

Calculation of channel sag is done to establish:

- Number and spacing of garter spring spacers,
- Garter spring loads,
- Clearance for fuel passage,
- Clearance to reactivity mechanism guide tubes.

Calculations of diametral creep are used to establish:
- The allowable garter spring spacer diameter,
- HTS coolant flow conditions at end of life,
- That pressure tube end of life diameter is conservatively less than the allowable creep ductility established from tests in test reactors.

Calculation of axial creep is used to establish:
- Bearing lengths,
- Bellows extension,
- Feeders and feeder connection stresses,
- Wall thinning of the pressure tube.

To accommodate the pressure tube permanent elongation (about 150 mm) due to the creep and growth that occurs during reactor operation, axial movement of a fuel channel is permitted by the end fitting bellows and sleeve bearings, as well as by the annulus spacers located between each pressure tube and calandria tube.

5.3 Clearances with Interfacing Components

An assessment is made of component clearances and how they are affected by:
- Temperature and Pressure
- Manufacturing Tolerances
- Installation Tolerances
- Hydrostatic Loads
- Differential Pressure Tube Creep
- Radiation Growth

The most important of these clearances are:
- End Fitting Inboard End to Calandria Tube Insert
- End Fitting to Lattice Tube Shielding Shoulders
- Bellows to Adjacent Feeder Pipe
- Feeder Cabinet Insulation to Fuelling Machine Snout
- Journal/Bearing Engagements
- Fuel Cavity Length
- Bellows Weld Location

5.4 Design and Analysis Documentation

Fuel channel documentation that must be prepared is:
- System Classification List (SCL)
- Design Requirement (DR)
- Design Description (DD)
- Design Manual (DM)
- Stress Report (SR) for pressure boundary components
- Design Drawings

6. Fuel Channel Installation

The fuel channels are assembled and installed into the calandria vessel at the reactor site following installation of the calandria.

The main installation operations are the roll expansion of the pressure tube ends into the two end fittings, the welding of the end fittings to the bellows and the installation of the positioning assemblies. Following installation and inspection of all channels, the feeder pipes are connected to the end fittings.

Special tooling and detailed working procedures are used for fuel channel installation. The crews are trained in the proper use of the tools on full size mock-ups prior to actual installation of the fuel channels in a reactor core and the entire operation is closely controlled to ensure the level of quality assurance needed during the fabrication of an ASME Class 1 vessel.

Fuel channels can be replaced during the life of the station, using proven tooling and procedures similar to those used during the initial fuel channel installation.

7. Fuel Channel Operating Conditions

The heat transport system operating pressure is one of the key elements in optimizing the CANDU systems. High primary pressure permits high secondary pressures and increased efficiency. However, high primary pressure also requires thicker walled pressure tubes, and this results in a fuel burnup penalty. Recognizing these considerations, the operating pressure of CANDU units at the reactor outlet header has always been set at about 10 MPa.

Table 3 shows values of key channel design conditions for all commercial CANDU units Conditions documented are flux, pressure, temperature, HT flow rate and its quality. Peak pressure inside channels has increased from about 9 -½ MPa for Pickering to about 11 MPa for CANDU 6 and Darlington. There are essentially only two values of maximum flux for all operating units. One value of maximum flux applies to all Pickering units, which has a flattened flux profile, while all other units have a cosine flux shape whose maximum flux is about 30% higher. Peak temperature of channels has slightly increased with each channel design from being just below 300˚C in Pickering to 313˚C for Darlington. Maximum HT flow rate for channels has also increased from its Pickering value of just over 20 kg/s to closer to 30 kg/s for newer channel designs. The quality of the HT flow in channels has also changed from being zero at Pickering, i.e., no boiling, to allowing a few percent quality in more recent designs. Darlington maximum flow and quality are a bit less than at the Bruce or CANDU 6 units because Darlington feeders have a much smaller variation of flow.

Table 3
CANDU Fuel Channel Design Conditions

	Startup Year	No. of Channels	Max. Flux (n/cm2/s)	Peak Pressure (MPa)	Peak Temp. (°C)	Max. Flow (kg/s)	Max. Quality (time avg.%)
Pickering A	1971/72	1560	2.76×10^{13}	9.44	297	21.8(avg.)	0
Pickering B	1983/86	1520	2.76×10^{13}	9.48	297	21.8(avg.)	0
Bruce A	1977/79	1920	3.69×10^{13}	10.16	308	23.90	3.45
Bruce B	1985/87	1920	3.69×10^{13}	10.30	308	25.50	4.68
CANDU 6	1983/84	1520	3.71×10^{13}	10.98	312	27.78	5.81
Darlington	1990/93	1920	3.69×10^{13}	11.05	313	26.10	1.51

All channels have the same HTS chemistry specification, i.e., pH is 10 to 11 and dissolved D_2 is 3 to 10 m*l*/kg. The oxygen concentration remains below 10 ppb.

8. Operating Performance

Table 4 lists the 24 Candu units that are currently operating. The performance of Pickering A and Bruce A has been greatly affected by "delayed hydride cracking" (D.H.C.), and the later units have benefited from the experience gained.

Delayed Hydride Cracking has the following basic characteristics in Zirconium alloys:

1. Initiation of a crack begins at a point of stress concentration such as a notch, scratch or defect.
2. There must be a component of stress exerted in the material and the speed of crack progression is proportional to the stress level beyond some initiation point. Crack progression will stop if the stress is removed.
3. The metal must contain Hydrogen or Deuterium and the higher the concentration the greater the susceptibility to D.H.C., because D.H.C. will progress only if Hydrogen or Deuterium comes out of metal solution as a brittle hydride precipitate. Precipitation of hydride occurs when solubility limits are exceeded.

Referring to Figure 24 the solubility limits vs temperature is demonstrated and is the key to operational limits at full power and temperature of say 300°C the unit may well operate with 50 mg/Kg. with no progression of D.H.C. because the Hydrogen and Deuterium remain in solution. However the dilemma for the operator is in shutting down the unit to a cold condition or starting up the unit from cold. Since he must maintain pressure above liquid saturation through the transients, it means there will be precipitated hydride and stress levels high enough to progress D.H.C. So categorically, the cold-pressurized reactor state created the highest risk of D.H.C.

All pressure tube leaks or ruptures that have occurred in Candu units have been a D.H.C. mode of failure; only the initiating circumstances have been different.

Table 4
Spacer Design for CANDU Operating Reactors

CANDU Unit	In-Service Date	Initial Spacer Design
Pickering 1*	1971	Loose
Pickering 2*	1971	Loose
Pickering 3*	1972	Loose
Pickering 4*	1973	Loose
Bruce 1	1977	Loose
Bruce 2	1977	Loose
Bruce 3	1978	Loose
Bruce 4	1979	Loose
Pickering 6	1982	Loose
Point Lepreau, New Brunswick	1983	Loose
Gentilly, Quebec	1983	Loose
Wolsong 1, Korea	1983	Loose
Embalse, Argentina	1984	Loose
Pickering 5	1984	Loose
Bruce 6	1984	Loose, but repositioned before startup
Pickering 7	1985	Loose, but repositioned before startup
Bruce 5	1985	Loose, but repositioned before startup
Pickering 8	1986	Loose, but repositioned before startup
Bruce 7	1986	Loose, but repositioned before startup
Bruce 8	1987	Tight
Darlington 2	1990	Tight
Darlington 1	1992	Tight
Darlington 3	1992	Tight
Darlington 4	1993	Tight

**Rehabilitation of these units has removed the initial spacers and replaced them with the tight spacer design.*

8.1 Pressure Tube Deformation

In-reactor pressure tube deformation occurs at significantly faster rates in a reactor than during creep tests performed outside of the reactor. Therefore, because early deformation predictions were made from empirical equations derived primarily from out-reactor tests (and only a limited amount of in-reactor tests), provisions incorporated to accommodate deformation in the initial Pickering and Bruce fuel channel designs were insufficient to last their 30 year design life. The end fitting bearings are too short and the spacings between feeder pipes are too small.

Improved equations for predicting pressure tube deformation, based on both theoretical models and periodic measurements taken on the fuel channels in the

initial Pickering and Bruce reactors, have now been developed to accurately predict all of the in-reactor creep and growth effects. These equations are accurate within 25%, which includes scatter in the measurements and variations in metallurgical properties between tubes. Because these new equations were used to design all CANDU reactors built after the initial Pickering and Bruce reactors, all later units are expected to be able to accommodate all fuel channel deformations that will occur during 30 years of reactor operation.
To avoid the problem of channel bearings being too short, it was necessary only to provide longer bearings in new designs. Similarly, to prevent feeder pipes from contacting, it was necessary only to increase their initial spacings. These increases were minimized by arranging to accommodate half the total pressure tube axial deformation at each end of the reactor.

To cope with predicted pressure tube diameter increase and fuel channel sag, the size and number of annulus spacers associated with each fuel channel were changed from the values defined for the initial Pickering and Bruce reactors. Twice as many smaller diameter spacers have been used in the later fuel channel designs.

8.2 Leaking Pressure Tube Rolled Joint Cracks

Pressure tube leaks have occurred at the rolled joints for both Pickering Units 3/4 and Bruce Unit 2 which required a large scale inspection and pressure tube replacement program.

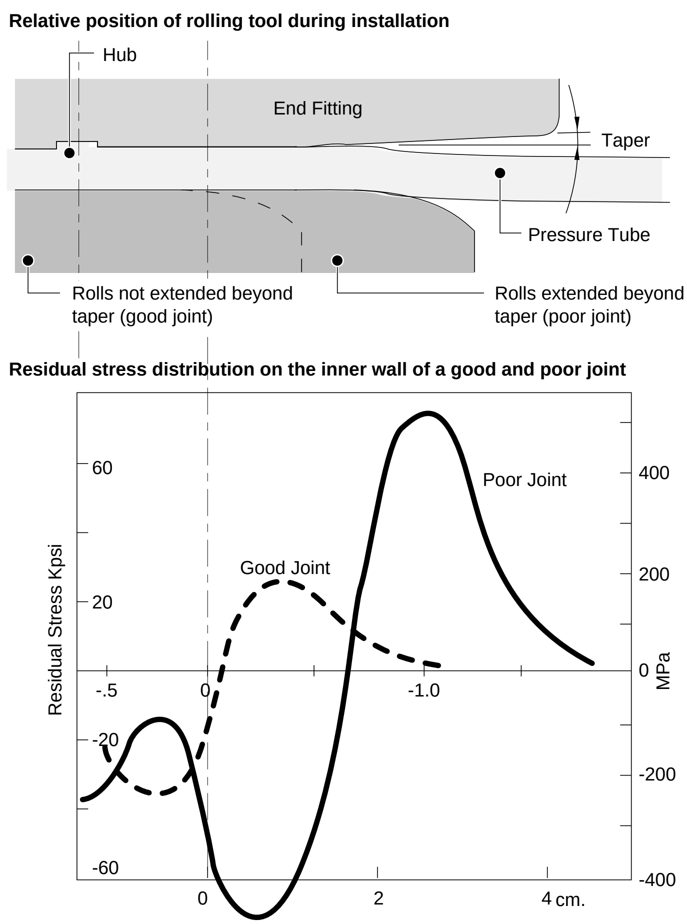
Examination of the cracked tubes showed that short axial cracks had formed by the delayed hydride cracking mechanism as a result of a very high residual stress in the portion of the pressure tube immediately adjacent to the rolled joint. An incorrect rolling procedure, which permitted the roll expander (see Figure 15) to be inserted too far into the pressure tube, had expanded the tubes where they were not supported by the end fitting. This procedure produced very high residual stresses. Figure 22 shows the difference in typical residual hoop stress profiles between properly rolled and "over-rolled" joints.

Remedial measures have been introduced in all CANDU reactors built since the initial Pickering and Bruce units so that delayed hydride cracking is not expected to occur in these newer stations. Improper rolling procedures used for the initial Pickering and Bruce rolled joints have been revised to avoid over-insertion of the roll expander and rolled joint design have been modified to make it less sensitive to roller location. These changes reduce the likelihood of any further over-rolling of pressure tubes and excessively high residual stresses in rolled joints.To further reduce the residual stresses experienced in the initial Pickering and Bruce rolled joints, a modified rolled joint assembly procedure was also initiated. The clearance between the pressure tube and the end fitting bore has been reduced so that the diametral assembly clearance in the joint now ranges from an interference of 0.18 mm to a clearance of only 0.05 mm, whereas the initial Pickering and Bruce designs had a tube/hub clearance ranging from

0.18 to 0.53 mm. The very tight assembly fit of the new zero clearance rolled joint is accomplished by thermally expanding the end of the end fitting to permit insertion of the pressure tube before roll expansion of the pressure tube into the end fitting.

Fortunately, time at operating temperature slowly annealed and reduced stress in the rolled joints. This has stopped crack progression and limited further failures at Pickering units 3, 4 and Bruce 2 after the initial early failures.

Figure 22
Rolled Joint Residual Stresses



8.3 Leaking Pressure Tube at Manufacturing Flaws

In 1986, channel N06 of Bruce Unit 2 leaked at a manufacturing defect, near the pressure tube outlet end, after the defect had propagated by delayed hydride cracking. Cold pressurization caused the defect to rupture during a leak test after the unit had been shutdown. The calandria tube also ruptured allowing fuel pencils to be blown into the calandria.

Inspection techniques available at the time of manufacture and installation of this tube were not capable of detecting the manufacturing flaw which was the initiating defect. Improvements to the inspection technique for later units provide very high confidence that pressure tube defects of this kind were detected before the units entered service. In addition, the phenomenon of `a shrinkage pipes lap defect' in ingots was studied and the parameters during ingot fabrication were changed to eliminate these `shrinkage pipes'. This is expected to significantly reduce the probability of such lap type defects in pressure tubes.

8.4 Pressure Tube Rupture

In general, if a small defect exists, or develops, in a pressure tube the consequence will be leakage, before rupture. The existence of such a Leak-Before-Break (LBB) behaviour for pressure tubes is demonstrated in the preceding discussion of the detected pressure tube leaks that have occurred at operating units. However, the following event did not exhibit LBB and led to the early retubing of 2 units.

On August 1, 1983, the Zircaloy 2 pressure tube in fuel channel G16 at Pickering Unit 2 ruptured without prior detectable leakage. It was found that the failure of this pressure tube was associated with a combination of factors:

- A larger than anticipated pickup rate of deuterium, from the primary heat transport and annulus gas systems, to levels significantly exceeding the terminal solid solubility (TSS) for deuterium in Zr-2. TSS values as a function of temperature are shown in Figure 24.
- Movement of the outlet end garter spring had permitted contact between the pressure tube and calandria tube. This resulted in a significant thermal gradient in the pressure tube and deuterium migration towards the cooler surface.
- Growth of hydride "blisters" at the cold spot of the pressure tube outer surface, with the resulting stresses initiating a defect in one of them.
- Growth of a delayed hydride crack from the defect, leading to ultimate failure.

Since any hydrogen/deuterium that remained near the inner surface of this tube was in solution, when a crack initiated near the blisters it could not propagate to the tube inside surface by the delayed hydride cracking mechanism. Crack propagation occurred only in the axial direction until the remaining thin web on the inside surface was overloaded and broke. Because the length of the resulting through-wall crack exceeded the critical crack length, there was a sudden extension of the crack to a length of about 2 meters.

There were two primary causes of this Pickering pressure tube rupture. One cause was the high hydrogen/deuterium level in the Zircaloy 2 tube, created by corrosion of its inner surface by the coolant contained inside it. The second cause was a significant mislocation of one annulus spacer, which allowed the hot pressure tube to sag and contact the cool calandria tube surrounding it. This contact created a large temperature gradient in the wall of the pressure tube. Both of these causes have now been corrected.

All CANDU reactors built after Pickering Units 1 and 2 have used Zr-2.5 Nb pressure tubes instead of Zircaloy 2 tubes. Examination of several Zircaloy 2 and Zr-2.5% Nb pressure tubes has shown that the deuterium pickup rate of Zircaloy 2 material is significantly higher than that of Zr-2.5% Nb.

The design of annulus spacers has also been revised so that they now are a tight fit on the pressure tube and can not become displaced.

8.5 Pressure Tube Fitness-for-service Assessments

Draft Fitness-for-Service Guidelines have recently been developed to provide acceptance criteria and evaluation methods for assessment of the integrity of the Zr-2.5% Nb pressure tubes in operating CANDU reactors. The Guidelines provide the methodology for the evaluation of specific conditions, in a single tube, such as manufacturing flaws, in-service generated flaws, and hydride blisters formed at points of contact between a pressure tube and its calandria tube, as well as for the evaluation of the generic degradation of pressure tube properties in-service. The Guidelines complement the rules of Section XI of the ASME Code and the requirements of the Canadian Standards Association (CSA), CAN3-N285.4, periodic inspection of pressure tubes in operating CANDU nuclear reactors. These inspections ensure that unacceptable degradation in component quality is not occurring and the probability of failure remains acceptably low for the life of the reactor.

For pressure tubes, a key consideration in the assessment of their fitness-for-service is to study their possible hydrogen embrittlement. Therefore, it is important to know the hydrogen/deuterium content of pressure tubes. Until recently, data about the ingress of deuterium into pressure tubes from the primary heavy water coolant that flows through them could only be obtained by removing tubes from operating units. However, such data can now be obtained by a non-destructive examination of operating tubes that involves scraping a small sample from a tube, as is illustrated in Figure 23. As reactors get older, greater statistical knowledge is required. By example inspection of all 390 channels was completed to support the last 16 months of Pickering Unit 4 operation before retubing.

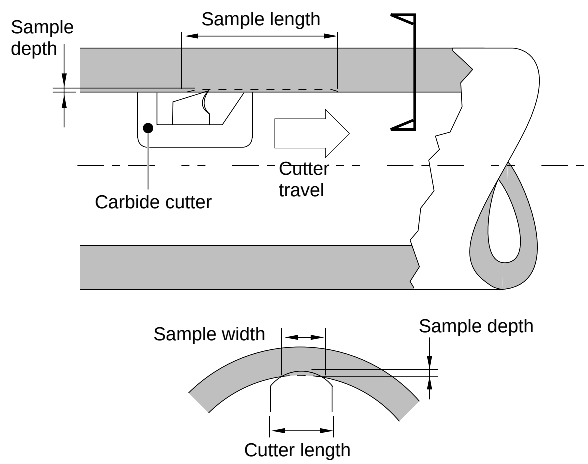
The Guidelines consist of three sections. The first section describes the requirements that must be met to qualify the tubes for continued service. It contains evaluation procedures and acceptance criteria for assessing crack-like flaws, notch type flaws, pressure tubes in contact with their calandria tubes and generic changes in material fracture toughness. The second section provides the material properties database information needed to carry out the assessments. The third section provides the technical basis for the acceptance criteria and evaluation procedures, as well as justifications and descriptions of the databases. The draft Guidelines were issued to CANDU reactor operators for trial use, and released to the Atomic Energy Control Board of Canada for review and comment, in May 1991.

The assessment methodologies outlined in the Guidelines are briefly discussed in the following.

8.5.1 Sharp Flaw Assessment

For the evaluation of pressure tubes found to contain sharp or crack-like flaws, analytical procedures are defined for calculating the sub-critical flaw growth that could occur during the evaluation period. Acceptance criteria have been developed to ensure there is adequate protection against failure by unstable fracture, plastic collapse and delayed hydride cracking (DHC). In addition, if the hydrogen/deuterium concentration is high enough that DHC could occur at operating temperature, it must then also be demonstrated that, if the flaw were to penetrate the tube wall, the leak would be detected and the reactor shutdown before the postulated crack becomes unstable, i.e., Leak-Before-Break (LLB) must occur, not rupture.

Figure 23
Schematic of a Pressure Tube Inspection Tool To Measure Its Deuterium Content



8.5.2 Blunt Flaw Assessment

To assess if pressure tubes containing notch type flaws may be exempted from the crack-like flaw evaluation criteria, it is necessary to demonstrate both that the flaw is not sharp, and that it will not become sharp during the evaluation period. For CANDU pressure tubes, initiation of a crack at the root of a notch type flaw can occur because of fatigue crack initiation or delayed hydride cracking.

In the Guidelines, the assessment of blunt-type flaws utilizes a two-step approach. The first step involves an evaluation to determine if crack initiation by fatigue or DHC will occur during the evaluation period. If crack initiation is predicted, then the flaw must be treated as crack-like.

If the flaw remains blunt, and TSS (see Figure 24) is not exceeded at operating temperature, then it is only necessary to show that the flaw has adequate protection from failure by plastic collapse. If the hydrogen/deuterium concentration is high enough for DHC at operating temperature, then LBB must also be demonstrated.

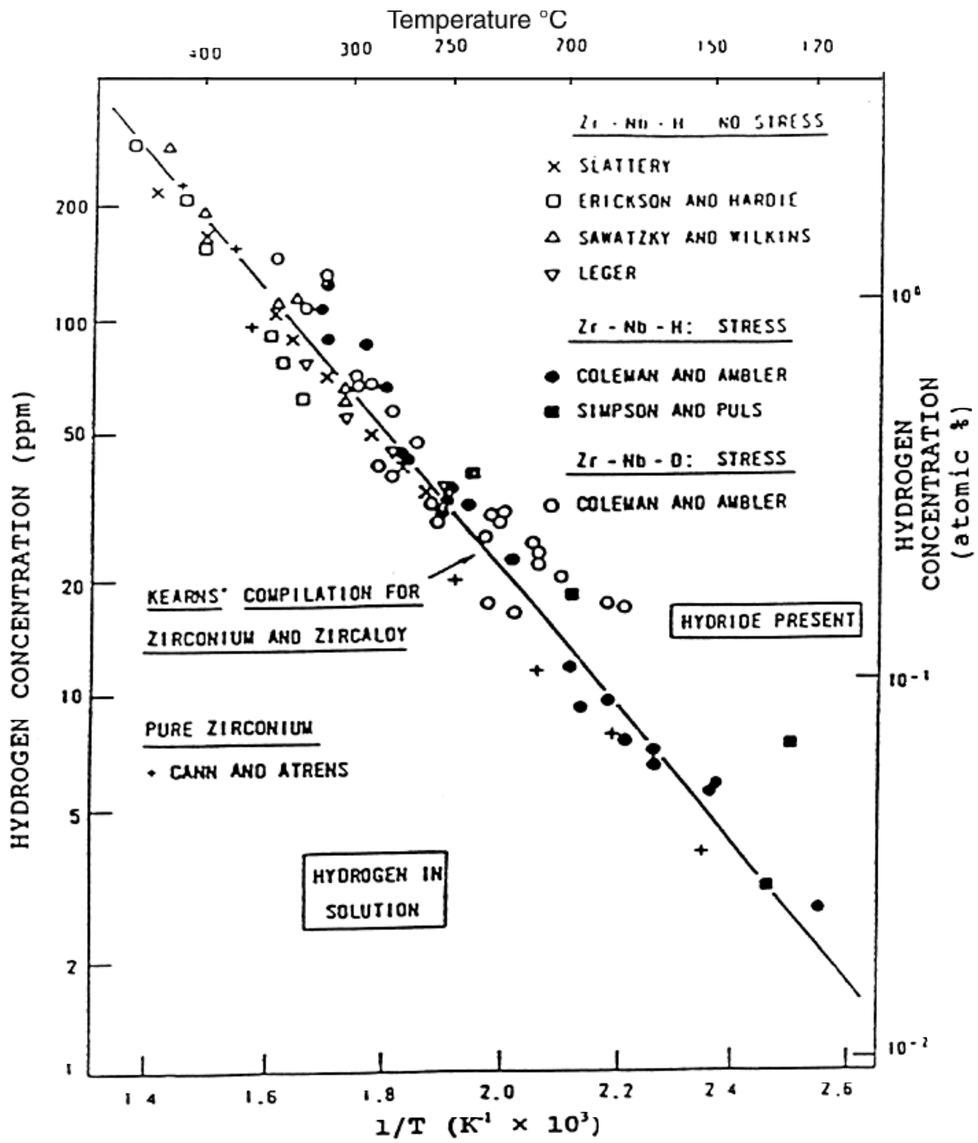
8.5.3 Blister Assessment

For the evaluation of pressure tubes in contact with their calandria tubes, and with sufficient hydrogen plus deuterium to develop hydride blisters at the point of contact, analytical procedures and acceptance criteria are being developed to ensure that the blisters will not grow beyond a maximum acceptable size during the evaluation period. Otherwise, the contact would have to be eliminated.

8.5.4 Generic Assessment

For the evaluation of the generic degradation in fracture properties for pressure tubes, analytical procedures and acceptance criteria are provided to ensure that there continues to be adequate protection against unstable fracture.

Figure 24
Comparison of Measurements of Solubility Limits (TSS) of Hydrogen in Zirconium Alloys



Criteria to assess the acceptability of the tube fracture properties when hydrides are not present in the pressure tube during nominal full power conditions only consider changes in the material properties due to fast neutron fluence. To assess the acceptability of the material fracture properties when hydrides are present at normal full power temperatures, both the effects of increased hydride concentrations and neutron fluence on the material properties have to be considered.

It is proposed that an evaluation procedure similar to that of Appendix G of ASME, Section III, be used. This procedure requires that a conservative flaw be postulated based upon the manufacturing inspection and in-service inspection techniques applicable to the pressure tubes of the reactor unit being evaluated. It must then be demonstrated that the stress intensity factor at the postulated crack tip is sufficiently less than the fracture toughness of the material.

For conditions in which Figure 24 indicates that the hydrogen/deuterium concentration is greater than the terminal solid solubility (TSS) at nominal full power operating conditions, it must be additionally demonstrated that there is a low probability of having manufacturing flaws that could be susceptible to DHC, that there is a low probability of initiating DHC at in-service induced flaws such as crevice corrosion, fret marks or fuel scratches and that leak-before-break (LBB) is assured.

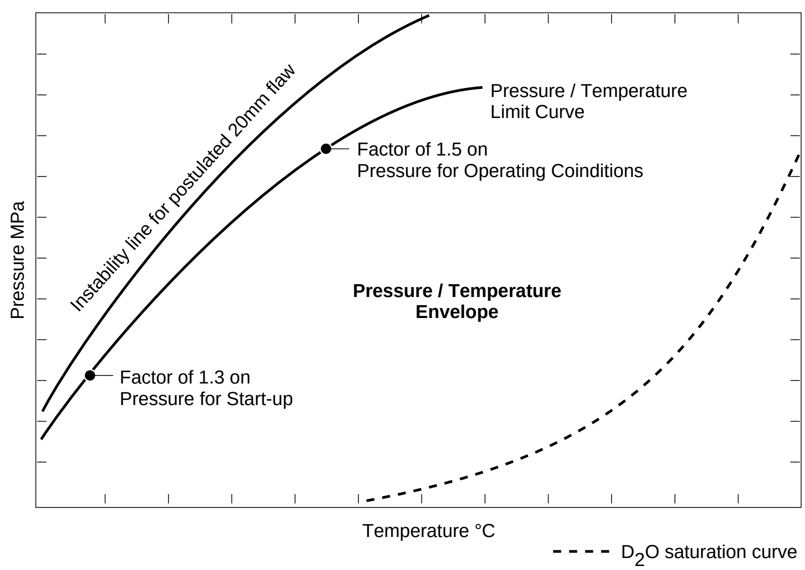
Such generic assessments would define a pressure vs. temperature operating envelope like that shown in Figure 25.

8.5.5 Leak-Before-Break Assessment

A key part of the pressure tube fitness procedures is a defense-in-depth demonstration that tubes will operate in a condition that if they do fail, they will leak at a rate sufficiently large that the leak will be detected and the reactor shutdown before unstable crack propagation occurs. This condition is known as Leak-Before-Break (LBB). The Guidelines use of LBB is a defense-in-depth because it must first be shown that DHC can not occur, and then it must be shown that even if DHC did occur, its consequence would only be LBB.

Figure 25

Pressure / Temperature Envelope for Pressure Tubes



To assure LBB in CANDU pressure tubes it is required that:

- the crack length at wall penetration be less than the Critical Crack Length (CCL) for unstable propagation;

- the leak from this through-wall crack be detected and the reactor put into a cold, depressurized condition before the crack length exceeds the CCL.

Although the nuclear industries outside Canada are also developing procedures for the application of LBB, these are for application to stainless and ferritic steels in light water pressure vessel reactors and are not directly applicable to Zr-2.5% Nb pressure tubes because of several reasons, like:

- The consequences of failures in steel vessels and piping in light water reactors are different from those of a single pressure tube failure in a CANDU reactor.
- The annulus gas monitoring system in CANDU fuel channels provides a very sensitive leak detection capability which allows for a rapid response.
- The following is a description of the acceptance criteria for the deterministic and probabilistic LBB approaches.

a. Deterministic LBB approach. The proposed acceptance criterion for the deterministic LBB approach is:

$$t > T$$

where:

t = the lower bound calculated time for an assumed through-wall crack to grow from its length at penetration to the critical crack length (i.e., the time available to detect the leak and to take action);

T = the upper bound response time of the leak detection system, which includes the time required to confirm the presence of a leak at normal operating conditions, and the time required to reach a cold depressurized condition.

This criterion assures that leaking coolant would be detected, and the reactor shutdown, before a through-wall delayed hydride crack becomes unstable.

The methodology for calculating the time t, as illustrated in Figure 26, is

$$T = \frac{CCL - L}{2V}$$

where:

L = the upper bound length of the crack at initial leakage, which is about five times the pressure tube wall thickness;

CCL = the lower bound critical crack length from slit burst tests on tube sections (this is the minimum length of an axial through-wall crack that would be unstable at the temperature and pressure being evaluated);

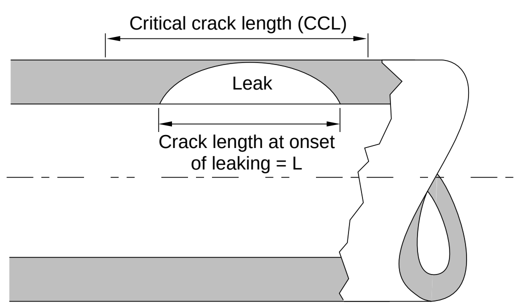
V= the upper bound delayed hydride crack growth rate in the axial direction at the temperature being evaluated.

Although this deterministic LBB assessment is not used as the primary defense against unstable rupture, it still incorporates a conservative approach by using the worst combination of parameters.

b. Probabilistic LBB Approach. The proposed acceptance criterion for the probabilistic LBB approach is to demonstrate that the probability of $t < T$ is less than 1×10^{-3}, which corresponds to an event frequency of less than 10^{-3} occurrences/reactor-year.

In the CANDU design, failure of a pressure tube is not assumed to be an incredible event as the design incorporates a number of safety systems to accommodate such an event. In CANDU safety reports, which are an essential part of CANDU licensing, an event frequency of $< 2 \times 10^{-3}$ occurrences/reactor-year is used for the scenario in which both the pressure tube and calandria tube are assumed to fail. The proposed LBB acceptance criterion of $\leq 10^{-3}$ occurrences/ reactor-year was conservatively selected so it would not contravene the safety analysis.

Figure 26
Pressure Tube LBB Criterion



$$\text{Time available to take action} \cong \frac{CCL - L}{2V} > \text{Station response time}$$

(V = Crack Face Velocity)

9. Summary

The fuel channels in CANDU reactors, which use thin-walled zirconium alloy pressure tubes, represent a specialized application of pressure vessel design. The fuel channels have made a significant contribution to the very high capacity factors attained in CANDU reactors since they allow on-power refuelling.

Fuel Handling

Training Objectives

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

- List the reasons why the designers chose "on-power" fuelling for CANDU reactors instead of "shutdown" fuelling.
- List the major design requirements of "on-power" fuelling systems for CANDU reactors.
- Trace the normal fuel path for all CANDU designs.
- Compare fuel flow path differences and similarities between Pickering/CANDU 6 and Bruce/Darlington Fuel Handling Systems.
- Compare the F/H design concepts of the A.E.C.L. design Pickering/CANDU 6 and C.G.E. designed Bruce/Darlington.
- Describe the step by step sequence of fuelling operation - and the corresponding reactor dynamics of the Pickering/CANDU 6 type of Fuel Handling System.
- Describe the sequence of fuelling operation at a Bruce/Darlington reactor and focus on the differences compared to Pickering/CANDU 6.
- List of operational Policies, Procedures and routines considered necessary before committing to fuelling any specific fuel channel.
- Identify the major components and their function of the Pickering/CANDU 6 Fuelling Machine.
- Identify the major components and their function of the Bruce/Darlington fuelling machine and draw comparison to the Bruce/Darlington design.
- Generate a list of F/H auxiliary support systems and show comparisons between the Pickering/CANDU 6 and the Bruce/Darlington systems.
- Describe a block diagram to illustrate the basic Fuel Handling Control System and understand the basic modes of control.
- Describe the current strategy of fuel storage in water bays and dry storage flasks.
- Describe the commitment to safeguards of the Canadian Government and be able to list the basic types of safeguard technique used on Canadian reactors.
- Generate a list of hazardous situations, and scenarios plus the type of hazards expected. Consider strategies to minimize hazards.

Table of Contents

1.0 Fuel Handling Introduction

The Canadian designed CANDU (CANada Deuterium Uranium) nuclear reactor is based on a heavy water moderated and cooled reactor that currently operates on a once-through natural uranium fuel cycle in a horizontal Fuel Channel/Pressure Tube Core. There is no need for enriched fuel in the CANDU system.

"On power" fuelling versus "off power" fuelling provides the following advantages:

- The Candu enriched core has a low excess reactivity and a new core requires refuelling after about 150 full power days, therefore with "off power" fuelling, frequent unit outages would be required with resulting costly reduction in capacity factor. The only economical alternative to "on power" fuelling would have been to enrich the core or provide enriched "boosters" to decrease the fuelling outage frequency.
- Continuous "on power" fuelling optimizes fuel burn-up by progressively moving bundles from low flux outer regions to central core and *vis-versa* so all bundles can be removed after equal burn-up periods, translating into low fuel cost.
- "On power" fuelling allows the addition of small incremental positive reactivity gains on an as required basis (about 0.1 to 0.2 mk/day), this minimizes the required depth of negative reactivity control, maximizing neutron economy and low fuel cost.
- "On power" fuelling provides a "flexibility" of operation on short term basis allowing for developing trends such as flux tilt or "hot" and "cold" spots. Defective fuel can be removed from core to minimize the bulk fission product in the heat transport system. Locating defect fuel is dependent on having a system for locating it; the most successful system has been a "delayed neutron detection system" with sampling from each channel (eg. CANDU 6 Pt. Lepreau).

On-power fuelling of CANDU reactors originated with the 20 MW(e) Nuclear Power Demonstration (NPD) reactor which entered service in 1962. It was recognized at an early stage that CANDU reactors, burning natural uranium fuel with inherently low excess reactivity margins, would be more economical to operate if they utilized on-power fuelling. This was further demonstrated with on-power fuelling of the 200 MW(e) Douglas Point reactor in 1968. These early fuel handling systems formed the basis for design and development of equipment used in the succeeding commercial units.

Commercial power plants have been built in single-unit and multi-unit stations. Some of the multi-unit stations are equipped with dedicated fuelling machines (F/Ms) for each reactor unit, whereas others are equipped with a shared system.

2.0 Candu Specific On-power Fuelling Requirements

The purpose of the fuel handling system is to provide on-power fuelling capability at a rate sufficient to maintain continuous reactor operation at full power.

During on-power fuelling, the fuelling machine becomes an extension of the reactor fuel channel end fitting and is subjected to the pressure in the primary heat transport system. The fuelling machine is designed as a reliable high integrity device. Portions of it are equivalent to static pressure vessels and meet the requirements of the ASME Boiler and Pressure Vessels Code, Section III Class 1. Other portions, such as elastomeric hoses, the failure of which would result in release of fluid, are designed to CAN/CSA-N285.2-M89, Requirements for Class 1C, 2C and 3C Pressure Retaining Components and Supports in CANDU Nuclear Power Plants. Additionally, the on-power fuelling concept requires the fuelling machine pressure boundary and its support structure to be seismically qualified to the requirements of CAN3-N289.3-M81, Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants.

Since the fuelling machine must visit different reactor fuel channels, fuel transfer port and auxiliary ports, its support system must provide the required transport mobility, for routine refuelling operations and fuel storage. This must be done without loss or jeopardy of reactor building containment, loss of shielding or inadequate fuel cooling.

The F/H system and equipment is designed for reliability and ease of maintenance between fuelling operations to minimize overall incapability of the station due to F/H equipment unavailability. The access and maintainability are designed with the principle of reducing radiation exposure to the public operators and maintainers following ALARA philosophy. (As Low As Reasonably Achievable)

All equipment is designed to withstand ground motion earthquake conditions appropriate to the station site.

3.0 System Description

The fuel handling system is provided for reception and storage of new fuel, for fuel changing with the reactor at full power and for temporary storage of irradiated fuel.

For a typical single-unit CANDU 6 station, a self-contained fuel changing system is provided (Figure 1). For multi-unit stations such as Pickering `A' and Pickering `B' comprising four reactors each, dedicated fuel changing systems are provided for each unit. (Figure 2) However, the fuel transfer systems for the units are integrated such that irradiated fuel is transferred routinely via a conveyor to a central irradiated fuel storage bay (Figure 1), these are A.E.C.L. specific designs.

In other multi-unit stations such as Bruce A' Bruce B and Darlington A, a "shared" fuel handling system is adopted, whereby the fuel changing equipment comprising two fuelling machines (F/M) can service any of the reactors (Figure 3). The two F/Ms are mounted on a common trolley which can traverse a tunnel interconnecting the different reactor units. Irradiated fuel is therefore transported from each reactor, inside the F/M, to the fuel transfer port that is located at the common fuel storage bay. These stations are C.G.E. specific designs.

There are two variations in the basic design of the fuel changing system. The Pickering/CANDU 6 design was derived from Douglas Point. Rajasthan Atomic Power Station (RAPS) and Madras Atomic Power Station (MAPS) in India are nearly identical to Douglas Point. They all incorporate fuel separators on the F/M head to sense the axial position of the fuel bundles as they pass between the F/M and the fuel channel, to act as a fuel string stop, and to push the fuel into the magazine rotor stations in the F/M. During fuel changing, a constant bore passage is provided by a guide sleeve between the F/M magazine rotor station and the fuel channel pressure tube. Fuelling is carried out in the direction of fuel channel coolant flow.

Karachi Nuclear Power Project (KANUPP), Bruce and Darlington were derived from Nuclear Power Demonstration (NPD). With this concept, the fuel bundles are placed inside hollow cylindrical tubes called carriers that are placed in the F/M magazine rotor stations. During fuel changing, the fuel bundles remain in the carriers until they are discharged directly into the fuel channel pressure tube or the fuel transfer port. Fuelling is carried out against the fuel channel coolant flow. A latch mechanism that is incorporated at the downstream end of each fuel channel, just outboard of the pressure tube rolled joint, maintains the fuel string axial position in the pressure tube. The aforementioned separator assemblies that are required on the Pickering/CANDU 6 F/M are not used.

All of the aforementioned fuel changing systems utilize two identical F/Ms at each end of the fuel channel.

Figure 1

CANDU 6 Fuel Handling System

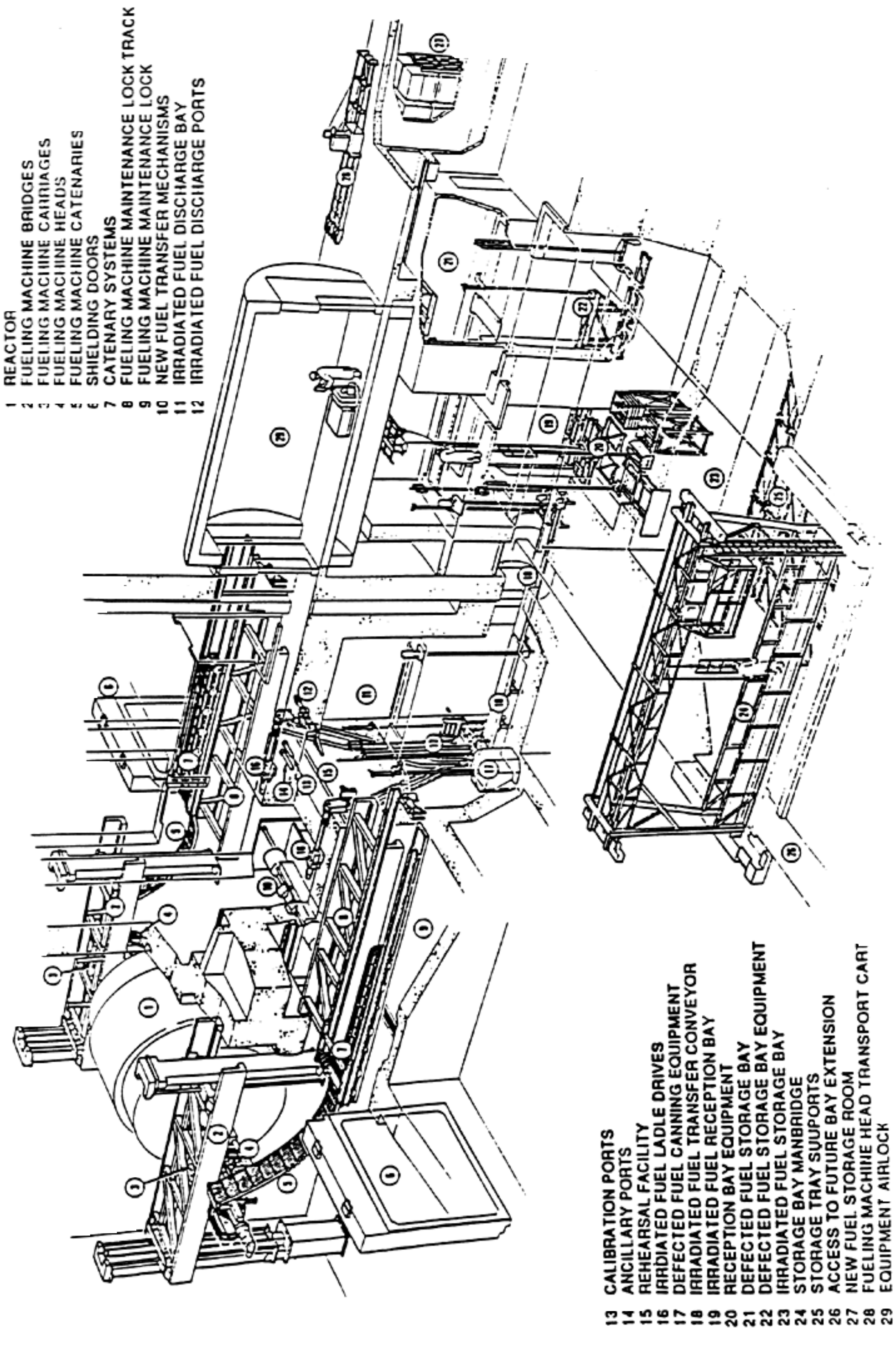


Figure 2
Pickering Fuel Handling System

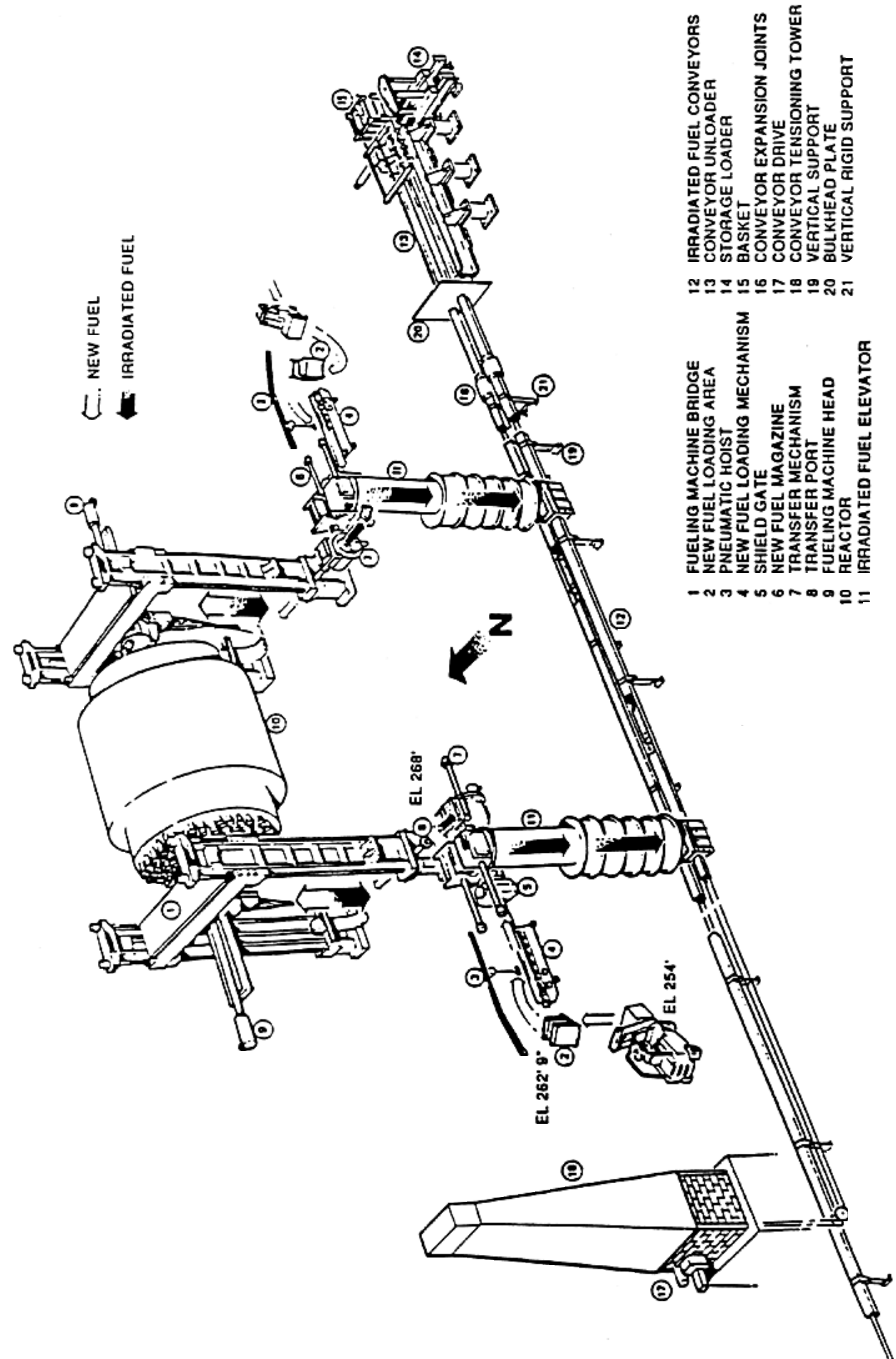
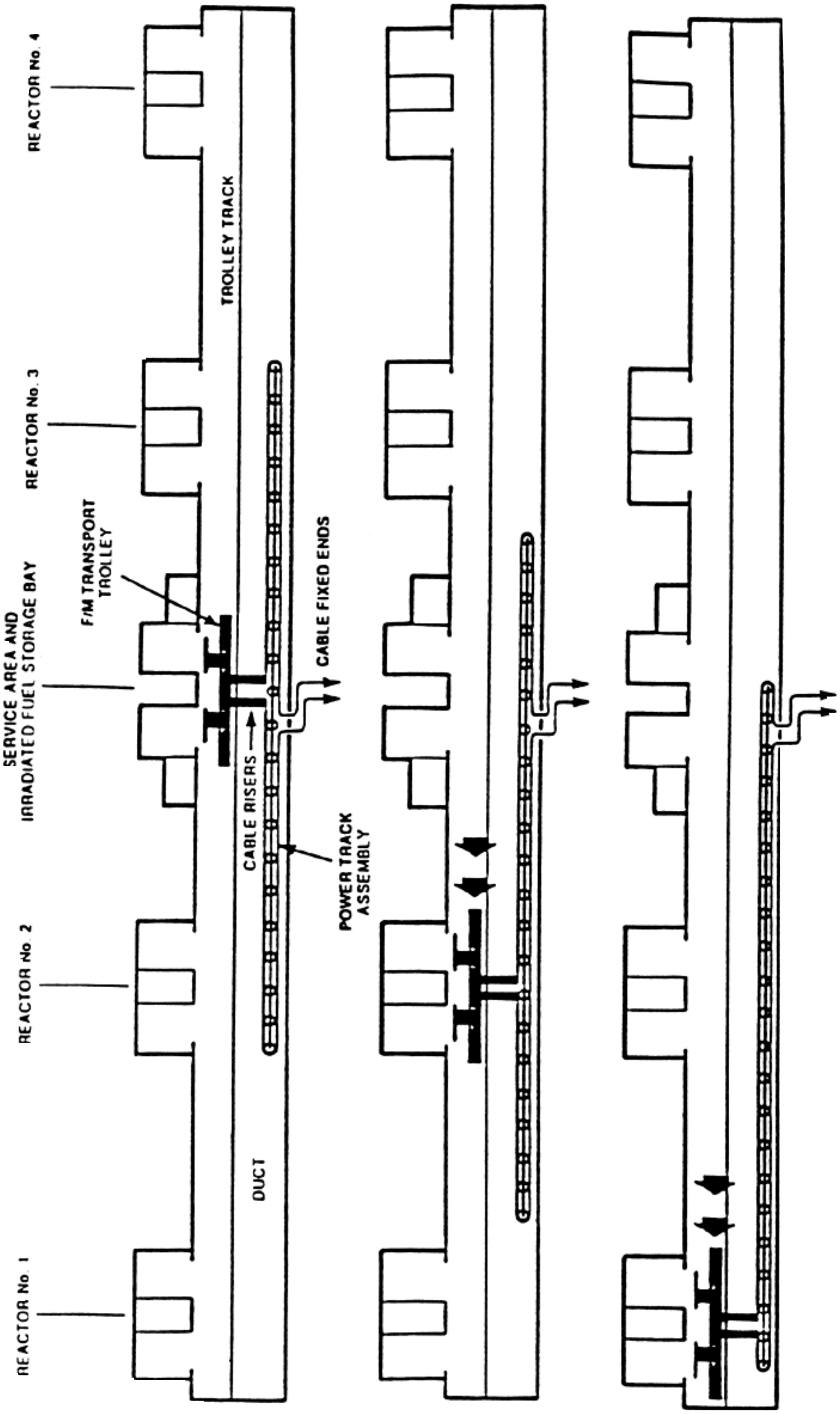


Figure 3
Bruce Multi-Unit Concept



3.1 On-power Fuelling History

"On power" fuelling at all stations over the years has met the primary objectives reasonably well, in economic terms, station incapability (lost generation) due to Fuel Handling unavailability on average has been less than 1% by comparison "off power" fuelled reactors of the world incur station incapability factors in excess of 10%.

The Candu fuelling systems have not been without problems and challenges. Constant development has been required to obtain acceptable reliability of water lubricated components such as ball screws and mechanical seals for fuelling machines. There have been numerous incidents of drive component failure and fuel damage requiring lengthy recovery operations, and development of recovery tooling and special procedures. Maintenance of Fuel Handling Systems accounts for a major portion of station radiation dose expenditure, due to severe contamination of the system internals where exposed to irradiated fuel and D_2O.

The CANDU 6 F/M head designed by A.E.C.L. is a derivative of the Pickering machine and much of the modular components are nearly identical. Hence, a common development program embraces the two systems. Although the Bruce/Darlington F/Ms designed by C.G.E. are different, there is commonality in the basic technology. This also applies to the different fuel transfer and storage systems that are tailored to meet the different station layout and performance requirements. CANDU 6 is a single unit station, whereas Pickering and Bruce/Darlington are multi-unit stations.

3.2 Description of CANDU On-power Fuelling

When the F/Ms are connected to the reactor fuel channel end fittings during fuelling, They become extensions of the primary heat transport system, when the channel closures are removed. Thus the F/M pressure boundary is designed in accordance with the ASME Pressure Vessel Code, Section III Class 1 requirements. While in transit from reactor to fuel transfer port, the F/M utilizes its own fuel cooling system.

The normal fuelling rate for a CANDU 6 reactor is 112 bundles or 14 fuel channels per week under normal steady state fuelling conditions. Fuelling normally consists of charging and discharging eight fuel bundles on each fuel channel visit. For other CANDU designs see Table 1.

To perform this fuelling requirement, the F/M basically comprises a snout, separators, magazine and a ram assembly. The snout allows the F/M to clamp on and seal to the reactor fuel channel end fitting. The magazine provides a number of rotor stations to house fuel bundles and the F/M and fuel channel hardware components, such as channel closures, shield plugs, ram adaptor and other items, that are required to carry out the fuelling operation. The ram assembly performs the mechanical actuation of the fuel channel and F/M hardware components. Two of the ram motions are provided by water lubricated ballscrew mechanisms that are supported on water lubricated anti-friction bearings. The ram and magazine drive shafts penetrate the pressure boundary and are driven by oil hydraulic motors. A third ram incorporates a water-actuated telescopic mechanism.

The separator assemblies monitor the fuel string location during fuel changing, and can hold back the fuel string against the hydraulic drag of the coolant in the fuel channel during fuel bundle separation and magazine rotor indexing.

The F/M and its support system must be designed to meet the seismic requirements pertaining to each reactor site. This is to ensure the integrity of the pressure retaining components during a seismic event. Analysis covers both the attached and unattached cases, relative to the reactor fuel channel. Some of the analytical models have been validated by actual tests in the engineering test facilities. The equipment is designed to withstand a seismic event without fission product release but not necessarily to remain fully operational except for maintaining fuel cooling capability.

Used fuel is placed in water-filled fuel storage bays. As the fuel storage bays become full, used fuel may then be placed in interim dry storage at the site after a cooling period of at least five years.

Table 1

CANDU Fuelling Rates

CANDU Station	Net Output per Reactor MW(e)	Normal Steady State Fuelling Rates (Fuel Bundles/ Week/Reactor/Unit)
Pickering A and B	508	80
CANDU 6 for: Gentilly-2 Point Lepreau Cordoba Wolsong	600	112
Bruce A and B	750	161
Darlington A	850	161

While Pickering has two fuelling machines dedicated to each reactor, Bruce/ Darlington have common sets of two F/M's on a trolley that can operate on different reactors. Two parallel trolley tracks and 3 trolleys serve 4 units, intending to give greater operational flexibility and less redundancy of capital equipment. There is no common agreement on which system is superior.

3.2.1 Fuel Bundles

Fuel is fabricated from uranium oxide that is sintered into pellets that are sealed inside zircalloy tubes. Generally, 37 of these elements are grouped together to make up a fuel bundle that is 102 mm diameter and 495 mm long (Figure 4). These elements are held together by end plates. Small pads maintain correct inter-element spacing and bearing pads that support the fuel bundle assembly in the fuel channel are brazed to the elements.

The fuel handling system is designed to limit axial compressive loads to 18 kN on irradiated fuel bundles during normal operation. Radial loads are limited to values equal to the normal weight of the fuel bundles.

New CANDU fuel bundles made from natural uranium are essentially non-radioactive and can be handled by personnel without protective shielding. Once the fuel bundles have been irradiated, handling must be carried out remotely with automated, remotely operated machines or with adequate shielding if handled with manually operated tools.

3.2.2 Fuel Channel Hardware

The commercial CANDU-PHWRs use horizontal fuel channels arranged in a square lattice grid.

Each fuel channel is composed of a zircalloy pressure tube that houses the fuel bundles in the reactor. The pressure tubes are provided with stainless steel end fitting extensions at each end (Figure 5). These tubes are mechanically connected using rolled joints. Within the end fittings at the inboard end, removal shield plugs (Figure 6) are provided. The ends of the fuel channels are provided with removable channel closures (Figures 7 and 8).

The fuel bundles in the fuel channel are not mechanically attached and are held together by the coolant flow hydraulic drag forces. In the Pickering/CANDU 6 design, there are twelve fuel bundles in each fuel channel and the fuel string rests against the downstream shield plug.

In the Bruce/Darlington design, there are 13 fuel bundles in each fuel channel and the fuel column is held by the fuel latch incorporated at the downstream end, located approximately one-half bundle length outboard of the calandria. Due to practical reasons, it is not possible to provide a mechanical fuel latch in the zircalloy pressure tube, precisely at the boundary of the calandria.

Figure 4
Fuel Bundle

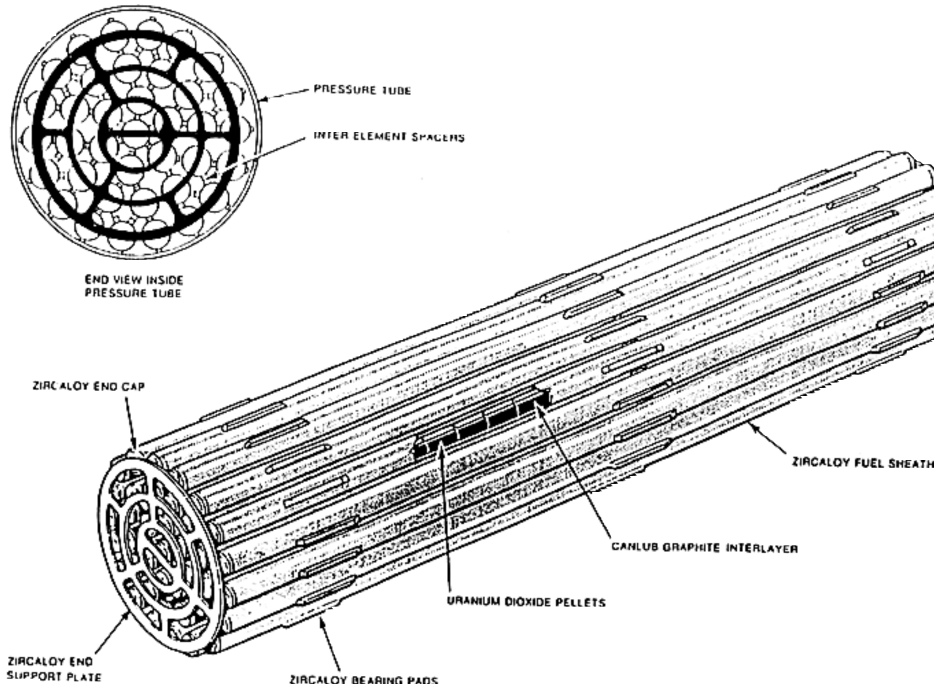


Figure 5
Pickering/CANDU 6 Fuel Channel

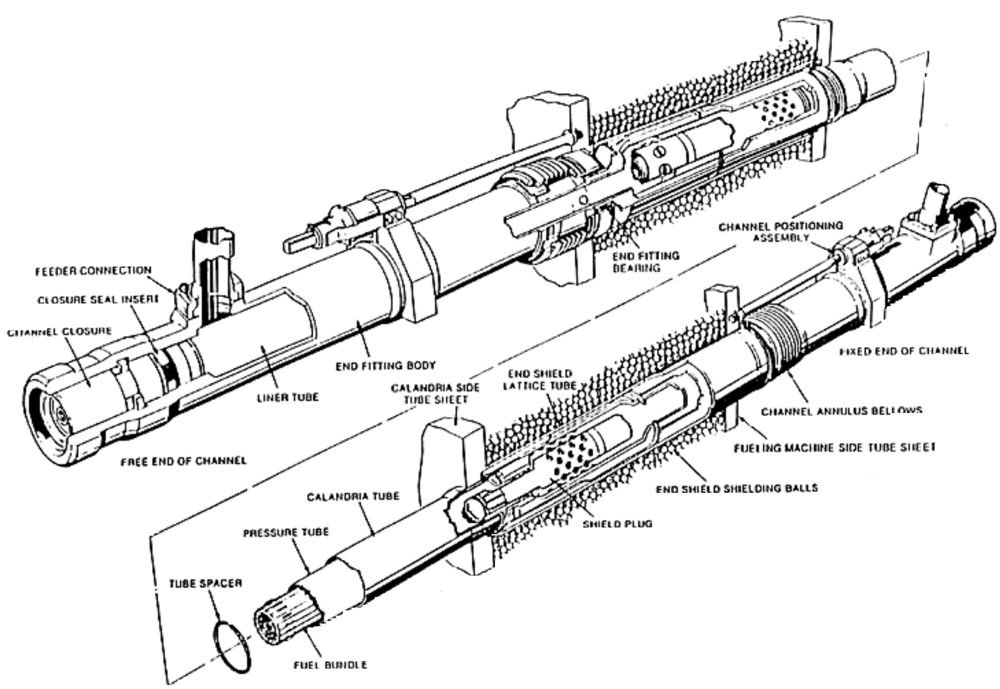


Figure 6
Pickering/CANDU 6 Shield Plug

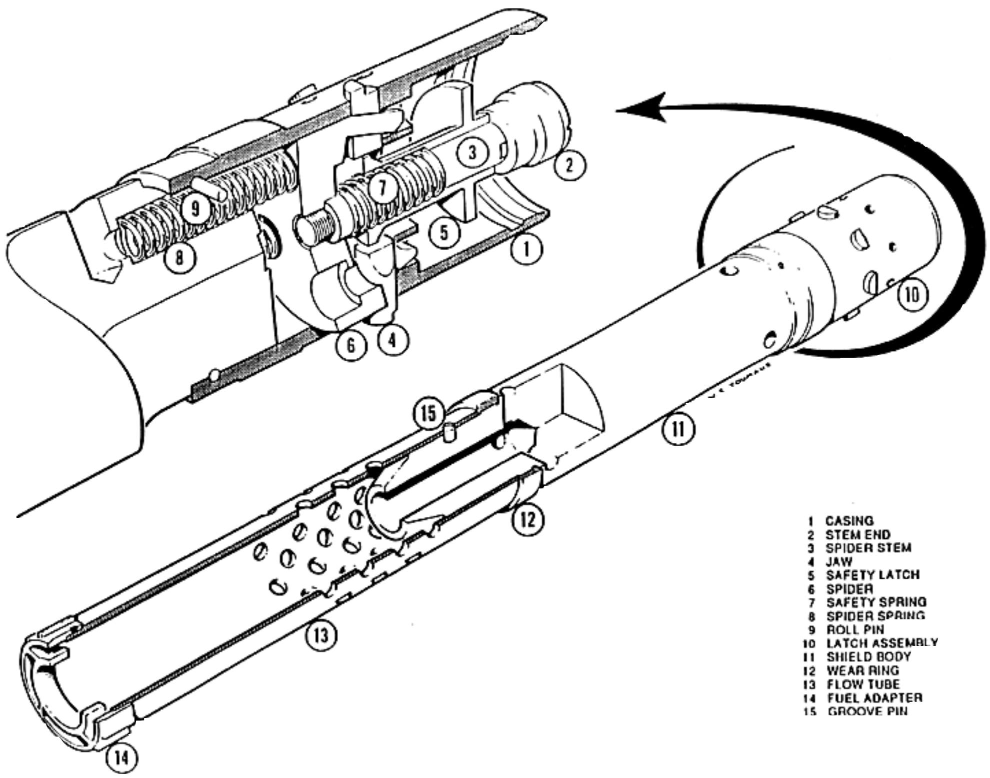


Figure 7
Pickering/CANDU 6 Channel Closure

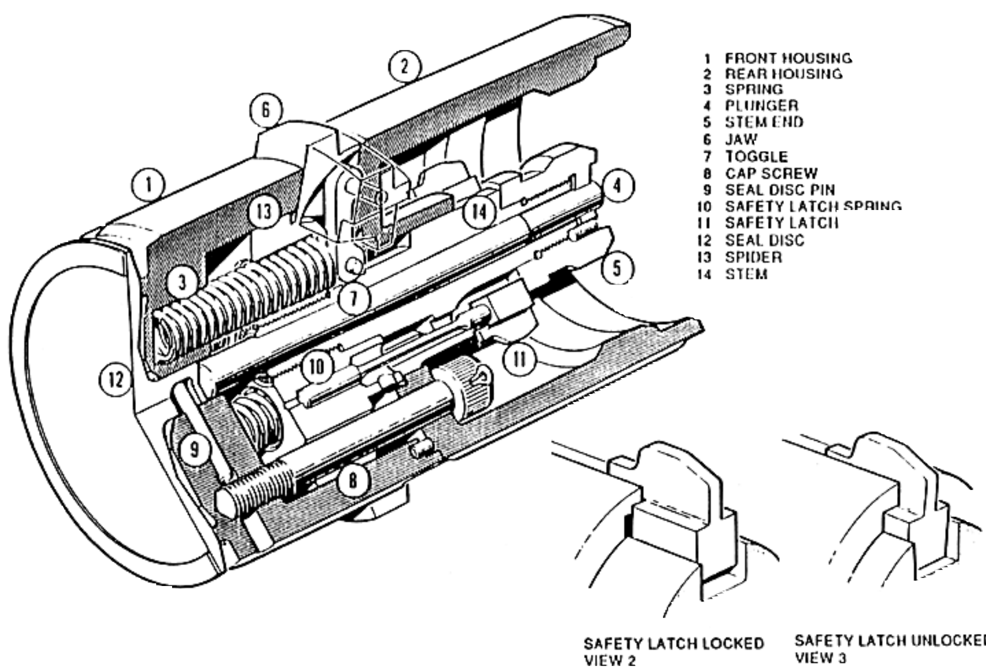
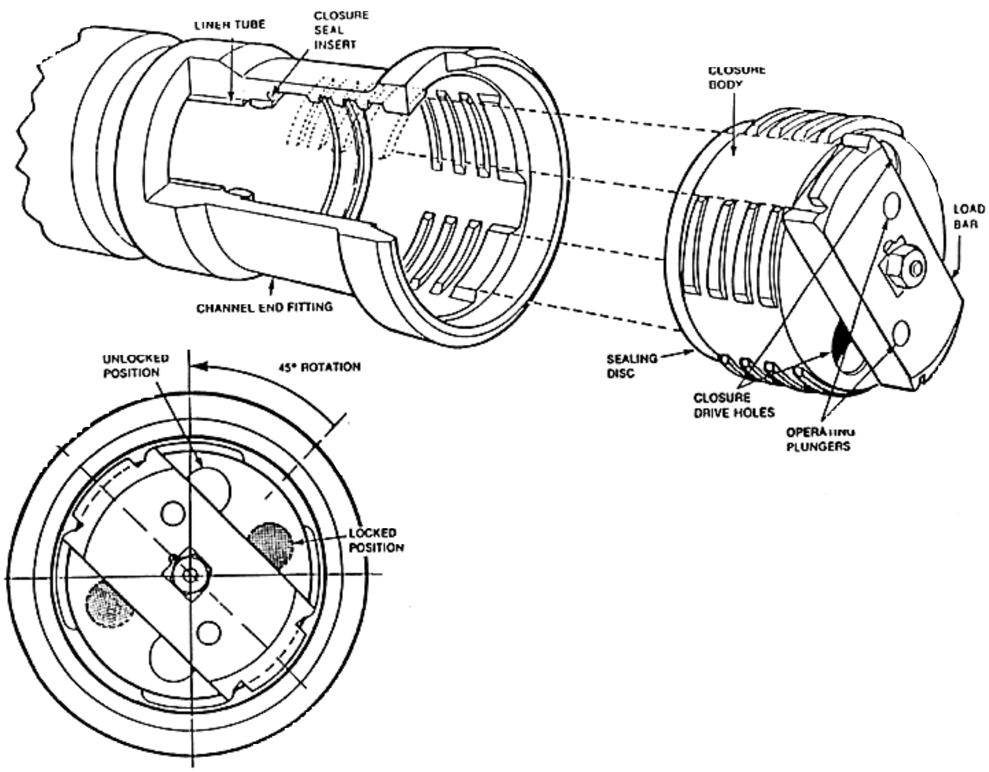


Figure 8
Bruce/Darlington Channel Closure



3.3 Process Description

The fuel handling system is made up of two fuelling machines, support structures and associated transport equipment supporting tracks. Each F/M has access to corresponding new fuel loading, irradiated fuel unloading and F/M calibration facilities. The overall system contains an irradiated fuel transfer system, plus all associated auxiliaries, power supplies and control systems.

The fuel handling system can be divided into three interrelated systems: new fuel transfer, fuel changing, and irradiated fuel transfer. Figure 9 shows the typical movement of fuel, from the new fuel storage area through the reactor to the irradiated fuel storage bay.

The fuel arrives at the site in pallets. Up to a nine-month supply is stored in the service building new fuel storage area. When required for station operation, the pallets are transferred to the new fuel loading area. There, the fuel bundles are uncrated, inspected and loaded into a F/M via the new fuel port and transfer mechanism. Once loaded with new fuel, the F/M traverses to the reactor face and connects to any one of the fuel channels. A second empty F/M connects to the same fuel channel at the other end of the reactor. Automatic fuel changing operations then commence with removal of fuel channel closures and shield plug and new fuel bundles being loaded at one end while the equivalent number of irradiated fuel bundles is received by the other F/M. Bundle movement is controlled by the two F/Ms, assisted by the coolant flow inside the channel. As the flow in each alternate channel is reversed for reasons of reactor symmetry, the F/M must be capable of operating bi-directionally, i.e., the upstream F/M can perform the functions of the downstream machine and vice versa.

In a typical eight bundle fuelling sequence for the CANDU 6 design, as shown in Figure 10, the following fuel movements inside the fuel channel can be identified:

- Eight new bundles are inserted, two bundles at a time, from the upstream end.
- The whole 20 bundle fuel column (twelve old bundles plus eight new bundles) is moved towards the downstream end.
- Eight old bundles are discharged, two at a time, from the downstream end.
- The remaining twelve bundle fuel column is moved back to the correct in-reactor position.

Figure 9
CANDU 6 Fuel Handling Sequence

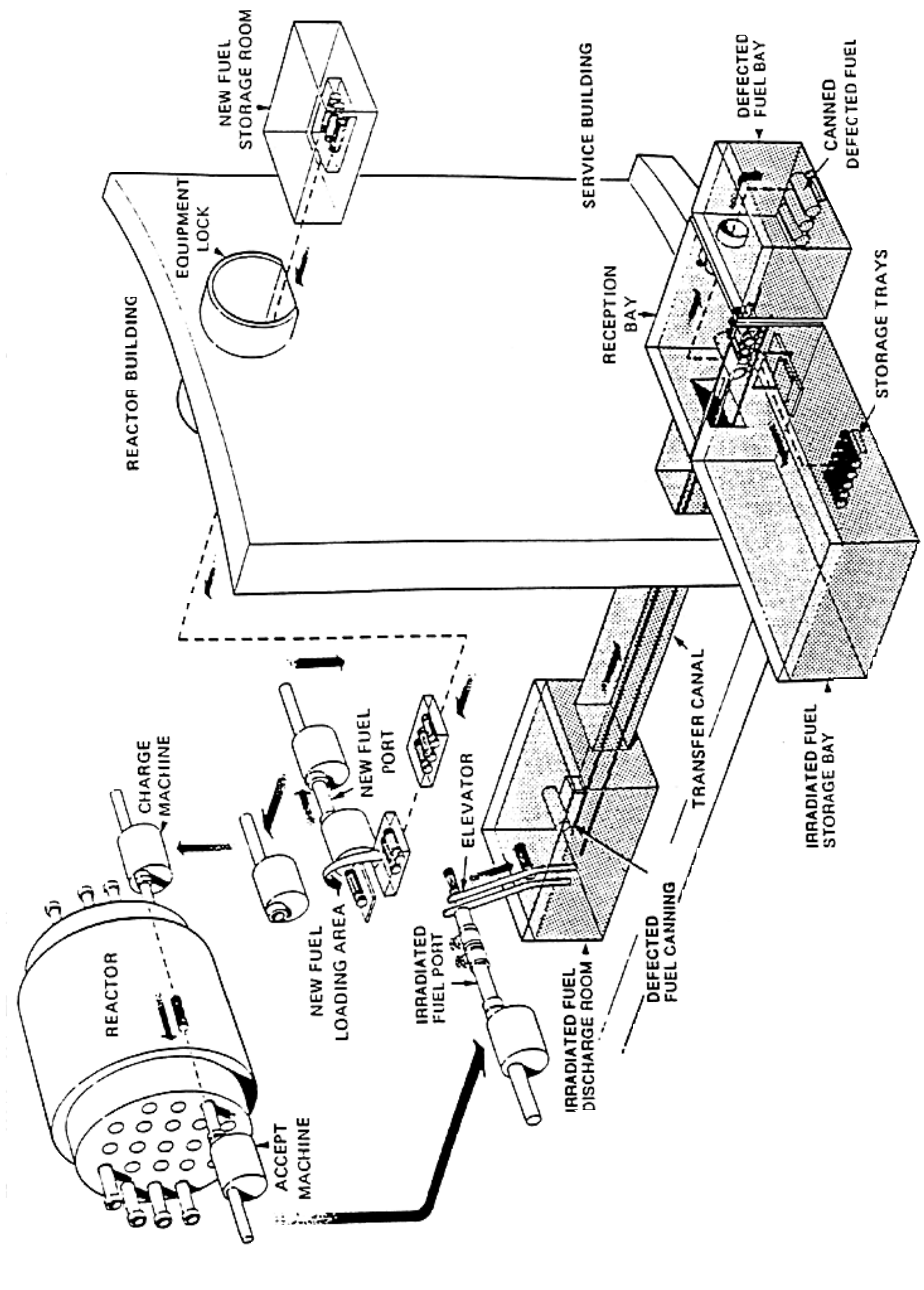
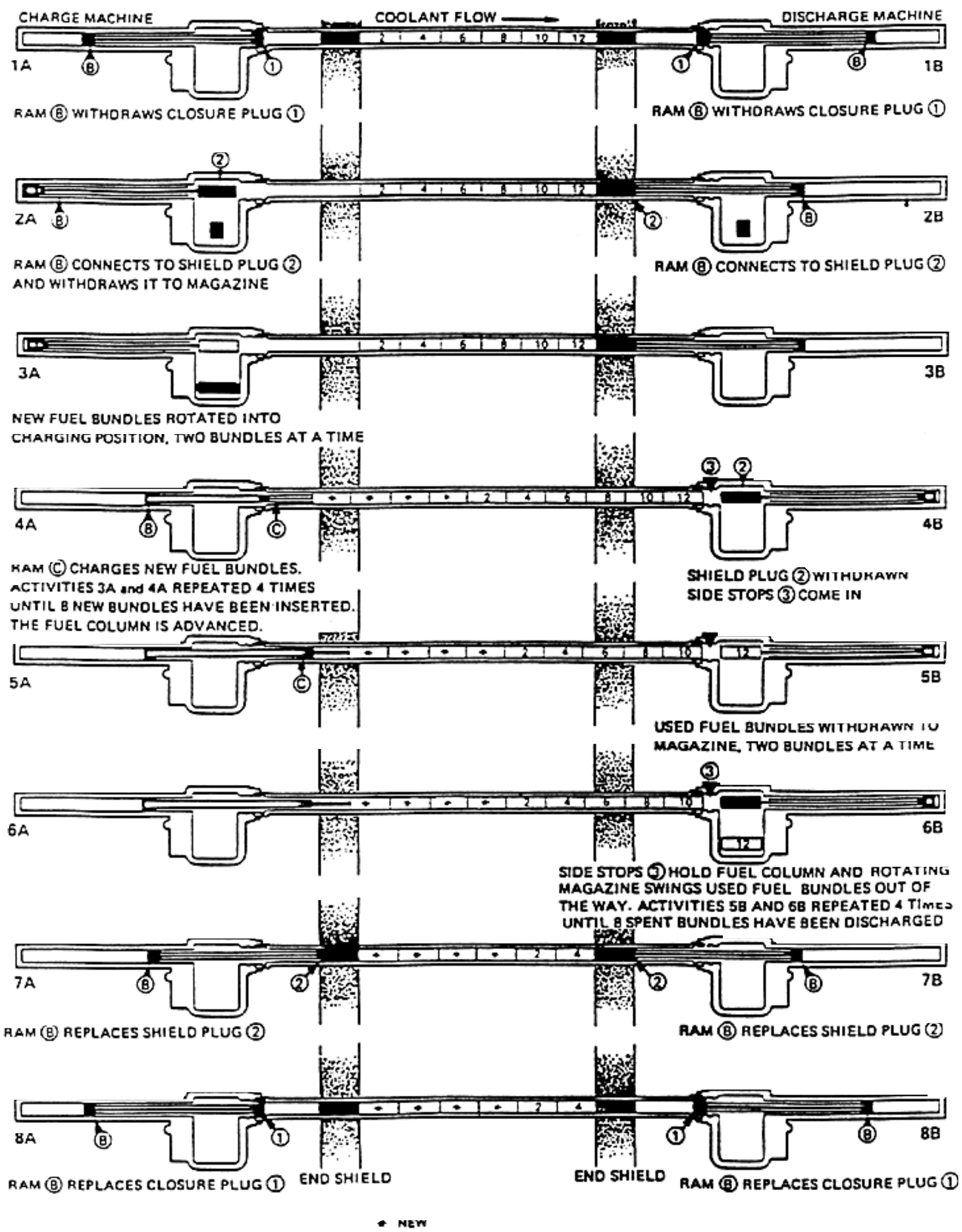


Figure 10
Pickering/CANDU 6 8-Bundle Fueling Sequence



On completion of the fuel changing operation, the downstream F/M traverses to the irradiated fuel port and unloads the irradiated fuel bundles to the storage bay via the irradiated fuel port elevator and conveyor. The irradiated fuel bundles are then placed in storage trays.

Once the F/M has unloaded the irradiated fuel, it goes to the new fuel port to pick up new fuel bundles, and traverses back to the reactor face to become the upstream F/M for the next channel to be fuelled. The other F/M, having remained at the reactor face, will also home onto the next channel and perform the functions required at the downstream end.

Articulated TV cameras, one in each F/M vault at the reactor face and one at the irradiated fuel unloading bay are provided to permit monitoring of critical operations and during breakdown conditions.

3.3.1 New Fuel Storage and Handling

New fuel is received in the central storage room that is located in the service building. This room accommodates the normal station inventory and temporarily stores the fuel for the initial reactor core load.

When required, new fuel pallets are transferred to the new fuel loading area. Here, the bundles are identified and are loaded manually into the magazines of the new fuel loaders. Ports from the loaders penetrate into the reactor vault where they are terminated by end fittings, similar to those on the reactor fuel channels.

Interlocked valves are provided at each end of the ports to prevent tritium from the F/M entering into the new fuel loading area, and to maintain atmospheric separation within the reactor building.

3.3.2 Fuel Changing

The on-power fuel changing equipment consists of two identical F/Ms at each end of the reactor, suspended on a carriage from tracks on a bridge or a trolley that extends the full length of the shielded reactor vault. Vertical and horizontal traverse of the F/M is provided to allow access to all the fuel channel end fittings. Powered shielding doors separate the reactor vault from the maintenance lock and, when closed, allow access to the F/M in the maintenance lock while the reactor is operating.

While in the maintenance lock, the F/Ms also have access to the new fuel ports to receive new fuel, to the service ports for calibration or service, or to the rehearsal facility.

Fuelling operations are performed with the equipment under remote automatic computer control. The shielding doors are opened and the two F/Ms travel along the tracks at each face of the reactor and are positioned on each end of the selected fuel channel.

Both F/Ms move forward to home and lock onto the fuel channel. After a leak test on the clamp seal, each F/M, which is filled with heavy water, is then pressurized to match the heat transport system pressure conditions. The temperature of the heavy water in the F/M is maintained at a lower temperature than that in the fuel channel and a steady inflow into the fuel channel is maintained to thermally isolate the F/M from the higher temperature of the heat transport system. The F/Ms remove the channel closures and store them in the F/M magazines. Guide sleeves are installed and the shield plugs are then removed from the fuel channel and new fuel is inserted at one end while irradiated fuel is discharged from the other end of the fuel channel.

Two fuel bundles can be inserted from each magazine position containing new fuel in the F/M head. Four to eight new fuel bundles are generally inserted on each visit, thus replacing four to eight of the fuel bundles in the fuel channel. Either F/M can load or accept fuel, depending on the direction of flow in the particular fuel channel being serviced.

When the required number of fuel bundles has been inserted, the shield plugs and channel closures are replaced and the closures are leak tested by the F/M. The two F/Ms then traverse to the irradiated fuel ports where irradiated fuel is discharged.

With four bundle shift fuelling, the F/M can fuel one channel and then fuel a second one, before returning to the irradiated fuel ports. For a CANDU 6 reactor, fuelling is required on about fourteen fuel channels per week, with eight fuel bundles being discharged at each visit to the reactor. The fuelling frequency varies proportionately with the size of the reactor and the number of fuel bundles exchanged. Table 1 gives typical fuelling requirements in "fuel bundles" for different reactors.

There are two distinct fuel channel designs and fuel changing processes. The fuel bundles are separated by fuel separators built into the F/M in one design, and by integral fuel latches in the fuel channel by the other.

3.3.2.1 Fuel Latch Method

The fuel latch method was originated with NPD and adopted in Bruce/Darlington. The characteristic features of this design are:

- The fuel channel contains 13 fuel bundles.
- Fuel latches that are located at the downstream end of each fuel channel restrain the fuel column against the hydraulic drag of the coolant flow.
- Channel closures and shield plugs incorporate a breech-type locking feature that requires rotary actuation. Correct radial orientation is mandatory.
- Movement of fuel bundles in pairs between the F/M and the fuel channel is carried out in fuel carriers.
- The fuel channel end fitting bore is larger than that of the fuel bundle to accommodate the fuel carrier.
- Fuelling is carried out against the cooling flow. When the fuel bundle is

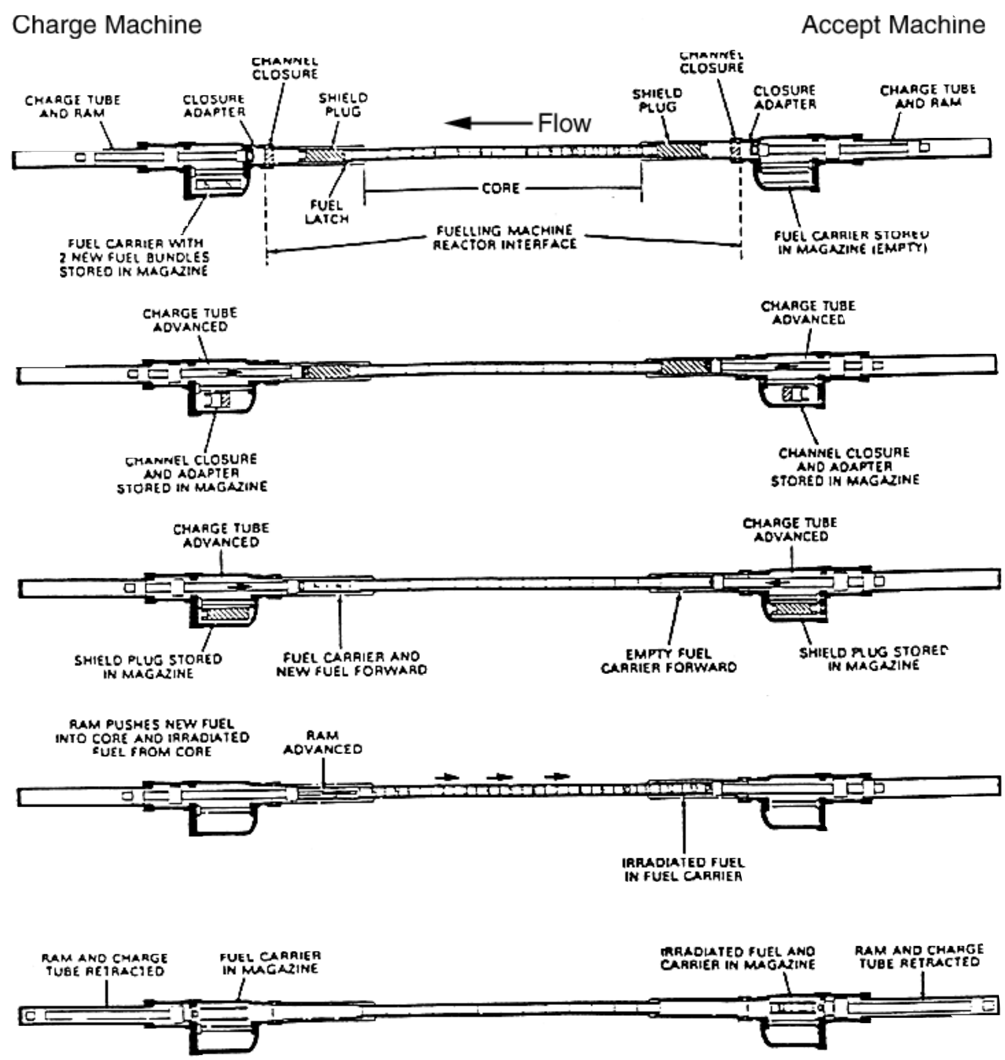
discharged from the fuel carrier into the pressure tube, it passes through the spring-loaded fuel latch which is incorporated in the end fitting liner tube, to prevent reverse motion.

By comparison to Pickering/Candu 6, it is simpler in concept, with a minimum of fuel bundle travel. During fuelling there is always 13 bundles in core, creating only positive reactivity change with only minor channel flow variance.

The fuelling sequence for Bruce/Darlington is depicted in Figure 11.

Figure 11

Bruce/Darlington NGS 2 Bundle Fuelling Sequence



3.3.2.2 Fuel Separator Method

The separator method that is adopted for the Pickering/CANDU 6 design originated with Douglas Point. The characteristic features of this design are:

- The fuel channel contains twelve fuel bundles, all located within the reactor core.
- Shield plugs and channel closures feature radially extending jaws that are actuated by axial motions. Radial orientation is not important.
- The fuel column is restrained by the downstream shield plug jaw mechanism.
- During fuelling, a constant bore passage is provided between the pressure tube and F/M magazine station.
- The fuel bundle position is sensed by the separator assemblies on the F/M that also restrain the fuel column against hydraulic drag during fuel changing, separate fuel bundles and push pairs of fuel bundles into the F/M magazine station.
- Fuelling is carried out in the direction of flow.

In the high flow channels, there is sufficient flow to allow the fuel column to move without assistance. Figure 10 depicts the fuelling sequence. In the original Douglas Point design, the F/M ram was used to push the fuel column in the outer low flow channels, during fuelling. However, this resulted in activation of the F/M ram components that made it difficult to service the F/M. Consequently, a free piston called the Flow Assist Ram Extension (FARE) tool was developed for Pickering/CANDU 6 to create the impedance necessary to move the fuel column.

3.3.2.3 Fuel Sequence Detail for Candu 6

Figure 10 demonstrates an 8 bundle sequence, different sequences are used depending upon core location and reactivity state. Example:

1. **2 bundle shuffle** - used in a new core taking 2 bundles from the downstream end and reinserting the low burn-up bundles in the upstream end of adjacent channels. This improves burn-up in the core that has not reached equilibrium state.
2. **4 or 8 bundle change** - frequently used in the central high flux regions. 4 bundle change provides better burn-up than the 8 bundles which is also sometimes too large a positive reactivity boost for zone control.
3. **10 bundle change** - used only in the outer core lower flux regions, to minimize channel visits with a reasonable reactivity gain.

Refer to figure 10 for the following sequence:

After homing and clamping both fuelling machines to a reactor channel and pressurizing the F/M's, both closure plugs are removed. This now allows ambient temperature D_2O from each F/M to flow into the channel 2.27 *l*/s, this is ensured by setting the magazine pressure higher than the reference channel pressure. The injected cool D_2O depresses the channel outlet temperature by 10 to 20°C. It is operating policy to trend and monitor the outlet R.T.D. throughout the fuelling sequence. Very significant information is provided by the temperature profile, firstly it provides confirmation that the correct channel has been selected, additionally since channel power decreases about 50% through the sequence and channel flow resistance changes "pattern" is generated. Operator monitoring of this "pattern" will give confirmation the sequence is progressing normally. Deviation from pattern could alert the operator to a process problem or channel blockage. Channel high temperature alarms are used to detect major channel blockage.

The second picture in Figure 10 shows F/M-A removes the shield plug and stores it in the magazine while F/M-B latches on to the shield plug but does not move it since it is holding the string of fuel against the channel flow.

The third picture in Figure 10 shows new fuel is added 2 bundles at a time into the position vacated by the shield plug.

The fourth picture in Figure 10 shows the 2 F/M's must now operate in unison with the downstream F/M retracting the shield plug into its magazine with the string of new fuel pushed tightly to follow the original string free acceleration in channel flow must be avoided because resulting collision of bundles would create major fuel bundle damage. Hydraulic fuel separators (3) hold the string while the shield plug is stored in the magazine.

The sequence shown in Figure 10 is for centre core region where channel flow is high enough for "flow assist" fuelling so that the "c" ram of the upstream F/M does not have to follow the fuel string beyond the shield plug position, avoiding ram activation. In the outer core regions channel flow is too low to completely move the string into the receiving F/M so a flow assist ram extension (FARE) is inserted after the last pair of new bundles. The FARE tool is the size and shape of a shield plug fabricated of zircalloy (for neutron economy) with a flow restriction orifice which generates sufficient ram force to push the fuel string.

Pictures 5 and 6 of Figure 10 show 8 bundles, 2 at a time are separated off the string and stored in the magazine. It is not shown by the diagram that after removal of 8 bundles only 6 bundles reside in core reducing channel power by ≈ 50% with lowest hydraulic resistance, channel flow is maximum, so at this point the largest channel outlet temperature depression occurs.

Pictures 7 and 8 of Figure 10 show after removing the 8 bundles the fuel string is pushed back to normal core position with the original 4 bundles from the upstream end now residing in the downstream end. The upstream F/M retrieves the "FARE" tool (if used) and both F/Ms close the channel. The operator will now see zone control response to the sudden addition of about 0.2 mk. Minor boiler level control change may also be seen due to a shift in zonal power.

3.3.3 Irradiated Fuel Transfer

Two distinct methods are used to discharge irradiated fuel from the heavy water environment in the F/M to a light water storage bay that is located outside reactor containment.

Method 1

Wet discharge is featured at Pickering. Irradiated fuel is discharged from the F/M through the discharge port into a heavy water-filled transfer mechanism. The heavy water level is lowered in the transfer mechanism and the irradiated fuel is discharged dry onto an elevator ladder that interconnects with a light water immersed conveyor system that transfers the irradiated fuel to the storage bay. The light water column in the elevator shaft forms a part of the reactor building containment boundary (Figure 2). The leg of water in the elevator is high enough to support a reactor building positive pressure of about 42 KPa(g).

Method 2

In CANDU 6, the F/M clamps onto the fuel transfer port and forms a leak tight joint and the level of heavy water in the F/M is lowered to below the snout level. As the valve on the fuel transfer port is open to the discharge bay atmosphere that is outside the reactor containment boundary, the F/M pressure boundary becomes a part of the reactor containment boundary. Irradiated fuel is discharged dry through the fuel transfer port onto the ladle on the elevator and lowered into the discharge bay that is interconnected to the storage bay through the reception bay (Figure 1) by a transfer conveyor.

Method 2 (alternative)

In the Bruce/Darlington design, the fuel transfer port is connected to an air hood that is submerged in the storage bay. When the F/M clamps onto the fuel transfer port, the level of heavy water is already below that of the snout and irradiated fuel is discharged dry onto a ladle in the air hood. After the irradiated fuel is placed on the ladle, it swings out from the air hood into the light water environment of the storage bay. The water surface in the air hood therefore forms a part of the reactor building containment boundary.

3.3.3.1 Wet Storage

Storage of used CANDU fuel in water-filled, reinforced concrete bays is provided at each site. Epoxy has been used extensively as a liner material in the bays. Stainless steel liners have also been used. The bay is considered as interim storage. Initial bay storage capacity is mostly based on ten years output at 80% minimum reactor operating capacity, plus one core load. An area is set aside in the storage bay for underwater filling of transportation casks.

Any defected fuel is sealed in cans with provision to vent noble gases and are stored in a separate area in the bay. The present achieved defected fuel rate is less than 0.06%. Defected bundles are later decanned and loaded under water into holding cans for long-term storage. An underwater fuel examination station is also possible to establish within the bay.

At Pickering, used fuel is stored in rectangular modules containing 96 used fuel bundles. The same modules are also used at Darlington. The modules are stacked in frames equipped with expanded metal mesh along the sides for seismic restraint and International Atomic Energy Agency (IAEA) safeguarding purposes.

At the CANDU 6 stations, used fuel is stored in trays contained 24 fuel bundles arranged in two rows, in a single layer. Trays are stacked, approximately 19 high and arranged in groups of two or four. Each group of stacks is provided with a cover that is held in place by vertical rods that integrate and tie the stacks together to resist possible seismic loads and for IAEA safeguarding.

At Bruce, used fuel is also stored in single layer trays that are stacked one on top of the other, but these trays are different from those used on CANDU 6, and are placed in stacking frames similar to Pickering for seismic restraint and safeguarding purposes.

3.3.3.2 Dry Storage

Subsequent dry storage of used CANDU fuel five years after discharge from a reactor is an AECB approved system of used fuel storage. The future size of the storage bays on CANDU stations now can be reduced to as little as six years capacity, including capacity for a full core load.

AECL has decommissioned Gentilly-1, Douglas Point and NPD reactors and uses concrete canisters (Figure 12) located outdoors at the sites, for interim dry storage. Pickering will be using an Ontario Hydro designed flask that will be directly loaded in the fuel bay with the 96 bundles storage modules.

Used fuel is transferred from the trays into cylindrical containers in the bay. The filled container is then raised into a dry shielded work station that is installed at the edge of the pool and a lid is seal welded, using remote handling equipment (Figure 13).

The sealed container is transported in a flask to the outdoor concrete canister for interim storage until it will be disposed permanently off-site at a future date. The minimum life expectancy of these canisters is 50 years.

Figure 12
Interim Dry Storage Concrete Canister

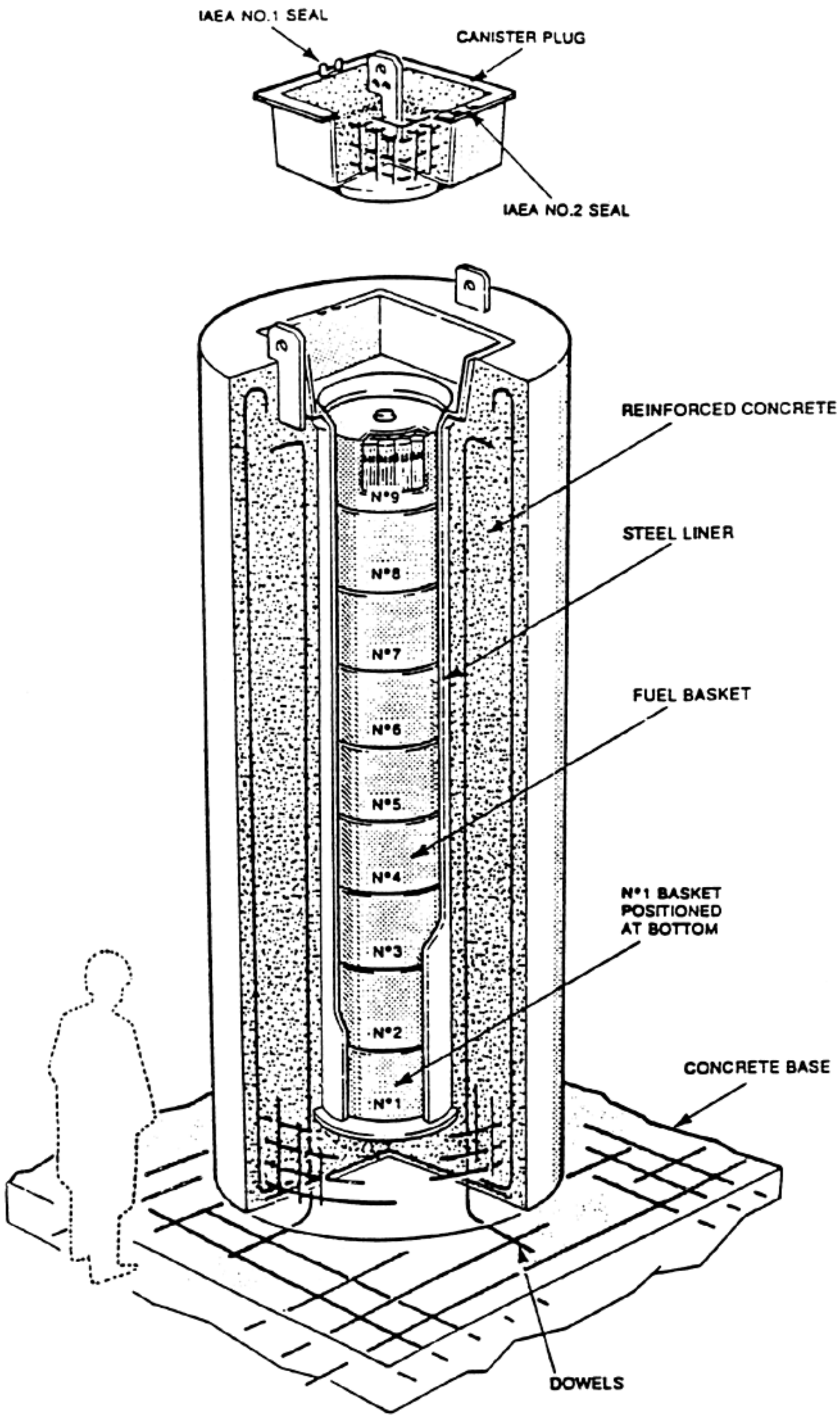
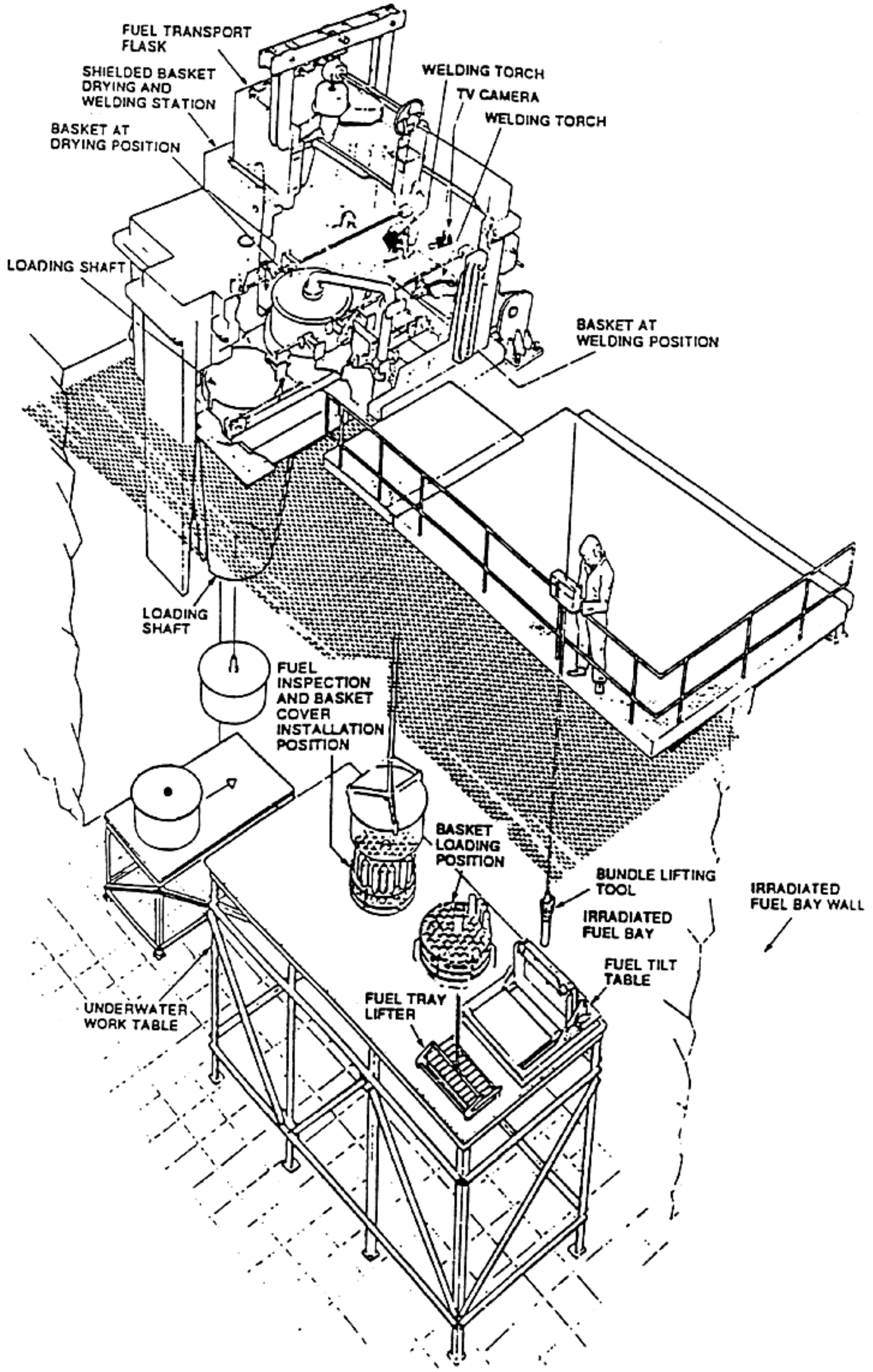


Figure 13
Dry Storage Basket Loading Equipment



4.0 Major Equipment/Components

4.1 Fuelling Machine

The F/Ms comprise mechanisms for manipulating the fuel bundles, both at the reactor face and at the new fuel and irradiated fuel ports (Figures 14 and 15). The mechanisms for each F/M include a snout assembly that clamps and pressure-seals the F/M to the fuel channel end fitting; a magazine that accommodates the fuel bundles to be inserted or removed, together with shield plugs, channel closures and F/M hardware; rams that remove and replace channel closures, and F/M hardware; and in the case of the CANDU 6 design, separators that separate and hold the fuel bundles during magazine storage.

4.1.1 Snout

The snout provides a pressure-tight joint between the F/M and the fuel channel or irradiated fuel port end fittings. In the CANDU 6 design, the snout is driven by oil hydraulic pistons that convert rotary motion into an axial clamping motion using a screw and gear combination. Sealing between the F/M snout and the fuel channel end fitting is achieved by a self-energizing bellows type metallic seal.

The Bruce/Darlington snout features a design that is similar to a double-hinged Grayloc clamp, incorporating a Grayloc seal ring. Actuation is provided by the central gearbox through a gear drive.

4.1.2 Magazine Assembly

The magazine housing assembly is a pressure vessel consisting of two main forgings. The CANDU 6 design uses a Grayloc clamp and seal ring to hold the housing together. A bolted flange joint with an `O' ring seal is featured on the Bruce/Darlington design.

The magazine front opening and mounting faces are provided for the snout, while the rear opening is provided for attaching the ram assembly.

For the CANDU 6 design, openings and mounting faces are provided for two separator assemblies, just behind the snout.

There is only about 5 cm of steel in the magazine housing to act as shielding in either design, thus with irradiated fuel in the magazine, radiation fields from the fuelling machine are very high and the machines cannot be approached by personnel. Any recovery operations from fuelling machine failure must be done with remote tooling.

The magazine rotor assembly houses the fuel bundles, and the F/M/fuel channel hardware. The magazine rotor is supported by a shaft mounted on D_2O lubricated anti-friction ball bearings. The magazine specifically houses five fuel stations (10 bundles), two shield plug stations, one snout plug station, two closure plug stations, one ram adaptor station and one guide sleeve station.

Indexing of the magazine rotor is achieved by an externally mounted commercial (Ferguson) mechanical indexing device. The drive is provided by an oil hydraulic motor and an electrically driven central gearbox for the Pickering/CANDU 6 and Bruce/Darlington designs respectively.

4.1.3 Ram Assembly

In the CANDU 6 design, the ram assembly is capable of three independently controlled axial motions; two of the motions are provided by mechanical actuation of ballscrews supported on anti-friction ball bearings in a water environment. The third motion is supplied by a telescopic heavy water actuated hydraulic ram. The ballscrews are driven by externally mounted oil hydraulic motors via shaft penetrations through the pressure boundary. The mechanical "B" ram can exert up to 26,690 newtons axial force to operate closure and shield plugs, the hydraulic "C" ram can exert up to 8,896.5 newtons to push fuel.

In the Bruce/Darlington design, the ram is also capable of three independent motions. Two of the motions are ballscrew driven axial motions, however, the third is a rotary motion provided by a gear drive, all operating in a water environment. The central gearbox that is equipped with two ac electric induction motors provides the actuation. The design has no ram motion beyond the fuel carrier position as compared to the Candu 6 design that can advance the hydraulic ram to the centre of the core if required.

Figure 14
Pickering/CANDU 6 Fueling Machine Head

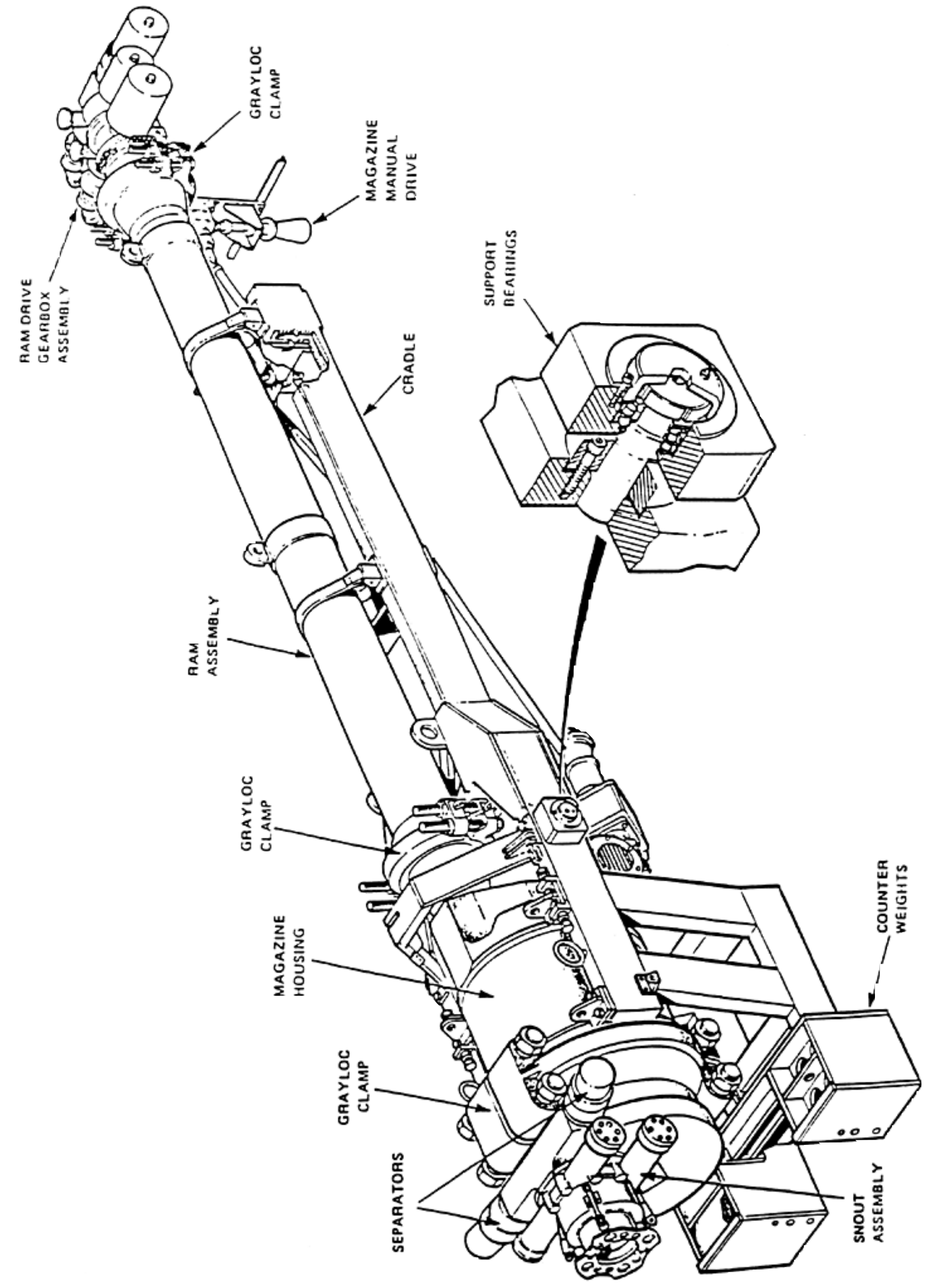
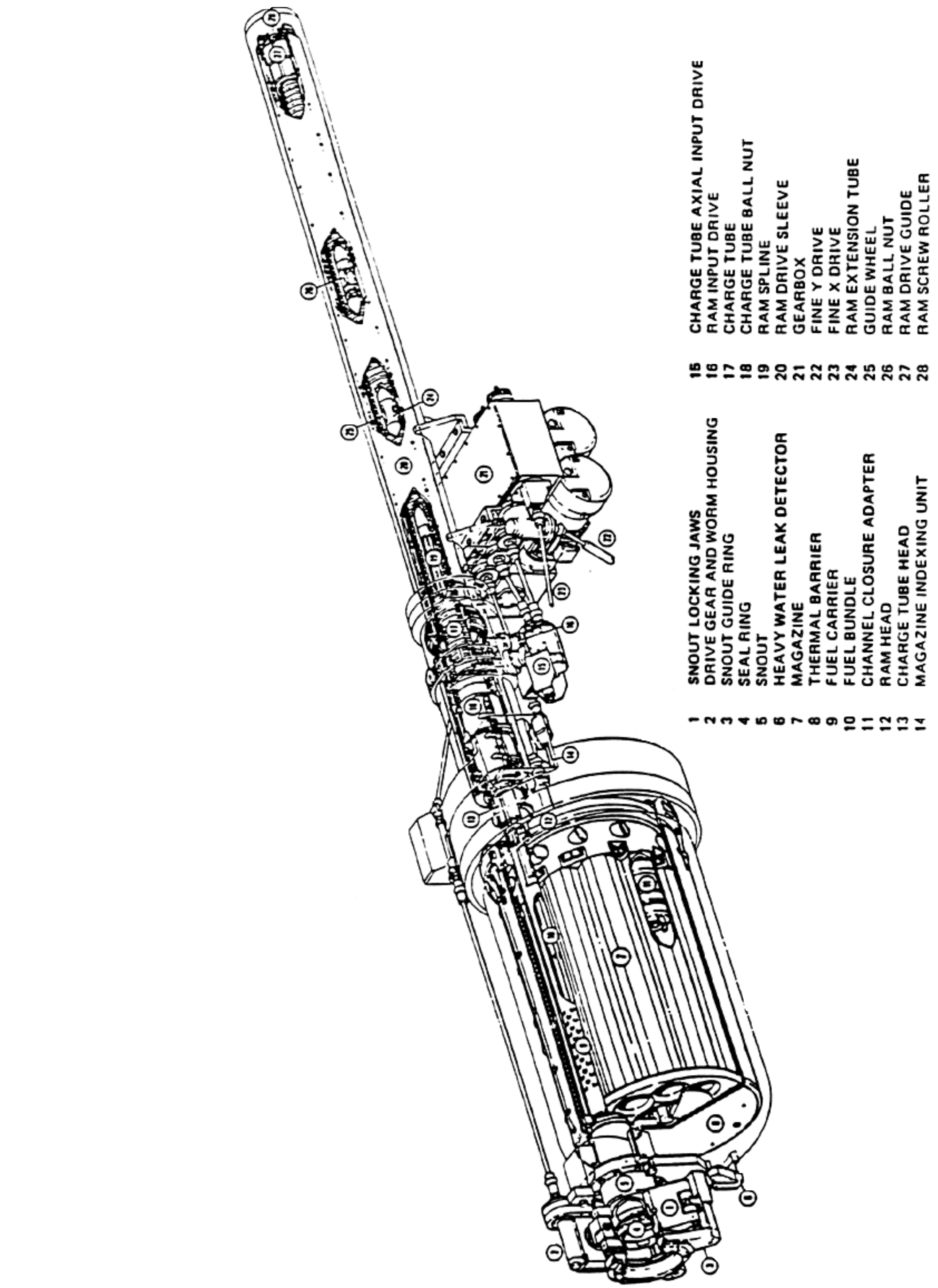


Figure 15
Bruce/Darlington Fueling Machine Head



4.1.4 Separators

Each of the CANDU 6 F/M incorporate two separator assemblies. The functions of this mechanism are:

- to sense the position of the fuel being fed into or being discharged from the reactor and to provide a signal to the computer to stop the ram at the correct position,
- to insert a stop device between two adjacent fuel bundles or between the shield plug and the fuel column, and to restrain the motion of the fuel column extending from the stop device into the reactor,
- to push the bundles that have been separated from the fuel column into the magazine to allow clearance for magazine rotation,
- to verify the presence of shield plug and FARE tool as they pass under the separators at various steps during the fuelling operation.

The two separator assemblies perform identical functions and operate in synchronism. They penetrate through the magazine end cover at a point just forward of the magazine tubes.

In the Bruce/Darlington design, fuel latches are incorporated at the downstream end of each fuel channel pressure tube to carry out a similar function. However, in order to sense the position of the fuel column as a pair of fuel bundles is inserted into the upstream fuel carrier, the upstream F/M ram ballscrew must backwind in reaction to the insertion motion of the downstream F/M ram ballscrew, indicating that the fuel bundles have fully entered the upstream fuel carrier to provide the removal permissive. At the downstream end, the fuel column will rest against the fuel latches to resist the hydraulic drag of the coolant flow. The empty downstream fuel carrier can then be retracted into the F/M magazine.

4.1.5 Flow Assist Ram Extension (FARE) Tool

During fuelling operations on low coolant flow channels, additional force must be applied to the upstream end of the fuel string to move it (and bundles into the discharge machine) downstream when the coolant flow drag on the fuel is insufficient to move the string. This force can be produced either by the C ram of the upstream machine or by a restrictive element in the channel which develops the necessary coolant-flow drag force without the use of the F/M ram. This drag force is created by the FARE tool.

The FARE tool is used in place of the upstream machine ram so that the latter does not enter the core and become activated and subsequently contribute to the dose rate of maintenance personnel working on the machine. The FARE tool (which also becomes activated when it passes into the core) is unloaded from the F/M in the same manner as irradiated fuel bundles before maintenance work is started.

4.2 F/M D_2O System

The F/M D_2O control system provides the heavy water environment at the required conditions to different parts of the F/M, and the controlled motive power for the F/M mechanisms driven by water-hydraulic actuators.

A heavy water environment in the magazine housing is required because:
•this region is in contact with the heavy water of the reactor primary coolant system during fuel changing, and
•coolant is required to remove heat of irradiated fuel bundles in the F/M.

4.2.1 Purpose

The purposes of the D_2O systems are:

- to supply a controlled flow of heavy water to the F/M to maintain the magazine at the desired temperature and pressures, and, when required, to raise or lower the temperature and pressure to a new desired level, primarily to cool the irradiated fuel.
- to supply a controlled flow of cooling water to various seals in the F/M head,
- to supply a flow of heavy water to the C ram in such a manner as to control the direction, force and speed of movement of the ram,
- to supply a flow of heavy water to the separators, to operate the actuators of the feelers, pushers and stops at controlled speeds and pressures,
- to supply a flow of heavy water at controlled temperature to the ram housing,
- to provide a method of detecting leakage of heavy water from the snout cavity (i.e., the cavity between the F/M snout plug and the channel closure plug) when the F/M is mechanically coupled and sealed to an end fitting,
- to provide a means of filling, venting and draining the F/M head.

4.2.2 Temperature Considerations

The heavy water for control of the environment in the F/M magazine housing is provided at temperatures between 40°C and 180°C . When the F/M is on-reactor the temperature control point for the magazine is 93°C. The magazine D_2O supply temperature is maintained at a higher temperature to compensate for cooling flow at 53°C to the F/M ram assembly.

The temperature difference between the heat transport system fluid and the fluid in the F/M is limited to avoid excessive thermal shock to the irradiated fuel bundles which have been exposed to the conditions in the reactor.

4.2.3 F/M D_2O Circuits

The D_2O system, or group of circuits, which directly serves one F/M is covered in the flowsheet general arrangement of Figure 16. The same circuit applies for each F/M. Some of the F/M water-hydraulic circuits are process water and the others are power water.

The process water circuits provide environmental control in the magazine housing and in the ram housing; environmental control at the local regions of hydrodynamic seals on the drive shafts for the magazine, the B ram and the latch ram; and the detection of leakage into and out of the cavity between the F/M snout plugs and the end fitting closure plug.

The power water circuits are for hydraulic actuation of the C ram; feeler, pusher and stops of the fuel separators.

4.2.4 Snout Cavity Leak Detection Circuit

Two separate checks are required during fuel handling (F/H) sequences to ensure that no F/M snout D_2O leakage exists and that the heat transport system boundary is intact after a fuelling operation. Each check is carried out after the respective seal has been made as described below.

The snout cavity leakage check is carried out after the F/M has been mechanically clamped and sealed to the end fitting, and before the snout plug is removed. The test ensures that a good hydraulic seal exists at the interface between the F/M snout and the reactor channel end fitting.

The channel closure leakage check is carried out after completion of the fuelling operation and when the channel closure and snout plugs have been replaced, but before the F/M is unclamped. The test will ensure that the channel closure has been properly replaced in the end fitting and that D_2O will not leak from the reactor after the F/M has been unclamped.

4.2.5 Snout Clamp Lock Hydraulic Circuit

Snout clamping action is carried out by rotation of the clamping barrel, driven by two gear racks which are oil hydraulically operated.

To ensure clamping action is maintained one of the racks is locked by the insertion of a pin. This pin is pushed into position by a water hydraulic actuator which is pressurized from the coolant pressure in the snout cavity. The piston in the actuator operates against a spring pressure so that when the pressure is reduced the piston is pushed back and the lock pin is withdrawn. Thus as long as the snout cavity pressure remains above a specified level inadvertent unclamping cannot take place.

4.3 F/M Oil Hydraulic System

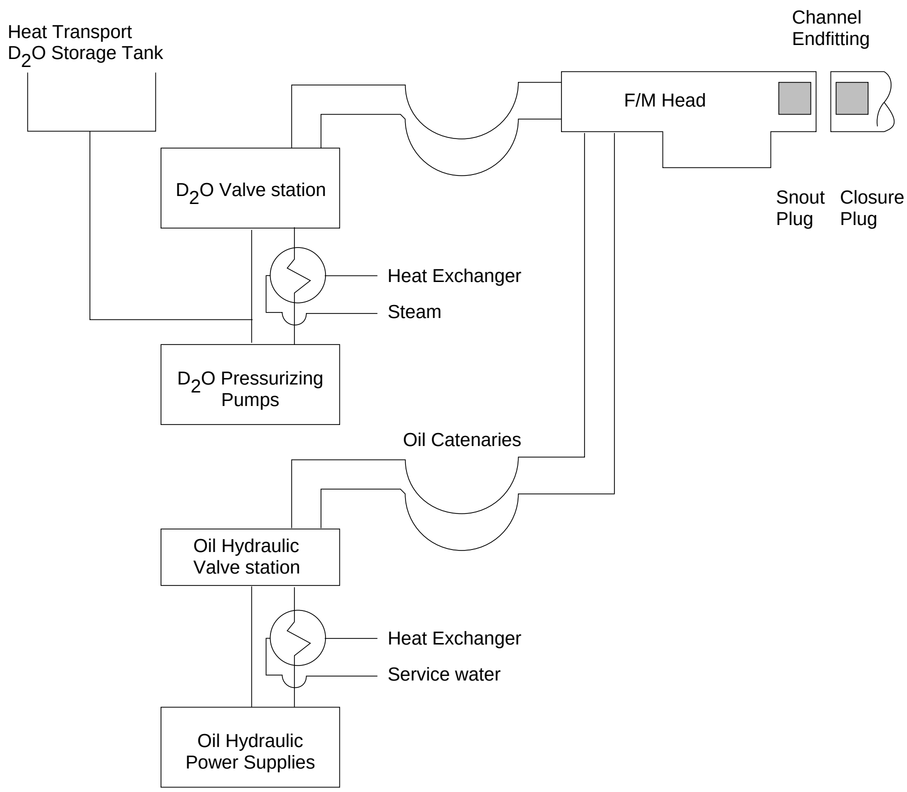
Several functions on the F/M head and carriage are carried out by means of oil hydraulic power provided by power supplies and their associated control systems through the catenary system.

The oil hydraulic system operates actuators, which, with their associated valves and tubing, are mounted on the F/M head and carriage.

The purpose of the F/M oil hydraulic system is to provide controlled conditions of flow and pressure to operate its associated actuators on the F/M head and carriage.

Figure 16

Pickering/CANDU 6 Fuelling Machine Head Hydraulic System Block Diagram



The overall F/M oil hydraulic system consists of two identical and completely separate systems, one for each side of the reactor (Figure 16). Each system comprises of an oil hydraulic power supply system and oil hydraulic control circuits, serving the F/M head and carriage.

4.3.1 Oil Hydraulic Power Supply System

The hydraulic power supply is composed of:
the pressure generating unit (power pack) including the oil storage tank,
the smoothing and filtering unit with its isolating valves (valve station).

4.4 Fuelling Machine Transport and Supports

The F/M must be able to traverse the whole reactor face so that access to all fuel channels is possible. The F/M must also be capable of visiting the new fuel port, irradiated fuel port and service port.

The F/M is transported by a F/M carriage mounted on a bridge that moves vertically with ballscrew drives. Horizontal traverse is achieved by the motion of the carriage along the rails that are provided on the bridge (Figures 17 and 18).

Heavy water, electric power and control signals are supplied to the F/M through a flexible catenary of hoses and cables which connects the mobile F/M to the station auxiliary systems.

The F/M is secured to the carriage through a suspension. The suspension is a gimbal assembly that allows the F/M to align properly with a fuel channel end fitting to reduce the forces exerted on the end fitting by the F/M. The gimbal assembly is restrained by spring stabilizers, except when the F/M is locked to an end fitting. The catenary hoses and cables are connected to the F/M by quick-disconnect type couplings which, in conjunction with a mechanical disconnect, enable the F/M head to be removed remotely in the unlikely event that irradiated fuel should become stuck in the F/M and cannot be unloaded by normal, operational means.

4.5 Summary of Auxiliary Support Systems

Table 2 is a basic comparison of auxiliaries for Pickering/Candu 6 vs Bruce/Darlington.

Table 2:
Comparison of Candu Fuelling System Auxiliaries

Type	**Source**	**Use Pickering/CANDU 6**	**Use Bruce/Darlington**
Electric	A.C. D.C.	Valve Power Instruments	Power Motors Instruments
Oil	Pressure Pump	Actuators Oil Motors	None None
Air	Station Supply	Fuel Transfer Actuators, Motors, Valves	Gap Sensing of Snout to End Fitting
D_2O	Pressure Pump (H.T. supply)	Fuelling Cooling "C" ram, Separators	Fuel Cooling

Figure 17
Pickering Fuelling Machine Maintenance Facilities

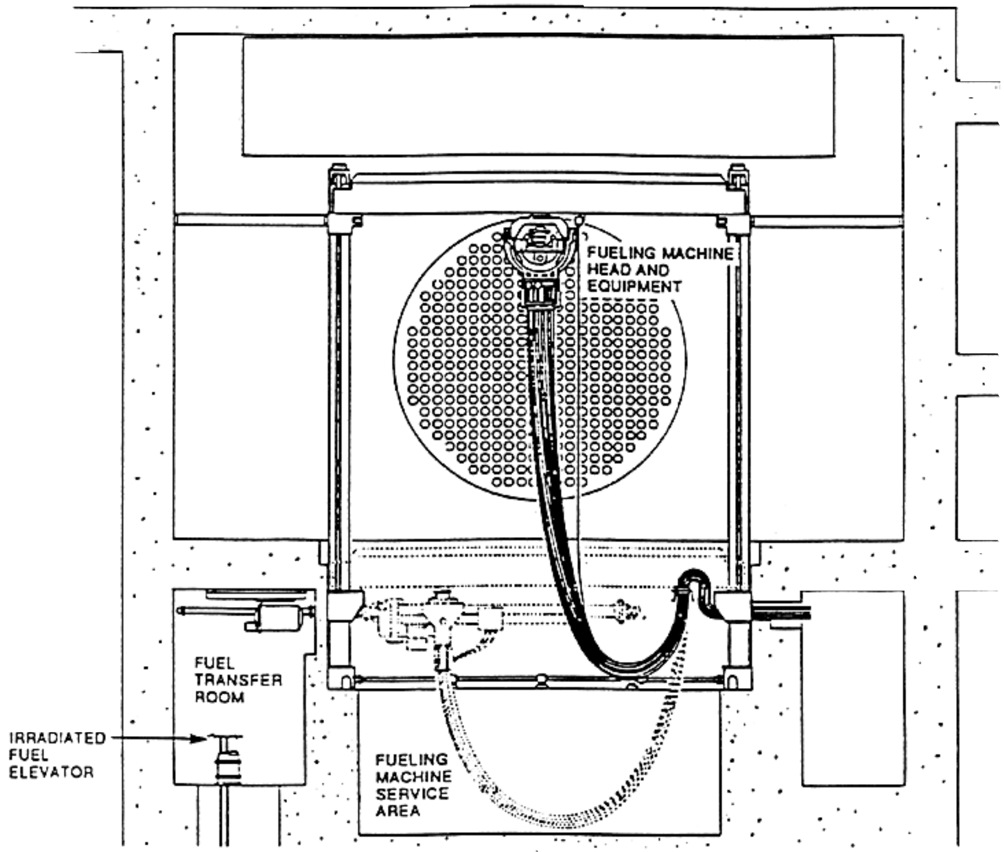
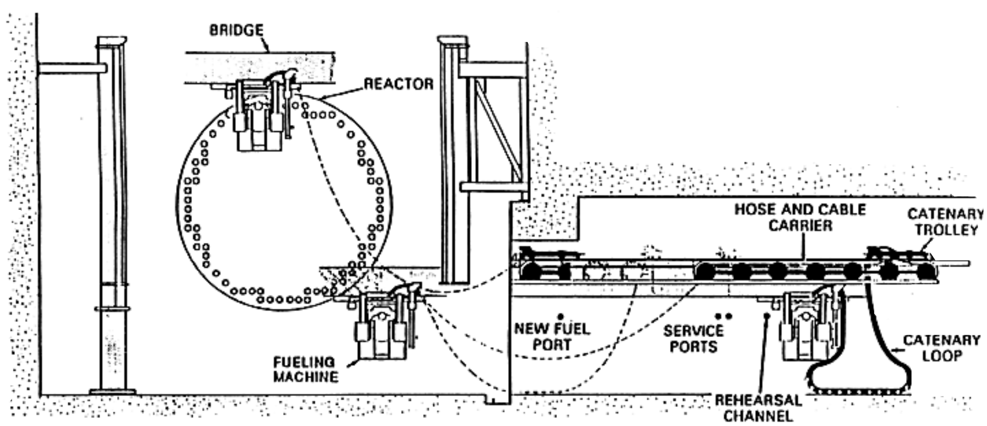


Figure 18
CANDU 6 Fuelling Machine Maintenance Facilities



5.0 Control, Monitoring And Diagnostics

5.1 Control System

Automatic control by means of computer is normally employed on the F/M and its transport system. The sequences and logic of operation are designed to maintain fuel integrity, and personnel and equipment protection during all phases of the fuel handling operation.

A fuel handling control console is provided in the station main control room. From here the fuel handling operations are controlled, except for the loading of new fuel into the new fuel transfer magazine and the operations of irradiated fuel storage. The main digital control computers are available for reactor control: one in operation and the other on standby. The fuel handling operations are handled by the standby digital control computer.

Two identical, complete and separate control systems (diagrammatically shown in Figure 19) are provided, one for each F/M. The only communication between the two systems occurs during fuel column movement in a channel, initially on channel closure plug removal to check that both machines are at the same fuel channel.

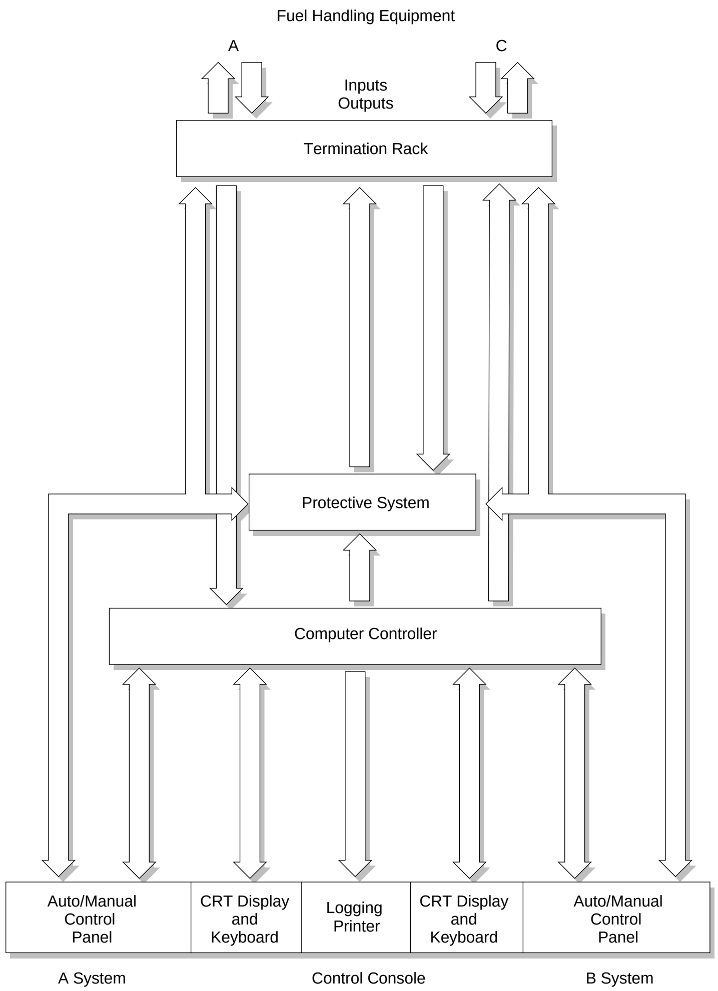
Utilization of automated computer operations provides control features and redundancy that greatly increase reliability and reduce operator error. Throughout the fuel movement, a sequence of logic permissives is applied in conjunction with position monitoring and feedback. Interlocks are provided to prevent undesirable operations.

The control system field components are also selected to ensure that they withstand the required radiation, temperature, humidity and other environmental conditions.

In the event that the automatic control mode is impaired, provision is made for the operator to interpret the information feedback and to take corrective action.

Some manual emergency drive provisions are provided on the F/M so that intervening action can be taken with special tool extensions that can reach into the reactor vault.

Figure 19
Fuel Handling Control - Block Diagram



The mode of control is selected at the control room fuel handling control console. Four modes of control are available: automatic run, automatic step, semi-automatic and manual operation.

Automatic run is the normal mode of operation. In this mode of control the station digital computer takes command once a fuel channel and "job" are selected, and continues until refuelling is completed. Communication between the computer and the instrumentation and control devices permits the computer to maintain control and perform the fuel changing operation. Only if a malfunction occurs should it be necessary for operator intervention.

In the automatic single step mode of control the computer controls the fuelling operation, however, after each step of a sequence is completed, the computer must be commanded by the operator to proceed to the next step.

Semi-automatic operation consists of keyboard statements supplied by the operator independent of any `job' or `sequence', which are then executed automatically under computer control.

Manual operation requires the operator to perform each step by operating the functional controls on the control console. The operator checks for completion of each step using the console indicators, and then initiates the next step.

A protective system, which is a set of logic relays, has all the output control commands (both computer controlled and manual) routed through it (Figure 19). Its function is to prevent, through the use of interlocks, major damage to equipment or hazardous operations for personnel. The logic of this system is separate and in addition to that provided in the computer. The interlocks can be bypassed by handswitches located at the control console only with proper authorization, which must be obtained beforehand.

The control room fuel handling console consists of two identical sections, one for each F/M and a control panel for common systems. Apart from the control panels, each F/M section contains a cathode ray tube for data display, an operate panel to communicate commands and select desired displays, and an alpha-numeric keyboard for operator communications with the computer.

5.2 Safety And Reliability

When the F/M is coupled to the reactor fuel channel, it becomes an extension to the primary heat transport system. The pressure boundary of the F/M is therefore designed to the requirements of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB for Class 1 components. The F/M head support system is designed to the requirements of ASME, Section III, Subsection NF.

An analytical model of the F/M head mounted on its support system, attached to the fuel channel and reactor structure is constructed to represent the mass, stiffness and degrees of freedom. This model is subjected to the design basis earthquake (DBE) Category A floor response spectrum, and worst-case dynamic loads are generated by analysis. ASME Code static analysis is used to perform stress analysis. Failure of the components could result in breach of the pressure boundary that would lead to a loss of coolant from the primary heat transport system.

Analysis are carried out to study performance of the fuel handling control system in the event of loss of coolant in the primary and secondary heat transport systems. This study includes potential flooding of electrical equipment, common mode failures, mechanical and electrical failures, and oil and D_2O cooling system failures, to ensure safety of operation.

Fault-tree analysis of all major fuel handling assemblies and critical components in the control system is carried out during design to ensure reliability. Data for this analysis are drawn from prior experience with equipment in CANDU plants and from extensive development laboratory testing.

5.3 Maintainability

The ability to meet the on-power fuelling requirements of the fuel handling system depends greatly on equipment maintainability during routine servicing and breakdown or repair operations.

In the design, considerable attention is paid to eliminate unnecessary tight tolerances on mechanical components and system controls.

Other factors that contribute to maintainability are remote monitoring, calibration and adjustments to the drive system. Modular construction of sub-assemblies that contribute to rapid exchange of components and packages are also important contributing factors to maintainability.

Maintenance areas for the F/Ms are located adjacent to the reactor vault and access is possible when reactor vault doors are closed with the F/M in the maintenance lock (Figure 18).

5.4 Safeguards

The International Atomic Energy Agency (IAEA) safeguards nuclear facilities by nuclear material accounting, complemented by a combination of containment and surveillance measures. This approach allows the IAEA safeguards objectives to be achieved with minimum intrusion upon routine plant operations.

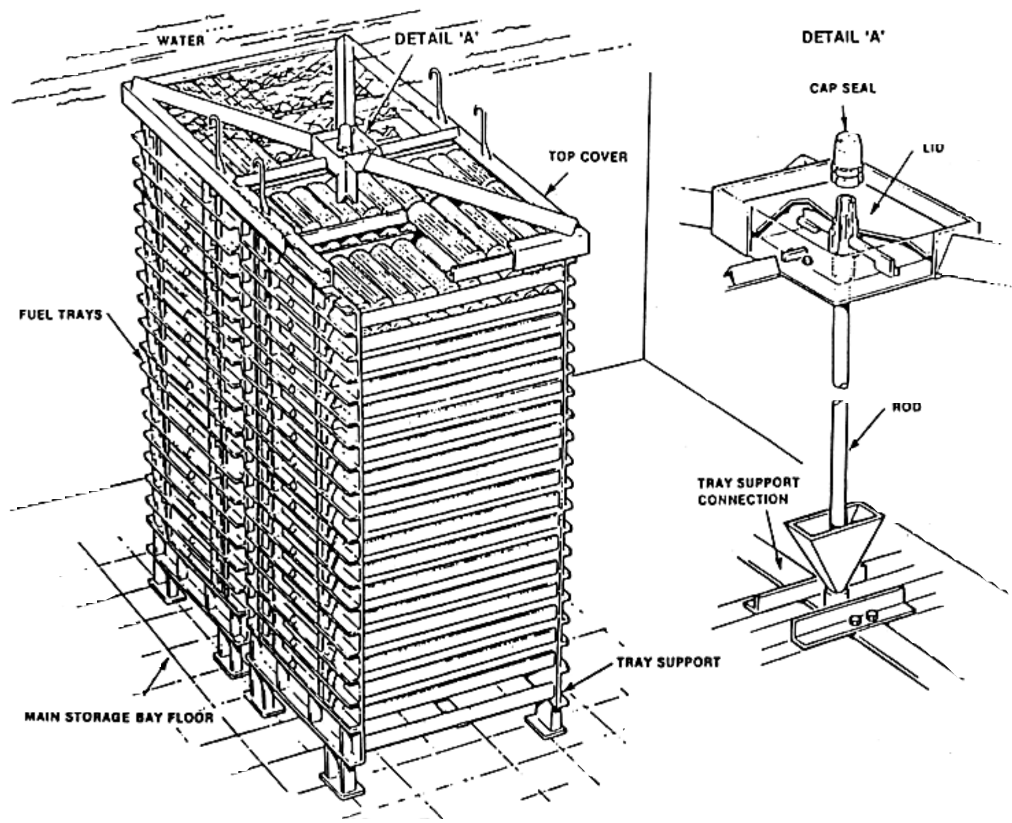
Television and film cameras are deployed along the fuel transfer path, including the reactor vault area, F/M maintenance vaults, fuel unloading area, and irradiated fuel receiving and storage bays. Bundle counters are located close to the fuel transfer ports, to monitor the fuel flow and keep an independent, irradiated fuel inventory. Darlington has a bundle counter system that monitors bundle movement directly from the core.

The station's civil, electrical and mechanical design includes provisions such as penetrations through the containment boundary, embedments, signal and power cables, brackets and supports for the above equipment and for devices supplied by the IAEA.

Containment measures in the fuel storage bay include tamper-indicating frames or covers that prevent any fuel bundle from being removed from the storage tray without breaking the seal or damaging the frame or covers (Figure 20). Surveillance measures also include optical devices such as closed circuit television and film cameras, and monitors such as fuel bundle counters and thermal-photo luminescent dosimeters.

Direct fuel monitoring of fuel storage operations is routinely carried out by I.A.E.A. inspectors as an exercise in recognizing normal fuelling activities versus diversionary operations.

Figure 20
CANDU 6 Fuel Storage Stack with IAEA Safeguards Containment



6.0 Fuelling Rates, Policies And Procedures

Table 1 indicates typical fuelling rates to maintain equilibrium, to maintain these rates requires a high rate of Fuel Handling availability. The Fuel Handling Systems are in a constant state of start-up, operating, shut-down or maintenance. There is a severe consequence if close control is not kept on every activity so operators adhere to many policies and procedures to maintain safe and reliable Fuel Handling and reactor operation. The following is a list of typical station policies and procedures, and prerequisite conditions the operator should address before fuelling.

- Is the F/H system in a ready state? (ie) No work permits are in effect, equipment has been tested, the fuel storage system is available, is failed fuel expected, are authorized operators available.
- Is access control of the fuel handling system in effect? (ie) Are shield doors locked and keys accounted for.
- What is the history of the channel about to be fuelled? (ie) Have there been any closure plug or shield plug problems?
- Is the reactor in a transient condition? (ie) Fuel movement during transient reactor conditions would add undesirable transient condition to reactivity control.
- Is there sufficient range of control available in the specific zone control element? (ie) If not low enough in the control range, the zone will be driven outside control limits by the added reactivity and reactor power reduction would be required to avoid regional overpower. The regional overpower instrumentation must also be calibrated frequently in certain reactors (Pickering B).
- Is the reactor in a shut down state? (ie) Special approvals are required to add reactivity in shut down state because of the limited amount of positive reactivity control.
- Has the channel or adjacent channels been fuelled recently? (ie) Fuel Physics personnel must follow a number of basic rules with regard to timing, spacing and fuelling sequence for optimum flux shape and issue specific "fuel change orders" to the operators.
- Is the adjuster configuration consistent with reactor power level? (ie) If there has been insufficient fuelling then adjusters may be withdrawn to compensate, but is done with specific reactor power limits. The operator must be very aware of the effects that fuelling will have and whether adjuster movement is pending, causing additional transient.
- Is there at least one temperature detector (RTD) available to indicate the possibility of flow blockage in the channel during fuelling? Has the fuelling operator trended the channel temperature?

7.0 Potential for Radioactive Release and Radiation Hazard to Operator

New Fuel

New fuel does not present significant radiological hazards. Stacked new fuel bundles for example in storage will give gamma dose rates of a few tens of μSv/h. If mechanical handling equipment is not available, leather gloves should be worn when handling new fuel bundles to reduce the contact dose rates from beta radiation from the UO_2 which penetrates the sheathing.

New fuel contains small quantities of uranium daughter products but since the fuel is enclosed in a sheath and there is no driving force to move the uranium daughters out of the sheath, very little radioactive material escapes. The total annual escape from this source is negligible.

Irradiated Fuel

Spent fuel, known also as irradiated fuel, presents significant radiological hazards and extensive precautions and great care in handling must be taken to provide acceptable levels of occupational and public safety.

The spent fuel handling system moves irradiated fuel bundles containing large quantities of fission products. Most fuelling operations do not result in any significant release of radioactive material because the fuel sheath is intact. In a few cases where there are defective fuel element sheaths, larger quantities of radioactive material may be released from the fuel element and some of these can escape from the fuel handling system into the containment atmosphere. Note that about 0.1% of discharged fuel bundles have defects (i.e., about five bundles per year per unit), thus only several detected spent fuel bundles are handled per year by the system. Operating experience shows a typical pin hole defect releases about 20 Ci of fission product noble gases during a typical fuel handling sequence. Other species of radioactive materials are not very volatile due to the chemical condition of the heavy water in the fuelling machine head. Thus the movement of defective bundles results about 100 Ci of noble gases being released per year.

Some CANDU stations incorporate an off-gas management system to test off-gases prior to release to environment. Active gases are diverted to the actual ventilation system and released via the stack to the atmosphere.

Systems which handle spent fuel bundles must be heavily shielded to reduce dose rates in accessible areas to low levels. The spent fuel bay stores the fuel bundles discharged from the reactor. The bay water provides shielding and cooling. Radioactive material enters the spent fuel bay water by escape of fission products from the irradiated fuel bundles, by some removal of crud from the fuel element surface releasing activated corrosion products into the bay, and by some transfer of radioactive materials with small quantity of heavy water carried over with the fuel bundles. All these contribute to the inventory of radioactive material in the spent fuel bay, and a build-up of radioactive material in the fuel transfer system.

The spent fuel bay has a dedicated ventilation system which has a filtering system consisting of a pre-filter + HEPA filter + charcoal filter + HEPA filter; air flow passes through this filter train and discharges via the stack. The stack monitor measures particulate, radioiodine and noble gas activity released to the environment.

While the above discussion accounts for the routine amount of radioactive release, there have been several cases of complete bundle destruction while in transfer, releasing large amounts of activity, requiring "boxing up" the ventilation systems and requiring months of operator recovery and clean-up.

The most vulnerable point in the transfer operation is when the bundle is in air during transition from D_2O cooling into storage bay water. Irradiated fuel freshly removed from the core can only sustain about 2 minutes in air without risk of sheath rupture.

Operating experience has shown that it is a good strategy to keep a defective bundle in the fuelling machine until the gaseous release is complete but contained within the heat transport system rather than proceeding to release it to the "open" fuel storage bay system. While "canning" equipment is available in most fuel storage systems the majority of gaseous release occurs before it can be sealed in the can.

To maintain the fuel handling system maintainers can expect exposure to the entire spectrum of radioactive hazards including tritium, component activation, loose and fused contamination due to fission product and corrosion products.

Appendix A

Codes and Standards

A.1 CSA Standards

A.1.1 Purpose

The CSA N285 series of standards has been produced to provide uniform rules for the design, fabrication, and installation of pressure-retaining systems and components in CANDU nuclear power plants. CSA Standard CAN3-N285.1 provides direction into the ASME Code requirements to properly relate the design and construction of specific CANDU components. In other cases, CSA Standard CAN/CSA N285.2 rules are provided where the ASME Code does not address CANDU design needs. The design fabrication, installation, commissioning and operation of nuclear facilities in Canada are also subject to the Atomic Energy Control Act and Regulations. Therefore, additional requirements may be imposed by the Atomic Energy Control Board of Canada.

The specific objectives of the series are:

- to establish rules relating to authorization, approval, and acceptance, where such rules differ from those specified in the ASME Code;
- to specify requirements for materials and rules for the design, fabrication, installation, examination, inspection, testing, and repair of pressure-retaining systems and components, where such systems and components are not covered by the ASME Code;
- to establish rules for classification of systems and components based on the rationale and criteria consistent with the Canadian safety philosophy, as set forth by the Atomic Energy Control Board;
- to set up rules for the periodic inspection of CANDU nuclear power plants;

- to provide interpretation of the rules contained in the standards for nuclear power plants systems and components.

A.2 Description

The CSA Standards that are pertinent to fuel handling design are as follows.

General Requirements for Plants and Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants (CAN3-N285.0)

This standard specifies the general requirements for the design, fabrication, and installation of pressure-retaining systems and components in CANDU nuclear power plants. Most of these requirements govern the Canadian administrative system of classification, registration, and quality assurance, where they differ from the ASME Boiler and Pressure Vessel Code.

Requirements for Class 1, 2 and 3 Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants (CAN3-N285.1)

This standard specifies the requirements for design, fabrication, and installation of Class 1, 2, and 3 pressure-retaining systems and components in CANDU nuclear power plants, as defined by CSA Standard CAN3-N285.0. Classes 1, 2 and 3 basically include nuclear systems and components excluding containment systems. To a large extent these classes are adequately covered by the ASME Code, however, some additional technical rules complement the ASME Code.

Requirements for Class 1C, 2C and 3C Pressure-Retaining Components and Supports in CANDU Nuclear Power Plants (CAN/CSA-N285.2)

This standard applies to pressure-retaining components of CANDU nuclear power plants that have a Code Classification of Class 1C, 2C, or 3C as defined by CSA Standard CAN3-N285.0. These classes include components that would be classified as Class 1, 2 or 3 components according to CAN3-N285.0, but do not have their requirements given in CAN3-N285.1. Essentially, these are Class 1, 2 or 3 components for which ASME Boiler and Pressure Code rules do not exist, are inapplicable or are insufficient. Use of a non-ASME code material does not mean the component will be classified as 1C, 2C or 3C.

These rules complement those in CSA Standards CAN3-N285.0 and CAN3-N285.1 for the design, fabrication, installation, examination, and inspection of CANDU nuclear power plant components and supports.

Requirements for Containment System Components in CANDU Nuclear Power Plants (CAN/CSA-N285.3)

This standard specifies the requirements and establishes the rules for design, fabrication, and installation of pressure-retaining containment system components. This standard does not cover the requirements for systems.

Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants (CAN3-N289.3)

This standard applies to those structures and components in CANDU nuclear power plants that require seismic qualification by analytical methods.

Seismic design requirements for commercial structures and industrial plants have existed in Canada for many years through the National Building Code of Canada (NBCC), which is mandatory throughout Canada. The seismic design of nuclear power plants requires special consideration for the safety of the public. The seismic design philosophy for CANDU nuclear power plants is based on principles established by the Atomic Energy Control Board of Canada.

A.2.1 Supports

Fuelling machine supports in CANDU reactors are composed of structural supporting elements and mechanisms. Portions of them have mobile functions not usually found in supports for pressure-retaining components. For example, an elevating bridge and carriage is commonly used to move the F/Ms from one reactor channel to another during on-power fuelling. A ballscrew and nut assembly is often used to produce bridge and carriage motion as well to provide support.

Mechanisms that produce or control motions and carry support loads, whose failure would result in a loss of support, must satisfy a stress analysis, experimental stress analysis, or load rating, similar to that required in the applicable ASME Boiler and Pressure Vessel Code section.

These mechanisms must be equipped with controls and interlocks to prevent motion of the support that could result in overstressing a pressure-retaining component or its support. It must also be possible to verify the operation of controls and interlocks.

A.2.2 Threaded Connections and Tube Joints

Where threaded fasteners are permitted by the ASME Boiler and Pressure Vessel Code, use of helical coil or other metallic inserts are allowed with appropriate destructive testing on representative samples. Non-welded tube joints may be used subject to several testing and design requirements.

A.2.3 Reinforced Elastomeric Hose Assemblies

This section applies to elastomeric components, such as hoses, where their failure results in a significant release of fluid. Seal components such as `O' rings and gaskets whose failure results in a leak rather than rupture, are not covered. Elastomeric hose may be used in Class 2 and 3 systems. Elastomeric hose may be used only if a failure will not cause the operating personnel to exceed their dose limit. Hose materials must be reinforced and compatible with the contained fluid. Hose design must be qualified by proper testing and documentation, and the fabrication of hose assemblies is subject to several test requirements.

A.2.4 Reactor Channel Closure Safety Lock

A safety lock is needed on each channel closure to prevent it from being unintentionally released from a fuel channel. The safety lock must be a positive mechanical locking device. Frictional type locking devices are not acceptable.

A.2.5 F/M Safety Lock

A positive mechanical lock is required to prevent the F/M accidentally unclamping from the reactor fuel channel, when the channel closure has been removed by the F/M. Frictional type locking devices or electrical interlocking devices are not acceptable. A control or manual release to override the safety lock is not acceptable.

The safety lock must be engaged prior to removal of the channel closure by the F/M and remain engaged until the channel closure has been inserted and secured.

A.2.6 Vessels

Design of vessels and their attachments that form part of the containment boundary must comply with the requirements for Class 2 components of CSA Standard CAN3-N285.1 which gives direction into the ASME Boiler and Pressure Vessel Code.

A.2.7 Airlocks and Transfer Chambers

Airlocks and transfer chambers providing access through the containment boundary must meet the following requirements:
doors must continuously maintain the containment boundary,
airlocks or transfer chambers must have pressure-relief systems and minimize the spread of radioactive substances,
the shell of a metal airlock or transfer chamber is considered a vessel. If the shell is made of concrete, it is considered part of the containment structure to be designed in accordance with CSA Standard CAN3-N287.3.

A.2.8 Seal Plates

Seal plates may be used between the containment structure embedment and process system components that penetrate the containment structure, to form a portion of the containment boundary. Seal plates may also provide an anchor or support function for the penetrating system.

A.2.9 Electrical and Mechanical Penetration Assemblies

Electrical and mechanical penetration assemblies shall be designed to comply with the requirements of the applicable class within CSA Standard CAN3-N285.1.

A.2.10 Flexible Bellows and Seals

Flexible bellows and seals must accommodate movements between the containment structure and penetrating systems. When non-metallic flexible bellows or seals are used to perform the containment seal function for the penetrating systems, dual seals shall be installed with provision for in-service testing by pressurizing the space between the seals.

Appendix B

Glossary

ALARA: an acronym for As Low As Reasonably Achievable.

B Ram: a fuelling machine component that withdraws and inserts the channel closure and shield plug, and positions the fuel column. The `B' ram is driven by ballscrews.

Canadian Standards Association (CSA): a non-profit, independent, private sector organization that serves the public, governments, and business as a forum for national consensus in the development of standards, and offers them certification, testing, and related services.

The standards are written, reviewed, and revised by committee members, who represent users, producers, regulatory authorities, representatives from industry, labour, governments, and the public.

The standards cover a number of program areas: environment, electrical/electronics, construction, energy, transportation, materials technology, and welding.

CANDU: an acronym derived from CANada Deuterium Uranium that identifies a series of nuclear reactors developed in Canada. They are fuelled with natural uranium and moderated with heavy water (D_2O).

CANDU 6: the CANDU 600 MWe series nuclear reactors that have been built at Point Lepreau, New Brunswick; Gentilly-2, Quebec; Embalse, Argentina; Wolsong, South Korea; and Cernavoda, Romania.

Capacity Factor: ratio of the averaged operated load for a certain duration to the total rated capacity of the station (maximum load that the station can carry for a long term of steady conditions).

Carriage: a vehicle that supports the fuelling machine head on the bridge. It controls the horizontal positioning and `Z' motion of the fuelling machine head at the face of the reactor.

Catenary: an assembly of flexible hoses and cables connecting the mobile fuelling machine to the stationary auxiliary power systems. The hoses carry D2O, air and oil. The cables carry power and control signals to the fuelling machines.

Channel Closure (Closure Plug): a removable plug which prevents leakage of the heat transport system coolant from the end fitting. The plug is removed and replaced by the fuelling machine during the fuelling operation.

Decay: the decrease in activity of a radioactive material as it spontaneously transforms from one nuclide to another or into a different energy state of the same nuclide.

Defected Fuel: irradiated bundle that has developed a through-wall defect in its cladding and/or at the end cap weld. It can release fission products to the pool water, if not contained.

Deflectors: ballscrew nut components that deflect the balls from the ballscrew thread to the return tubes.

Design Basis Earthquake (DBE): the maximum ground motion of the site that has sufficiently low probability of being exceeded during the operating life of the nuclear power plant, for which unacceptable radioactivity releases can be avoided.

D_2O: see Heavy Water.

Double-Ended Fuel Changing: Loading and unloading of fuel bundles where two fuelling machines are connected to opposite ends of the fuel channel. One fuelling machine accepts irradiated fuel and the other inserts new fuel. All fuel changing is done on-power in CANDU reactors.

End Fitting: the end components of the fuel channels projecting out of the calandria face. The end fittings allow connection of heat transport water feeders and attachment of the fuelling machine head during refuelling.

Floor Response Spectrum (FRS): the response of equipment mounted on a particular floor (elevation) of a structure, when the structure is subjected to the design seismic motion.

Flow Assisted Ram Extension (FARE) Tool: a tool used during refuelling to assist pushing the fuel bundles along the low flow channels. The tool is essentially a free piston that uses the coolant flow in the pressure tubes to provide a ram force.

Fuel: natural uranium in the form of compacted and sintered cylindrical pellets of uranium dioxide (UO_2).
Fuel Bundle: an assembly of fuel elements (fuel sheaths containing nuclear fuel pellets) and end plates, ready for insertion into a reactor.

Fuel Channel: a pressure tube and its end fittings which house the fuel bundles and direct the flow of primary heat transport coolant over the fuel to carry the nuclear heat to the steam generator heat exchangers.

Fuel Element: a cylindrical, hermetically-sealed, zirconium-alloy sheath containing fuel pellets stacked end-to-end. Thirty-seven fuel elements constitute a fuel bundle for CANDU 6 reactors.

Fuel Transfer Port: the port that penetrates the containment boundary providing a site for the fuelling machine to clamp onto, in order to transfer fuel between the fuelling machine and the fuel transfer mechanism.

Fuelling Machine (F/M): a mobile, remotely-controlled apparatus for extracting irradiated fuel and inserting fresh fuel. The fuelling operation is carried out while the reactor is at full power operation.

Bridge: a horizontal beam across the face of the reactor supported between two vertical guide columns by electrically-driven ballscrews. The ballscrews move the bridge vertically across the face of the reactor.

Fuelling With Flow (Flow Assist Fuelling): new fuel bundles are inserted in the fuel channels in the direction of coolant flow. In the high flow channels, coolant flow is sufficient to move the fuel string unaided.

Gimbal Assembly: a suspension that secures the fuelling machine head to the fuelling machine carriage. It also facilitates the alignment and engagement of the head to the reactor or fuel transfer port. The gimbal assembly also accommodates thermal expansion, contraction and misalignment of the fuel channel assemblies.

Grapples: a fuelling machine contingency tool that allows removal of fuel bundles by one fuelling machine only.

Guide Sleeve: a fuelling machine tool that provides a constant bore passage for fuel bundles between the end fitting liner and the fuel bundle magazine stations.

Heavy Water (Deuterium Oxide or D_2O): water in which ordinary hydrogen atoms have been replaced by deuterium atoms. Natural water contains one heavy water molecule for approximately every 7000 ordinary water molecules. D_2O has a low neutron absorption cross section, hence its use as a moderator in the CANDU reactors. D_2O is about 10% heavier than natural water, but has the same appearance and chemical properties. It has a higher freezing point and boiling point than ordinary water.

Hydraulic C Ram: a fuelling machine component that pushes the fuel bundles from the fuelling machine magazine into the fuel channel. The `C' ram is driven hydraulically with D_2O.

Hydrostatic Seal: sealing in which the sealing faces fluid film thickness is maintained by supplying high pressure fluid at the seal face.

Irradiated Fuel: nuclear fuel which has been irradiated in a reactor to the extent that it is no longer economic as a power producer, that is, the fissionable isotopes have been consumed and fission product poisons have accumulated. It is also termed "spent fuel" or "used fuel".

Latch Ram: a fuelling machine component that acts in conjunction with the `B' ram to actuate the latches on the shield plug and channel closure. The latch ram is driven by ballscrews.

Magazine: that part of the fuelling machine head located behind the snout and separators. It comprises a rotor and drive shaft and is housed in a cylindrical pressure vessel. The rotor has chambers for new or irradiated fuel bundles, channel closures, shield plugs, snout plug, ram adapter and the guide sleeve and insertion tool. During fuelling, the magazine can be rotated to align any chamber with the axis of the snout.

Moderator: a material such as heavy water, graphite or light water used in a reactor to slow down or moderate the fast neutrons produced by fission, thus increasing the likelihood of further fission.

Natural Uranium (NU): uranium whose isotopic composition as it occurs in nature has not been altered (0.7% by weight of U-235). The design of CANDU reactors keeps neutron wastage so low that natural uranium can be used as fuel.

New Fuel Transfer Mechanism: a mechanism comprising a magazine and indexing unit, a ram assembly, a loading trough and a new fuel loading ram. Its purpose is to transfer new fuel from the new fuel loading area to the fuelling machine.

Nuclear Power Demonstration (NPD): the prototype CANDU nuclear reactor built near Rolphton, Ontario, designed to prove the technical feasibility for future, large-scale reactors. The NPD began feeding electricity into Ontario Hydro transmission lines in 1962. Decommissioning began towards the end of 1988.

On-Power Fuelling: in CANDU reactors, refuelling is carried out while the reactor is on-power, so that no shutdowns for refuelling are required.

Point Lepreau: the 600 MWe CANDU 6 nuclear rector in Point Lepreau, New Brunswick, which began service in 1983.

Pressure Tubes: the high strength zircalloy tubes that penetrate the calandria and contain the fuel bundles and pressurized heavy water coolant.

Primary Heat Transport System: the heavy water coolant system that removes heat from the fuel bundles and exchanges the heat to the light water which drives the turbines. The primary heat transport coolant flows through the pressure tubes and is separate from the moderator water.

Ram Adaptor: a fuelling machine head attachment that provides a profile corresponding to that of the fuel and forms an extension of the ram assembly during fuelling operations. At other times the adaptor is stored in the magazine.

Ram Assembly: the section of the fuelling machine that includes the ram housing, drive assembly, B ram, C ram and latch ram. The rams position, insert and withdraw the fuel bundles and associated hardware required for refuelling.

Reactor Face: the end planes of the calandria vessel, where the fuel channels can be accessed.

Return Tubes: pathways in the ballscrew nut that allow the balls to recirculate around the ballscrew threads.

Safeguards: a system of technical measures within the framework of international non-proliferation policy entrusted to the IAEA in its Statute and by the Non-Proliferation Treaty (NPT).

Separators: components of the fuelling machine that sense the fuel column position, restrain the fuel column, separate the required fuel bundles from the column and move the required bundles into the magazine.

Shield Plug: removable plug which provides shielding against axial streaming of neutron and gamma flux from the reactor end fittings. The upstream plug also redirects the coolant flow axially through the fuel channels, eliminating coolant swirling and uneven bundle cooling.

Snout: the front end of the fuelling machine head. The main components of the snout are the antenna assembly, clamping mechanism, seal and outer and centre supports. The snout homes onto a selected fuel channel assembly during refuelling and clamps and pressure seals itself to the end fitting.

Snout Plug: a latching mechanism and seal assembly which seals the snout of the fuelling machine head when it is off the reactor face. This permits the interior of the fuelling machine to be maintained full of water at the operating temperature and pressure.

Storage Bay: a large pool of demineralized water in which irradiated fuel is stored while fission products decay.

Uranium Oxide (UO_2): ceramic grade uranium oxide from the refinery. When compacted and sintered during fuel manufacturing it becomes a ceramic, having characteristics of chemical and radiation stability, good gaseous fission product retention and a high melting point.

X Motion: horizontal motion of the fuelling machine towards and across the reactor face.

Y Motion: vertical motion of the fuelling machine across the reactor face.

Z Motion: motion of the fuelling machine parallel to the longitudinal axis of the fuel channels.

Human Factors

Table of Contents

Introduction

In this section we will be talking to you first about human error, and then a bit about latent human failures in complex systems. Since the area of human error is such a huge one, this brief presentation will serve just as introduction, so that you may see the several ways in which human error may be classified.

Before we get into any examination of human error let's be sure we are all working from the same definition. As a general definition: human error can be considered to be a human action that exceeds some limit of acceptability for a system.

You'll notice that we have considered the "system" within our definition. This is because an error cannot be defined in a vacuum. It must be related to the overall properties of the system, since it is the system which defines the limits of acceptable performance. Each system is different, with different limits, and what may be considered an error in one system may be perfectly acceptable in another system.

Most human errors are unintentional actions that are inappropriate to the given situation. Also, the term itself should not be taken to imply anything bad about the person committing the error. It does not mean that the operator was bad, or stupid or careless. In fact, most errors result from a flaw in the system, and are not attributable only to the operator.

Let us look now at the first system for the classification of human error.

SWAIN

The first system of classification, developed by Swain, consists of several categories of error as follows:

a) error of omission

b) error of commission

- extraneous act
- sequential error
- time error
- selection error
- qualitative error

a) an error of omission is one where the operator skips all or part of a task.

b) an error of commission is one where the task is performed incorrectly.

Within this category are several sub-categories.

- an extraneous act is one that should not have been performed because it diverts attention away from the system thus allowing for potential failure. An example would be someone reading a book while sitting at a radar scope, and who thereby misses a blip on the radar

- a sequential error is one in which a task or activity is performed out of sequence
- a time error is one in which the operator performs the task too early, too late, or not within the time allowed.
- in a selection error the operator is seen to make a wrong selection, such as choosing the wrong size of nail
- in a qualified error the operator wouldn't do something to a sufficient degree. For example. the operator may not close a valve tightly enough, thereby allowing for potential leaks.

This system of classification developed by Swain is fine as far it goes, but human output is related to system requirements, without consideration of human internal processes. Norman, who was more interested in the psychological bases for error, went a step beyond this, and has developed a two-group classification system. Before we look at this, however, a little background is necessary.

Background

Rasmussen proposed that human behaviour operates at three levels:
1) skill-based behaviour
2) rule-based behaviour
3) knowledge-based behaviour

1) in skill-based behaviour responses are made in a rapid, automatic mode, with a minimum of attention. The actions made are usually well-organized and highly practiced, though they are subject to conscious control from time-to-time, depending upon the level of practice. An example of skill-based behaviour would be braking a car or making steering corrections.
2) in rule-based behaviour actions are performed according to rules. After mentally reviewing rules the operator will make a decision and take appropriate action. Mental processing in this type of behaviour is conscious, slower than for skill-based behaviour, and is less automatic. For example, if light "A" flashes, push button "B".
3) knowledge-based behaviour is usually adopted to deal with unfamiliar situations where the stored rules are not sufficient. Here more general knowledge about the system's behaviour, the environment, and the goals to be fulfilled are consciously integrated to formulate a plan of action. An example here would be the operator who is trying to do troubleshooting for an unfamiliar malfunction in a complex system.

In summary, then, Rasmussen identified three different levels of operation along an automatic-attentional continuum.

Norman then took this information and proposed a 2-category classification system for errors which corresponds to the three levels of behaviour.

1) In the first category Norman identifies slips, which are prominent when the operator is working at the skill-based level of behaviour. These are errors

where the intention fits the goal, but because of a failure in the execution of the action, it is not carried out as planned. Slips occur when three conditions exist in an action sequence:

- the tendency for the inappropriate action to be selected is already high
- stimuli that would trigger the appropriate action are very similar to those that trigger inappropriate actions
- the operator is functioning in the skill-based mode,

For example, suppose the operator wanted to type "mechanize" but makes a slip. As a result the word comes out "mechanism" or "mechanics". In this case, the letters "mechani" triggered the completion of the word incorrectly.

Another example is seen in instances of stimulus-response incompatibility in controls and displays. Two lights, two switches. When functioning at a consciously controlled level the operator probably will not make the slip. When the attention is diverted elsewhere, however, and the operator goes into auto mode the slip occurs because the operator selects the switch from the position he subconsciously expects, since the tendency to do so is high (with proper stimulus-response compatibility).

2) In the second category Norma defines mistakes as actions which were carried out as intended, but which are not appropriate and which do not fulfill the goals. These are more likely to occur when the operator is working at a rule- or knowledge-based level, relying on conscious decision-making. These errors originate in the planning phase and are failures in the inferential or judgmental processes involved in the selection of an objective or the means by which to achieve it. In general, mistakes are likely to be more subtle and complex than slips, and as a result more dangerous.

Other work has been done which has produced a recognition of three classes of mistakes. The first, mistakes of bounded rationality, result because only a small aspect of the total problem can be held by working memory, information processing and attentional processes. This means that there will be a narrow focus on the problem. In addition, there are difficulties in maintaining the focus, and only small amounts of information may be remembered and applied to the situation. Finally, irrelevant information from memory and from the environment will intrude on the situation. Mistakes of bounded rationality will have the following characteristics:

- oversimplification
- search for objectives restricted
- objectives chosen will be short-term rather than long-term
- objectives chosen will be merely adequate rather than ideal
- fewer ways of obtaining objectives will be considered than are available
- plan-related factors based on global rather than subtle distinctions (ties in with oversimplification)

- possibility of complex interactions overlooked or underestimated
- thinking through of the consequences will be partial rather than complete

The second type of mistake, the mistake of imperfect rationality, is different from the first in that instead of stemming from the limitations of attention, it is due to the characteristics if the structures comprising/consisting of the long-term knowledge base. There are an apparently limitless number of these structures each one dealing with a particular aspect of the world. They can be called upon demand, but can also be triggered unintentionally by such factors as recency, environmental triggers and emotional significance. The characteristics of mistakes in this class are:
- too much association with past circumstances than is demanded by present state
- procedures too rule-bound, too rigid, too conservative
- solutions to previous problems will continue to be applied
- too little account taken of actual or potential change in the situation

These are similar to the habit intrusions of slips, but whereas slips are unintentional activation of low-order control structures, these mistakes arise from contextually inappropriate application of intuitive rules of thumb for both judgement and reasoning.

The third type of mistake, the mistake of reluctant rationality, stems from attempts to avoid or minimize the attentional effort required to pursue a new or unusual line of thought. Attentional effort along a new path cannot be maintained for very long,with the result that attention tends to drift toward habitual routes, or is pulled toward emotionally or contextually charged cognitive structures. People will adopt strategies, some efficient, some not, to deal with the cognitive strain in problem-solving tasks. For example, instead of performing a focused search to determine the root cause of an unusual and unique problem, the problem-solvers might test only what they consider to be reasonable hypotheses as determined from past experience (which provides some degree of comfort to the problem-solving situation). This type of mistake, then, results in an excessive reliance on what appear to be familiar cues, and to the application of well-tried problem solutions even in unique and unfamiliar situations, where novel solutions would be more appropriate.

In the next system of classification Reason has begun by proposing that there are several basic error tendencies which make up the roots of most , if not all of the systematic varieties of human error.

1) ecological constraints: we are human beings living on a terrestrial planet with normal gravity. Our body, information processing and sensory systems are suited to only a narrow range of conditions - those we normally encounter in day-to-day life. Regardless of technological advancement (eg.

space travel) we remain the same, and we will respond as though we were in our normal ecosystem. Regardless of how fast a computer may be able to supply information to us, we can only process it in a certain amount of information and at a certain speed.

2) change-enhancing bias: while something like a thermometer is designed to give consistent readings, the human psychological mechanism does not operate in the same way. The calibration is not fixed, and can vary from one occasion to the next. For example scents in a room appear to weaken over time. Same with other types of stimuli. (Adaptation Level Theory)

3) resource limitations: human beings possess finite information-processing resources. Resources are such things as memory capacity, processing effort and communications channels. The advantage to this is that only a limited number of cognitive structures will be activated at any one time. If this controlling feature were absent we would be overrun with incoming information and would not be able to derive meaning from the information and to organize our thoughts . (Information overload.)

4) schema properties: we tend to take the regularities of our world and use that information to develop schema, or cognitive structures consisting of knowledge about the world. Systematic errors can arise:
   - from fitting the data to the wrong schema
   - from employing the correct schema too enthusiastically, thereby filling in information where it doesn't exist in the stimuli (Gestalt)
   - from relying too heavily upon active schemata

   Such errors are likely to occur when new situations develop and the existing routines are not adequate to deal with the novel circumstances.

5) strategies and heuristics: success in carrying out an activity depends largely upon having chosen the correct strategy. An inadequate strategy will lead to particular types of errors, as will depending upon a few well-tried rules of thumb. For example, you have a library of 2000 books. You are intimately familiar with your books, having read them all. Unfortunately someone has moved one of your books, and now you need to find it. What is the fastest way to locate that book? Title, size, colour, thickness or binding (paper vs hardbound)?

Beyond this Reason has listed what he calls primary error groupings. These are the general types of error that one could expect to see.

1) false sensation: this occurs when there is a lack of correspondence between objective reality and our subjective experience of the world. False sensations can be elicited:
   - during and immediately after exposure to a steady input in any modality (eg. arms against wall)
   - in conditions of successive and simultaneous contrast (red/green, noise)
   - in atypical force environments
   - when viewing 2-D representations of 3-D objects
   - when viewing large-scale moving visual scenes (eg. Cinesphere!)

2) attentional failures: problems achieving or maintaining attention can be caused by:
   - coping with distraction
   - dividing attention between the performance of 2 concurrent tasks
   - monitoring
3) memory lapses:
   - forgetting list items
   - forgetting intentions
   - losing track of previous actions
4) unintentional words and actions: include slips of the tongue, Freudian slips, action slips.
5) recognition failures: situations where the operator wrongly identifies something which is not actually present, or fails to recognize something which is there. For example mishearing speech, misreading text, misperceptions of objects or people.
6) inaccurate or blocked recall: similar to the recognition failures, but is linked more to long-term memory failures than is the former.
7) errors of judgement:
   - psychological judgements
   - temporal misjudgments
   - misconceptions of chance
   - misconceptions of risk
   - misdiagnosis
   - erroneous social assessments
8) reasoning errors:
   - errors in deductive reasoning
   - errors in concept formation
   - errors in hypothesis testing

Summary

In this past section we have looked at three different systems for classifying human error. I hope it has to some degree shown you that the complexity of human error lies not in the observable features (Swain), but in the hidden internal processes. At the surface human error can be divided into a limited number of behavioral categories, though these tell us nothing about the origins of the error. The examination of the internal processes is more revealing in terms of possible causal factors, though it remains a difficult and complex undertaking.

Latent Failures in Complex Systems

Now that we all know a bit about human error and how it may be classified,, let's look at human error in complex systems.

In the past few years we have seen several major disasters which occurred in complex socio-technical systems which possessed elaborate safety devices. (Chernobyl, Challenger, TMI) These accidents arose from not one single cause, but from several diverse causal sequences, each alone incapable of causing the accident. In all of the accidents human failure played the dominant role.

Thanks to highly sophisticated safety systems, failures from a single cause are unlikely. However, the use of elaborate defence-in-depth renders the system opaque to those who control it.

The greatest threat to the system is from the accumulation of delayed-action hidden failures in the system, most of which originate from the organizational and material sectors. Before we continue, let's define what is meant by active and latent effect.

a) an active failure is one which has an immediate adverse effect.
b) a latent failure, either decision or action, with damaging consequences which may lie dormant within the system for a long time. They only become evident when they combine with a local triggering factor. (active failure, technical fault, atypical system conditions). They are most likely to originate from people who activities are removed from the human-machine interface: designers, managers, etc.)

There is a growing awareness that attempts to identify and deal with these latent failures will achieve greater benefit than trying to deal with only active failures.

In trying to deal with these latent failures we must recognize several factors:

1) One can consider the latent failure to be like a resident pathogen in the human body (eg. cancer). Everyone has the virus or predisposing conditions, and it only takes a trigger to initiate an illness. As with cancer, several distinct factors must be present, since each one alone is insufficient to cause the disease. The likelihood of developing cancer, or suffering a catastrophic accident, is a function of the number of pathogens, or latent failures in the system. The more the pathogens, the greater the probability that a given set of pathogens will meet the local trigger necessary to initiate an accident sequence.
2) The more complex, interactive an opaque the system, the greater will be the number of latent failures.
3) The higher the position in the organization, the greater the opportunity for generating latent failures, and the broader the reach of those latent failures.

Let's look at a system. There are five basic elements to any productive system:
1) decision-makers (architects & senior executives of system)
2) line management (people who implement the strategies)
3) preconditions (reliable equipment, motivated workers)
4) productive activities (actual performance of machines & people)
5) defences (safeguards)

For the various elements we can see different latent failures in various forms.
1) decision-makers: latent failures have their primary origin here (but are also introduced throughout the system). Even in the best of organizations decisions can turn out to be wrong. Decision-makers may set up defensive filters, thus removing the possibility of addressing the problem directly and thereby setting up countermeasures. May blame operator carelessness or incompetence for accidents.
2) line management:
   - poor decisions may result in undermanning, unsafe assignments
   - poor training may result in workers not equipped with adequate work skills and knowledge
   - poor maintenance planning may result in shoddy workmanship
3) psychological precursors: many of the above factors can lead to psychological precursors such as high workload, undue time pressures, motivational difficulties, which then create the potential for unsafe acts.
4) a psychological precursor can play a role in shaping an infinitely large set of unsafe acts.
5) in a highly protected system the probability of an isolated action leading to an accident is very small. Several causal factors, though, can create a "trajectory of opportunity" through the multiple defences.

Summary

Here we can see the various elements in an organization and how latent failures in the system can lie dormant until a trigger initiates an accident sequence. This would seem to indicate that the main thrust of accident prevention programs should be aimed at eliminating these latent failures.

Regulatory context in AECB

Any regulation stance that is developed to cover people in NPPs will have to be consistent with the existing regulatory philosophy and practice. Regulation of our nuclear power plants is based on the AEC Act and the AEC Regulations. The regulations require that all dealings in prescribed substances and equipment within the national nuclear industry be licensed by the AECB. The licensing approach is documented in a series of licensing guides that make clear the regulatory philosophy, criteria and requirements. These documents spell out the position that the licensee has the primary responsibility to develop the competence required to show that a plant will not pose an unacceptable occupational or public health and safety risk. We give licensees the flexibility to use the approach of their choice to achieve the stated goal, given that it meets with AECB approval. So we avoid "design-by-regulation" that would usurp the licensee's responsibility for safe design and operation.

A regulatory human factors program must be consistent with this traditional AECB concept of regulation. That is, an approach must be documented that states the philosophy of why there should be human factors activities in the industry, what these activities should be, and how this can be achieved. It should state the criteria that apply to those activities, and the manner in which the AECB would apply those criteria in assessment. This approach should not indicate to the licensees the details of how these criteria should be met, or a preferred path to achievement of the required goal.

So human factors regulation will conform to the regulatory philosophy and practice that has been established. In doing so, it will be based on the AEC Act and Regulations as is the existing regulation. It must also follow the existing licensing system. In the development of that regulatory practice, it was recognized that, regardless of the technology involved, human error and system failure may occur. A safety philosophy base on "defence-in-depth" was adopted that states that several independent measures must be taken to minimize the probability and the consequences of human error and system failure. In this it was assumed that it was sufficient to require that there be "selection of competent personnel and their subsequent training, qualification and requalification" (Jennekens,J. INFO-0002, Mar80).

In practice, since there are no substantive human factors requirements in the licensing documents, the solution to the human error problem has become institutionalized as several independent technical measures that represent the defence-in-depth concept.

One aspect of human factors regulation that is somewhat troublesome is the low level of capability in this area of work in the licensee's organizations. When we specify criteria we expect the licensee to have the expertise to understand them, decide how to meet them, and demonstrate to us that they can meet them. We can often be in the situation where the licensee personnel do not have an adequate understanding of the requirements, or criteria, and so the route taken to meet them may be subject to some criticism. Attempts to clarify the requirement can easily lead into a level of prescriptive detail that is incompatible with our regulatory philosophy.

Human factors regulation approach

The regulatory framework in the people area has three components at this stage. The first is a regulatory policy statement that has now gone through several revisions, but is getting closer to being a reality. The second is a series of guides documents that will indicate to the licensees a clear AECB position on these areas. The third component will be a series of more detailed topical guides that will address specific issues.

Human factors policy statement

As the name indicates, this document provides the policy position of the AECB on human factor issues. To make it clear that we do not tell the licensees what to do nor how to do it, the document (Consultative Document 119) states the criteria and the requirements that apply in regulatory review of licensee programs and business unit operation. The basis of this approach is that the licensee is required to incorporate in to activities the available knowledge on human behaviour in individual, group and organizational terms. That is, when a new station, unit, or unit component is planned, or when the manner in which existing units are operated and managed is being altered then known knowledge on the capabilities and the limitations of people shall be incorporated in those activities to ensure to as great an extent possible that the performance of all personnel on the job will not be adversely affected.

Regulatory licensing will require that documentation be provided that shows how this has been done, in terms of the information used, the manner of its integration into the conventional process, and the results achieved by its use. This is to be effected by the provision of a Human Factors Program Plan at an early stage of the change process. This document will tell the AECB what they intend doing, how they are doing it, and what they expect to get out of so doing. After the activity has been completed the licensee will provide to the regulator a Human Factors Program Report that will state how the plan of the HFPP has been implemented, any changes that had to be made in the face of reality, and the results that now have been achieved. Approval of each document after submission would be required before further progress.

Human Factors Regulatory Guides

We accept that the licensees organizations have a modest level of sophistication and knowledge about the people that they employ to design, construct, operate, maintain, administer, manage and plan their facilities. There is a considerable body of knowledge and methodology related to people management that has been developed and is being used in other industries and that has been shown to be quite effective. Our licensees are full of confidence in speaking on such matters. They have been doing it for many years. But their knowledge base and methods are essentially intuitive and inexpert. Guidance will be required so that they know what we expect them to do in general terms, where to look for approved information sources, what methods to employ in conducting their analysis, and how best to integrate all of these into existing processes.

The first AECB human factors guide document is a process guide. In terms of terminology,we regard a guide as a general level document. It is at a higher, more programmatic level than a guideline document, that would present detailed information required to carry out the processes required in the program. So for a program of the scale of the Bruce-A rehab, this document would indicate the general activities that they would be required to initiate and incorporate in the program of work, but would not indicate the design criteria that apply to those doing it. We had a consultant identify, review and gather expert user opinion on the available handbooks, standards, guidelines, specifications, and other references that could be applied to nuclear facilities' life cycle activities. These have been published and will serve as the acceptable detail source documents to which the licensees are referred.

The second guide document is an activities guide. In our formal professional education each of us adopts our discipline's approach to problems, and we got a bag of tools that we have been trained to use in resolving problems. What we are not trained in is recognizing that other disciplines have different approaches to problems appropriate to their own needs and have different tools in their bag to solve their own type of problems. This document outlines the methods, analysis, tests and procedures that are relevant to addressing human problems in organizations at work. The nuclear industry tends to use the tools familiar to technical people who are solving technical problems. That is why the results are usually inadequate when human problems are addressed, and why we hear the familiar statement that human problems cannot be analyzed and treated. It is a common attitude amongst technically trained people that one must use the right tool for the job, an attitude that we all share.

The third guide document will indicate how the existing and well understood design process should be made to incorporate human factors knowledge and principles. The fundamental point here is that technical design often take place independently of requirements for device use and maintenance. This can be counterproductive and not conducive to safety in operation. But this continues to happen. Designers cannot continue in the face of overwhelming evidence to he contrary that they have produced a good design that has been compromised by those who use it. Nor can analysis of those systems in operation, by expert technical assessment or by code analysis, be completely relevant to the real world if real world use and maintenance is not included in the analysis.

Human factors regulatory topic guides

The third part of regulation in human factors will be a series of detailed guides documents. Each will be directed to a specific topic, that has proved to be especially important in the running of this industry. I can see that conditions of working may well require a stand-alone document. It would set out criteria and requirements in areas such as hours of working, so that the complex issues underlying shift work will be spelled out and the regulatory requirements will be known clearly to both management and to labour as they enter into their tripartite contract negotiations. Another issue may well be guidance on the development of training programs and training courses. Later we will talk at more length about what we have achieved in this area. The criteria and requirements that have been developed during this effort should be documented in a more mature form. There is going to be a need for regulatory criteria that apply in the event of a strike by unionized employees so that the ad-hoc arrangements and agreements that are struck when such an event is imminent can be avoided and the utility knows in advance what will be expected.

Human factors objectives

For both human factors practice in industry and in human factors regulation there are some objectives that can be identified to give you a general idea of the direction that is being taken.

Analyze Behaviour

The primary activity undertaken to understand and so to help employees is to analyze behaviour at work. Behaviour is produced at an individual level such as the maintainer who has to remove some bolts and replace a gasket and replace the bolts. There also is a need for team work, or group behaviour, as when a crew has to remove irradiated cobalt adjuster rods at the reactor reactivity mechanism deck. The team coordination is developed through training sessions on a training device.

Behaviour also occurs at the level of the organization, such as when new policies are produced, roles and responsibilities are changed, reporting relationships are changed, or resources availability is modified to meet economic pressures. The higher in the organizational hierarchy the activity takes place, the broader is the range of its effects.

Context

Human behaviour can be classified as human error, and efforts are made to identify the probability of that act being made. But the action of an individual may be correct at one time and wrong at another. The individual may see the two situations as the same, but in fact they are not. The context makes the action right or wrong, but what the individual believes the context to be is the true basis for actions.

Multiple inputs/local triggers

We look for multiple inputs from different types and levels to the causation of events, and see how they all lead to the final local trigger or action that sets off the undesired outcome. The influence of the many activities that precede and lead up to the final action is difficult to trace and document, but only by doing so can an organization understand the nature of its less than acceptable activities.

Predictable, reliable necessary.

Human behaviour follows laws and regularities that have been documented and can be used in our situation. To that extent it is predictable if the context, or work situation can be described in adequate detail. Thus human behaviour can be considered to be a factor in plant operation to which we can apply a certain level of confidence, and the more that the situation can be defined, the higher that level of confidence in our prediction of the behaviour. That is what is required for safe plant operation.

So when we analyze plant events, we require specific information on the behaviour of all relevant individuals, and also information on the state of a large number of plant processes, policies, instructions, relationships, and desired outcomes.

Experience feedback/Design errors

When an analysis of difficulties in the plant indicates that design of equipment or components is central to the problem, we experience difficulty in having that fed back to effect a constructive change. When designers depart from conservative engineering knowledge and practice and attempt to be innovative, they may see the results quickly in failed component when it is put into operation. The fix may be expensive but it is a bounded problem. When the design path produces equipment that can only be operated or maintained with great difficulty, because human capabilities and limitations were not considered during the design process, then the ensuing problems are not as easily fed back for design changes. The Bruce complexity problem is an example of this issue.

Near misses.

Near misses are subject to investigation, analysis and feedback of lessons learned to operations etc. in several industries. The most widely known is aviation. It all depends on the attitude taken towards accident causation. The traditional AECB attitude is to be concerned with only those incidents that threaten the safety systems, and those that meet the criteria for being reportable to the AECB. Near misses are neither. But let us go back to the earlier comments about the multiple inputs that can be identified to an undesirable happening in the plant with a trigger action or activity that turns all these potential troublemakers into an accident. Let us suppose that they are all in place. There has been a decision made about a change in procedure but it has not been communicated properly to the line employees; there has been a turnover of responsibility in a relevant area and the new incumbent does not know an important piece of information, some maintenance has been done but the steps taken to put the equipment back into serviceable condition has been performed sloppily, and so on. None of these points by themselves can cause a notable outcome. They are of no concern to AECB inspectors. But they lie in wait for the key happening that will set up an event. But the last step may not be taken and there is a near miss. We should be aware of the structure of this event-in-waiting, and should try to ensure that it is not re-created.

Decay in managerial control.

Managers can make decisions that have immediate and observable bad consequences. But not often. The influence of management on event causation is usually a decay in managerial control over some prolonged period. It can be insidious and below the threshold of perception an awareness, even if anybody was watching. The data needed to make that judgment in a work situation are not usually being gathered nor monitored before the event. The signs are there, however, and can be brought to the attention of those willing to listen, and these listeners are hard to find. One must be aware, of course, that organizations may be subject to a constant flow of warnings of impending trouble from various sources, and may thus become indifferent to these. I agree with the several high-profile industry analysts who espouse this view of decay in managerial control, and also agree strongly that doing something about it is quite problematic.

Human capabilities and limitations

I am now going to spend some time reviewing the capabilities and limitations that people bring with them to work, and so are relevant to job performance. Knowledge of these and of other less obvious characteristics are needed to be able to review, analyze and understand what occurs in events, as well as in normal operations. (One can only understand the unusual i.e. the event, if one first knows the normal condition, from which the unusual is judged to be a deviation.)

Sensory Processes

People receive stimulation from the environment that is sensed by sensory organs. We accept as routine the incredible capabilities of these sensory devices,

and we tend to accept their limitations with equal lack of interest.
Basic point: these devices have developed through use over thousands of years to a level of functioning that is conducive to survival in a natural as opposed to man-made environment. Only in recent years, in evolutionary terms, people have been using their sensory apparatus when operating in manufactured environments with designed and narrowly constrained input and output availabilities and possibilities.

(a) Vision
Our visual sense can provide incredible detail about objects in the focus of attention, while noticing movement rather than detail in the periphery. This is a survival mechanism when predators may creep up on you. We can see a match being struck a mile away on a clear night, but with directional vision we cannot see a simple object 2 feet behind us. Information display design at the workplace has to recognize these capabilities and limitations of human vision.

Processing of information by vision is dominant both in our every day lives and in our lives at work. So most of the information that we provide to people at work is visual. We use the resolution power of central vision by arranging information so that it can be attended to without scanning. Detection of state changes, that is, changes in the information on a main display device, is an ongoing task. Searching for needed information, from a manual for example, or looking for the required information display from a panel of similar displays, is a frequent task. Vigilance refers to noting signals that appear very infrequently. Warning annunciators may only be illuminated on an exception basis. Having a whole bank of annunciators light up at once does solve the vigilance problem, but increases the problem of detecting the visual pattern and its significance. A common method of handling a large number of warning or advisory annunciator lights is to divide them into a main panel that indicates main categories of concern, and reference to a subsidiary display can provide needed detail on the selected category. The nuclear control room approach has been to have the computer prioritize the indications according to the present state of the plant, and only display the more important of these. AECL/CRL is engaged in a major research effort at present on this issue.

Visual information is displayed on several types of devices. Analog displays are the traditional scale and pointer type, usually electro-mechanical. Digital displays have been mechanical parts of electro-mechanical displays, but are now normally electronic. There are advantages and disadvantages to all of these depending on the application of their use.
Group displays are what the nuclear industry calls wall mimics, which is actually one example of a group display. These displays are used to allow group viewing of the information, so are usually confined to information of a general nature that does not change frequently. There is seldom an interactive capability with group displays.

There has to be a compatibility between the manner of displaying the information, and the manner of operating an associated control to allow interaction with the displayed information. There are cultural stereotypes that make us expect power to increase when we move a control forwards, or to the right, or clockwise, for example.

(b) Audition

Although we have detailed descriptions of the characteristics of visual objects we lack any precision in describing sounds. We may recognize a complex sound (it sounds like ...) but we can usually neither reproduce it nor describe it. But we can locate the source of a noise due to having ears on opposing sides of the head (doppler shift). We have a very limited capability for coding auditory signals and so auditory warning or advisory signals must be crude and distinctive (ambulance siren). Workplace signals must follow the same rules.

The two basic methods of communicating information is by voice speech and by written word. NPP control room staff take speech communication for granted. There are no studies done of their communication protocols, special vocabulary, and other technical conventions in speech. These are adopted in other work areas to combat ambient noise, misunderstandings, attention focuses elsewhere and so on. This type of special vocabulary is adopted by most work groups or technical and professional groupings. So it is possible that NPP operators also have their own trade vocabulary, and protocols for verbal communication, but it is not public knowledge.

(c) Other sensory systems

We have interesting sensory capabilities for vibration, since we can sense low frequency noise as structure borne vibration (earthquake). We are incapable of defining the characteristics of vibration, except in a rough qualitative way. This is associated with our sense of balance, that depends on sensory mechanisms in the ear. Get an ear infection and you may be unable to stand up. Our sense of smell is linked to the sense of taste. Frequently people taste, and so also smell, a small sample of a substance to identify it. In industrial situations this can be important sources of information. Of course smell is important if there is a fire! We also sense temperature, humidity and bodily surface cooling from air movement, that together influence our perception of bodily comfort. Because we process the most detailed environmental signals through vision, it has become more and more commonplace in man-made information processing environments to rely almost exclusively on vision. Other potential information channels can be used successfully however when we consider their omni-directional, qualitative or display-by-exception aspects that are useful as background sources.

Perception

Once the environmental data has been sensed it is filtered at a peripheral location according to complex rules that are not fully understood. The ear senses all noise that is in the appropriate range, but only a small amount of that ambient noise seems to even allowed in for perceptual processing. At the perceptual level, these raw data are associated with memory contents to attach utility or interest values to them. More data/information is discarded at this stage. You have all heard the trite phrase - selective perception. All perception is selective, by design, and cannot operate in any other way. But we can voluntarily change the selectivity of perception or attention, and often in response to changes in input signal. So if information is displayed prominently in a control room, it does not follow that operators will see it. That is, it may not be sensed, and if sensed it may be ignored, or not perceived if it seems to be of low importance or relevance to current tasks. Perception adds meaning to the incoming data stream. The control room operator can sense the same indicator change on successive shifts, can ignore the first but react quickly to the second. He knows that the contextual situations in which the signals occur are different so he reacts differently.
Perception is one of the cognitive functions. It personalizes the stream of raw data from the environment that is available to all in the same way.

Learning

Information must get into memory in the first place, so that it can be used to give meaning to sensory data. This is through learning, that is fundamental to training. Trainers teach, students learn. Learning means a relatively permanent change in performance that can be shown to be the result of experience. There are known conditions that promote learning, there are known processes that take place in learning, and we understand the mechanism of loss of information from memory over time, and the refreshing of that memory. A distinction can be made between education and training. Education is information acquisition, while training is skills acquisition. One applies learned information from an education process in the development of skills in training. But the current trend is to talk only of training, in both knowledge and skills, and the delivery of training is usually in the hands of those who know very little about how people learn, and sometimes may know little of the content material.

Motor abilities

Motor abilities are those human activities that involve manipulation of parts of the environment to effect a state change. At a simplistic level, one can claim that only when somebody handles a component of the plant is human actions of any importance. So the huge effort that has been dedicated to human error has focused on manipulation of physical items like switches and controls. Human reliability analysis have been limited to these actions, that is, to acts of commission by operators before the event. But if one does not know why the action was produced, then one cannot attribute a probability to its recurrence.

Cognition

Cognitive - the current jargon-word in this industry. Since motor activity has been addressed up to now, attempts are being made to address cognitive processes. Although there is the impression that cognitive behaviour was invented by the nuclear industry in the 1980s, and they will soon have it under control, it has long been known that cognition comprises all the various mode of knowing - perceiving, remembering, imagining, conceiving, judging, reasoning, decision-making and problem solving. Cognitive behaviour is part of almost everything that we do, that is, activities have a skills component and a knowledge component. Task analysis have to assess motor and cognitive components. Rasmussen's famous Skill/Rule/Knowledge classification of behaviour led to industry people describing behaviour in one or other of these categories. Most behaviour incorporates elements of all three. Although some consultants sell cognitive task analysis, one must consider that, if it is purely cognitive behaviour, it has no external,observable component. It is completely internalized in the individual, and not subject to objective measurement. Tasks that are analyzed have both cognitive and motor components.

Motivation

Motivation is an aspect of behaviour that underlies everything that we do, but is not accepted officially as worthy of inclusion in work organization. The loud voices of confident authority on any subject you wish to name will tell you that we cannot measure motivation, so it is not acceptable. In Canada, by law, we can only test and select individuals on the basis of characteristics that are proven to be job-relevant. So motivation and personality are excluded. But you all know that a highly motivated person can be better at a task than a person with more knowledge (or skills) but with low motivation. You all know that there are specific personality features that are appropriate for some jobs. Is it really sensible to put gregarious sociable extroverts out in the field in an NPP, where they may work for hours unseen and alone?

Small group dynamics.

There are tasks that require a team such as complex maintenance tasks. Administrative personnel work in groups, although not exactly teams. Managers have a network that is a form of team or brood. Well, maybe not a brood, since this refers to a group of beings that lay eggs. Teamwork is a skill that must be learned. Individuals come to the work life with different abilities to cooperate and coordinate in their task performance. They have to be trained to interact throughout job performance.

There is a lot known about the ways that small groups interact, depending on the tasks before the group, or the type of people in the group. There is usually some formal organization imposed on the group by the organizers, but an informal organization will often develop within the group. Leaders emerge, ideas people are identified, gatekeepers to other groups or to supervisory levels find their natural niche and so on. This is a very important aspect of NPP event analysis. The formal report will address an event based on the formal group structure, while the real sequence of events depended on the communication paths and the information focal points of the informal group functions.

Organizational behaviour.

Organizational behaviour is tricky. An organization is an idea that is shared between people who work together and cooperate to achieve a common goal. There are middle managers who will make tactical decisions on a daily basis, and there are senior managers who take the long view and make strategic decisions. Note that strategic does not mean that their thoughts are buried in the long-term future. There are many strategic decisions that are made daily and even off-the-cuff.

Stress.

Stress is a commonly used term that everybody understands but few define in the same way. This illustration shows what I mean. The common circular definition is that specific types of behaviour are noted in an individual, and these are recognized as stress reactions. Therefore the individual has been stressed by something. When we find out what, we can help in getting rid of the stress reactions. In engineering testing one may apply a stressor, and obtain an effect. In human terms, the stressor can be difficult to identify, and seldom is applied deliberately. A physical environment stress, such as high temperature, can exist in a plant, but the significance of it can not be assumed. Such stressors can have no effect, a degrading effect or can actually improve performance levels. Social stressors, such as rejection by colleagues, or having to work for a tyrannical supervisor, can only be called stressful after the fact, that is, when a degrading of performance or attitude is noted.

Performance Criteria

People's self assessment produces awareness of what they can do on a task. Subjectively one expects to be able to go so fast, and achieve dome degree of accuracy. Trade-offs are being made all the time between these two, depending on external demands or perceived importance. You may feel that accuracy is of prime importance when you are entering computer programming text on a keyboard. On the other hand, speed may the criterion if you are sending off a short ccMail message of low importance. When NPP employees are making that trade-off on the job, that perception may be different from then majority view and can result in criticism for inadequate performance. The individual's perceived trade-off was incorrect. With practice on a task the elements of the task become linked in a smooth flow, so that the motor performance of the task is faster, without sacrificing accuracy. Competent performance shows a fluency that is not evident in the performance of somebody who is under-trained or inexperienced.

Subjective Probability

There is an objective probability that can be attached to all future happenings. People use subjective probabilities in almost everything that they do, as a basis for their decision making, e.g. "I think it is going to rain today so I will carry an umbrella". These subjective probabilities are based on available hard data, as is available, has been perceived and has been retained. They are also based on a lot of personal attitudes, beliefs and experiences. So subjective probabilities have a component that is general and may be like everybody else's estimates. But they also have a large component that is individualistic, and so can be unique to the individual. We can have different people with different views on what to do in the same situation. Discussions to resolve those differing views can come close to the emotional, personal basis of the subjective probability estimation and so can be difficult to resolve objectively.

The AECH HF program

What we have been addressing so far are the factors that are important in assessing human behaviour at work. But we are at a disadvantage since we are so remote from the workplace in the stations. Our program of work includes reactive responses to requests from others (SAWRs), and also tasks that we have proactively initiated in order to anticipate or initiate activity. There are four main subdivisions of our program of activities, all having reactive and proactive internal elements and reactive and proactive external or contracted elements.

System human factors

The work already described on developing documentation on human factors regulation falls into this category. The development of the specifications for the guides documents was contracted out, and the RFP for turning the specifications into documents is now being sent out by DSS.

HPES is the Human Performance Enhancement System that was originated by INPO as a copy of the aviation accident review and analysis process. Ontario Hydro adopted it in the mid-80s and seem to have been determined to change it annually since then. Most of these changes degrade the validity and the utility of the findings from the process, so we fight strenuously to keep them on the straight and narrow path.

We have a PRD group that is responsible for event analysis review. When there is a significant involvement of human activity in the causal factors to the event, we assist by providing expert review and comment. We also try to maintain some tracking of the main trends by station, to anticipate the bigger problems that may be developing. This is extremely difficult to do. We have developed the list of important areas that have to be considered in a human factors assessment, but the utilities do not report most of the necessary data.

My group and the reliability group have been working together to develop a better approach to human reliability analysis than has been the case to now.

Unlike equipment that comes with specifications, manufacturer's data sheets, and possibly data on its use, people do not come with relevant data sheets, and the factors that can be very influential on consistency of job performance are either not identifiable or are not identified. So rather than assigning quantitative numbers to human acts, actions or behaviours, we are developing the outline of an AECB approach that will benefit from the knowledge and expertise in reliability assessment and in human factors in a cooperative manner.

Personnel oriented human factors

Work in this area we are concerned about is the effects of training, skills maintenance or selection on personnel in maintaining suitable levels of task performance on the job. It can also include aspects of the physical environment as they impact task performance.

The main point to be accepted here is that the raw material (the new employee) is put in the start and a fully competent employee takes a place in the operational unit at the end of the process. Everything in between is aimed at moulding the raw material into the form that the employer deems is necessary.

The criterion just noted was graduation form the formal training process, but that may not produce the desired products. One must analyze to demonstrate that the selection process is selecting in the good and selecting out the bad. Training must be analyzed to demonstrate that the material that is learned is what is needed on the job (it very often is not - but is believed to be so). There is a selection process in many training programs, so that not all students from a course are passed for advancing to the next level of training. Any failings in the selection or training process will be reflected in operational job performance.

Personnel programs

How does on assess the performance of station personnel when doing their tasks? Project site officers can do their rounds and assess the equipment status, but they are unable to get beyond superficial housekeeping as a measure of the people. We conducted a research project to try to identify any method being used elsewhere that could be useful to them. The answer: as expected, we cannot do anything in isolation. If we can convince the licensee to implement some new methods of performance monitoring, then we can audit their program. But the utilities that we deal with have no methods that they use to monitor personnel performance on the job. In fact, the union influence has been so strong that it would not be allowed. The airlines and the military have formal open cover inspections as the work is being conducted, and final inspection when the work is completed.

We also have been looking at the way that operators in a large, complex dynamic socio-technical system like a NPP manage to keep all the relevant items in mind. What is their permanent mental model of their operating plant, and what are the temporary versions of that model that are developed to meet temporary conditions. The utilities have a tradition of reacting to a human error incident by deciding that the procedures need to be improved. Personnel do what they are told to do in the procedure. If it is not in the procedure, and they did not do it, they cannot be blamed, even if they knew to do it by their knowledge of the plant. Procedures can never be complete in their coverage of all possible plant states.

There is current interest in converting procedures and operating instructions to computer files that can be accessed on a CRT when needed. This will be administratively easier, and should save all that money they spend on printing and distributing amendments. We have some reservations about that approach from the user's viewpoint, and there have been some serious cautions published about that by influential advisors to this industry.

We have been assessing the development of the training programs at CRL that have been rebuilt from the ground up. The criteria to be met have been outlined by the AECB. They have met them, after much critical comment and gentle guidance, and they have made a good job of it too.

Material-oriented human factors

The design of the equipment, the workstation and indeed of the whole facility can have effects on how personnel carry out their duties.

Equipment design.

At the simplest level you can recognize that switches that are installed reversed to the normal mode of operation will cause wrong operation. Switches that are unlabeled leave one wondering how to activate them. Information displays that are too cluttered make it difficult to find the bit of information that is needed. Poorly illuminated displays or controls can easily cause errors. Having to operate controls or look at displays that are located in places that are not easily accessible may not only result in them being misused, but in not be used at all.

Workstation design

The workstation includes the larger area that the person works in - the whole control room display area as compared to a single panel. It is common for people to have to time share between tasks, so that they move the focus of their attention from one specific portion of the workstation to another and there frequently is a connective link between these tasks. So there has to be a fluency in the design of the work area so that the movement of focus of attention is seamless.

Workstations can include a variety of devices such as dials, gauges, printouts, keyboards, switches, and many other types of information display or control actuation devices. There are standards that apply to the design of these as stand-alone devices, and there are also standards that apply to how to combine them into an integrated workstation. In the nuclear industry, the recommendations from TMI2 helped to focus attention on the design of main control rooms, so that workstation designs are at an acceptable level. There is a notable lack of application of these standards in the design of workstations in the rest of the plant outside of the MCR, however.

The interaction between people and computer-controlled display and control panels has generated the development of special principles of workstation design, normally collected together as human-computer interaction criteria. The critical issue in that area is the balance between people and computers in tasks assignment and in how they cooperate in tasks completion. That is, the concern is mainly with the interaction of human intelligence with the program logic that it is with the physical interaction with displayed information and keyboard design.

Facility design.
Also people have to do tasks in different parts of the whole station. Variation in design or in installation characteristics can force errors in their use. Different areas of the plant have to be like different areas in an office building. They are designed to support particular activities conducted by certain types of employees. Having the operations people and the technical support people in the plant at opposite sides of the bridge between the working plant and the clean office part does not foster good cooperation and communication. But such cooperation is in the best interests of the plant and of the AECB.

The environment of the work sites in the station are normally controlled at a suitable level for most human activities. The aspects of concern can be noise, heat, dust, radiation, chemicals, humidity, glare, etc. The research has been done and the results on the acceptable ranges for these environmental qualities have been built into the OH&S regulations controlled by provincial government. We can get into complicated areas however, when we have people doing specific critical tasks and the environment includes a mix of these qualities that are deviating from the optimum. Research on human capabilities under mixtures of environmental deviations shows that the effects can be additive, subtractive, multiplicative or can cancel each other out. If we describe these deviations as stress, so assuming that they are detrimental, we can make large errors in judgement. We have to know specifically what environmental qualities there are, and work out the combinations against the nature of the task to decide if any significant effect on human performance can be expected.

Material Program

When a new plant is being designed there are lots of areas where human factors personnel get involved in defining human requirements in plant and plant equipment design. For some time we have been looking only at modifications to existing plants. Although these may be human-machine interfaces, the utilities do not incorporate the user's needs. The Bruce-A rehab program was a test case, since it was large enough to make that input a requirement even in their eyes. We managed to convince them to prepare a HFPP but it was cut back to bare bones before we were even briefed on it. Dollars for hardware, or for making sure that the hardware can be used effectively?

There are advances being made in this industry in information automation, although it is not building on the considerable experience and knowledge that has already accumulated elsewhere. The information automation process can produce some unfortunate results if not done properly. There is an accepted way to introduce computer generated information into an existing operation. It has been developed and validated over many years in many different applications. A few years ago I cooperated with Roy Olmstead in a IAEA working group on information automation for nuclear power plants, and we got some good information into that publication. It seems that no utilities in Canada use it.

also: Workplace automation. Fuel handling. Expert systems.

Organizational human factors

It was mentioned earlier that actions at higher levels in the hierarchy have more diffuse influences on work performance. We start with an organization chart, that represents the formal structure of the organization.

This diagram (from a paper I gave in an IAEA conference 2 years ago) shows the influences on the design of the technical structure, such as maintenance requirements, human-machine interfaces, safety protective devices, etc. Independent of that, a personnel structure is designed. Different countries have different trends in personnel structure , and even our three utilities differ. Decisions are made on what shape and size the management group will be. The relationship between jobs is drawn up and supervision and command functions are established. The realities of available manning and staffing resources can modify these, but usually not very much.

The technical structure has to function, as has the personnel structure. The selection of the technical structure design determines its functions in large measure, since function is probably implicit in the design approach taken. The personnel process is not implicit in the design of the personnel structure. Consideration should be given at this stage to how to have people and computers work together to keep the plant running smoothly. That is the same function allocation process I mentioned earlier. One should not automate anything, least of all the information system, without doing a function allocation study first. Tasks are designed and the information requirements for performing those tasks are identified. All these other features should be specified at this stage, not after the plant is running. Finally, the technical and personnel structures allows the technical and personnel processes to function to generate the organizational output. All aspects of that structure and process is operating within 4 main environments: the physical, psychological, cultural and economic. If any part of this interdependent whole is modified when in operation, there can be repercussions in several different areas of the plant. The balance can be upset.

If the organizational design has been done as shown in the last diagram, then these other matters will have been settled. Responsibility, authority and accountability relationships are established; resources are allocated according to need and function; conditions of work are included in the design of the personnel process as it functions within the 4 environments; and the number and types of people needed and the career paths established for them so as to maintain a personnel flow will be established.

Personnel Subsystem

The AECB concept of human factors practice in the nuclear industry focuses on ensuring that the licensees maintain a high quality of human performance in individual and in group activities, on a continuing basis. The program that is adopted by the licensee to achieve that goal can be measured against a generic human factors program that incorporates the required components and criteria. This can best be described in terms of a Personnel Subsystem. An industrial workplace, such as a nuclear power plant, comprises two main components. These are the technical component of the hardware, software, equipment and furnishings of the physical installation; and the social component of the people who are employed in the plant. These together are often referred to as a socio-technical system. Each of the component parts is a subsystem of the larger system. The personnel subsystem is the term used here to depict the social component of the socio-technical system.

This includes:

1. people in the general population who have the basic qualities required for employment; (examples: high school grad; technical certificates; engineering degree; willing to work in Eastern Canada).
2. criteria by which these people are selected; (examples: suitability for shiftwork; aptitudes; general knowledge).
   (a) Staffing: Good selection procedures can reduce the variance in on-the-job performance by a significant degree. We cannot test people to see if they have the job skills we are looking for, since they have had no prior contact with the job, or any job in some cases. So we have to determine the kind of people that tend to be successful in that kind of work. For example, academic standing upon high school graduation has been found to be a good

predictor of success in job training for many employers. Generally, we test for skills, abilities and aptitudes that predict success, (we would like to assess potential also, but that is contrary to current human rights legislation) and we should also test personality variables but this also is unacceptable under present legislation.

3. a designed and documented process by which they are trained and retrained; (why needed: training required to provide company culture and structure: industry and company specific information; trade general information; trade specific information).
   Manning: means the employer undertakes to acquire enough of each type employee, assigned to crews or shifts, to carry out the work for each job to a satisfactory standard. In an industrial environment, there can be both overmanning or undermanning due to changing demands in specific areas or over time. So if Pickering NGS has 8 units, there are 5 operators per unit on each shift, and there are 5 crews working the shifts, then there is a manning level of more than 200 operators (some are away on training, on vacation, are sick or have other duties than operating).
4. designed and documented jobs (issues: fit within an occupational structure that meets the company's needs; provide a career structure; fit with other jobs in the process; connected to selection criteria; connected to training content).
5. reference documentation through which on-the-job performance is maintained; (what content: what to be done; when to do it; how to do it; who does it; reporting; recording).
6. hardware and software design; (important issues: can be used effectively and efficiently due to successful human-machine-task interface; evaluated to show that it does what it is supposed to do; maintainable).
7. organization and management (organization and management of people to support job satisfaction and high productivity).
8. a management structure (meets the needs of the social and the technical systems and meets corporate goals).
9. a designed and documented career progression structure (needs: milestones, requirements and pathways).

Organizational program

Hours of working has been the most important area of activity here, not by choice but in response to the licensees' initiatives. Introduction of 12-hour shifts includes aspects of management/labour relations; labour support of employees who wish to maximize their overtime income; rural locations where stretches of time off work is required to support a second job such as hobby farming; management's constant efforts to have flexibility in response to any and all eventualities; basic chronohygiene and chronobiological information that tells us how effectiveness and efficiency on the job can be influenced by different work/rest schedules and cumulative loss of sleep. It also involves the AECB in an area that traditionally was not considered to be a serious issue. And above all else, it involves the difference between normal operating conditions and outage conditions. Car dealers near NPPs sell most of their cars after an outage. The overtime money is high and the sleep levels are low. The efficiency of the personnel is open to serious question.

also: Description of the AECB activity in 12-hour shift work.

Fitness for duty.
Fitness for duty is euphemism for ensuring that employees are not suffering from the effects of indulging in proscribed substances. But there are serious difficulties faced by employers since physical attempts to control it are contrary to the Charter of Rights and Freedom (and we do not have a charter of responsibilities). We have assisted both Ontario Hydro and G2 in developing an approach that cannot be challenged under the Charter, meets the employer's need for safe employees and is acceptable to the AECB.

also: Supervision & communications

Human factors practice

The people who are employed to look at these issues originally were involved in personnel selection, and batteries of tests had been developed that assess the characteristics of people and match these with job requirements. The match was not good enough, since many still failed in training. That is costly. So what was known about human learning was applied to training programs, and separately the methods of presenting information was developed as training technology. The match could still be improved, since operational performance was below expectations. The main reason was that equipment was designed in such a manner as to force wrong behaviour. The human-machine interface was not acceptable, and training could not compensate for it. So human factors engineering developed as the study of equipment design for human use. That was a main finding from the TMI2 investigation - there had been inadequate input to the human-machine interface in the main control room.

Unfortunately, there were still inadequacies in the personnel production process. During the 70s and 80s there was a lot of work on defining the rest of the problem. And it is the most difficult. If you take an individual who performs well on selection testing, is a good student in training, and is provided with an ergonomically approved workstation, he can still be less effective on the job than we expect. The environment that he works in includes a certain type of supervision, some management influences, and an organizational climate that dictates a broad range of issues that are important to him. These effects can turn an effective employee into a problem employee.

Socio-technical systems

This has led to a focus on the people in a plant and the technology in the plant as an interacting system. Machines (except in unmanned automated systems) are only of utility if they are operated and maintained by people. They work together in a space defined by the immediate work group and the work area of other components and building features. This space is part of the larger plant facility that has a technical subsystem an a social (or people) subsystem. Everything within the outer boundary in the figure can be seen as a single interactive and inter-dependent system. A socio-technical system.

Immediately after the end to World War II, Eric Trist was among a group who founded the Tavistock Institute of Human Relations in London, England. He later became the Chairman of that Institute. In the 20 years of his association with the Tavistock, he established a new approach to the study of people in organizations. That could be called "action research". That is, instead of sitting in a laboratory conducting pure research studies of human behaviour that were later generalized, with some licence, to the real world, he conducted his studies in the workplace. This was part of the post-war reconstruction of industry, in which the Tavistock was addressing labour-management relations and ways to increase productivity without major capital expenditure. His original pioneering work was in coal mines, at the actual coal face with the miners, which looked at the work situation as a social system allied to a technical system. Trist later came to Canada and held a position with the Ontario Quality of Working Life Centre, and faculty positions in the Environmental Studies Department at York University, and in the Wharton School at University of Pennsylvania.

Let us consider the traditional view of people at work. Assumptions about people are brought into design and into operations. Often these assumptions are examples of Taylorism, named after the man who devised time-and-motion studies as a way of making workers do more in a more efficient manner. The fundamentals of Taylorism are:

1 people are unpredictable;
2 if they are not stopped by the system design, they will screw up;
3 the best solution is to eliminate people completely. Since this is seldom possible, we must anticipate all eventualities and arrange to have them dealt with.

The outcome is predictable. The familiar pattern of hierarchies of supervision and control are set up to make sure workers do what is required of them. Departments of specialists are set up to inject expertise into aspects of the process, but in reality are often providing the means to make elaborate control, measurement and information systems work. Designers of plants do know that they are working on a system that has both technical and people aspects, but invariably, they build their work around a lot of knowledge and thought on the technical side and very little on the social side.

In contrast, an organization that operates along socio-technical systems lines does not specify how tasks should be done in great detail. The standards to be reached are identified, criterion measures are established, but how it is to be achieved is specified in as little detail as possible.

The more that procedures are said to specify employee behaviour in detail, the less the employees can use discretion in doing their work. Any unanticipated occurrence, by definition, is not covered in the procedures. The procedure designer can anticipate many occurrences, and these are routine to the operators. When a situation arises that was not foreseen by the procedure designer, the employee has the choice of stopping the task to seek guidance, or of stepping out of the task and using his knowledge of the situation to decide what has to be done. A trained workforce that understands the basis of what they are doing on the job, and that is used to developing the “how” details while performing their tasks, there are fewer unanticipated occurrences.

Employees are people who are normally self-monitoring and adaptive in their behaviour in the face of changing conditions in their personal lives. (** note psychology of individual differences). At work, in the traditional hierarchy based on Taylorism, these people are required to stop being responsible, self-monitoring and adaptive. They have to follow the paths laid down by others, and when upsets occur their initiative or experience is not valued as highly as it should.

Boundaries at work can be defined in several ways. In many technical organizations, the boundaries define discrete technical processes that relate to machinery/equipment. We can think of the divisions between the Technical Section, the Fuel and Physics Section, Mechanical Maintenance, etc. These can act as stand-alone operations, within their own boundaries. They can be compartmentalized to operate according to their own rules with little concern for adjacent groups. A manager of such a group can ensure that his group does its assigned tasks, but ignore the boundary issues of information flow in and out of his group. That is, he can ignore the relations with other groups.

In socio-technical systems, the group employees accept the responsibility for much of the group’s technical work, as self-monitoring, competent and evaluative staff. The supervisor attends to the boundary issues. He ensures that the inputs and outputs for the group are properly maintained and adjusted. He negotiates with adjacent groups for solutions to identified problems. The nature of the boundaries then tend not to be predefined by artificial rules, but emerge appropriate to process and needs.

This is not blue-sky theory with no practical application. This is a way of doing business that has been working in North American industries for at least two decades.

Traditional and Socio-technical Organizations.

We can outline the main differences` between the traditional view of people at work and the new organizational approach which should, and is replacing the old way. The traditional organization was not designed to cope with rapid change, because that was not a feature of the low technology heavy industries. It was designed to be an efficient way of managing a relatively static workforce that had no expectations of advancement or change, and that was not subject to knowledge or skills development.

1. The traditional view sees employees as being complementary to the machines. Maximum specialization provides only a few components to the job to be performed, and there is little variation in tasks. This aims at employees being interchangeable components, like the machine components. In the socio-technical systems, the employee is complementary to equipment, and the job acknowledges the use of responsibility, control and the exercise of individual abilities.
2. The tradition of making tasks predetermined and based on routine procedures increases the repetitive nature of the work. With more flexibility and discretion in task performance, the individual reacts to feedback on his own performance, that is, is self-monitoring and adaptive.
3. These issues have implications for the design of training. The traditional view means that employees are trained to carry out the defined tasks as specified, and so training is limited to that narrow function. The alternative is to train in individual tasks or work modules that can be integrated under the individual's control into jobs as deemed appropriate.
4. Rather than assign jobs on a discrete task basis, work is performed by the employee in optimum work cycles. This relates to the variety in tasks and jobs that is then provided, rather than the "best way" that is predetermined for him, with the aim of reducing costs. The socio-technical systems approach creates optimum task groupings, encourages multiple and broad individual skills, and encourages employees to achieve higher level of internal rather than externally imposed control.
5. Variety in tasks is also part of the allocation of tasks between the employee and the machine. We are still faced with the assignment of tasks to machines, or to computer control, if at all possible. The employees are left with the bits and pieces that are left. Organization of work as described above opens up opportunities for development of the employee based on the job being enriched and satisfying. The talents and skills of the employees are used effectively in getting the job done.
6. Uncertainty exists, in varying amounts, at all times in the work place. The traditional hierarchical supervisory control absorbs this uncertainty upwards. Workers who have narrowly constrained jobs find it difficult to manage uncertainty or variety in work. Strict external controls have to be imposed to reduce, as much as possible, any signs of uncertainty. This requires layers of supervision plus varieties of specialist advisory staff.

> These all have the prime function of imposing and maintaining control. So we have tall pyramids in the organizational structure, that inevitably tend to develop an autocratic and paternalistic style of management. It is better to develop a measure of autonomy at all levels in the organization so that discretion can be used at the source of the uncertainty. This is more effective and less wasteful of resources and time.

The use of a flatter organizational structure, by reduction in the levels of supervision and direction, encourages horizontal as well as vertical communication. Flexible group resources are developed through this to meet environmental uncertainty. With reduction in the pyramids there is a growth in participative management, with mutually articulated groups of employees replacing the simple hierarchy.

In the traditional pyramid, each individual has to compete against his colleagues, and also defend himself against them. Groups compete with groups. System rewards, in the form of promotions or privileges, go to those who are the best at “gamesmanship”, the political games of the organization. It has been shown to be more effective to reward those who can cope effectively with the many interdependencies of the complex modern organization. Collaboration, not competition is valued. Collegiality is more productive than eliminating the perceived opposition. Success in this type of organization requires the employee to be constantly developing mutually-agreed trade-offs, in a negotiated rather than imposed order.

It is important to see that traditional structures discourage risk taking, since that can cause feelings of uncertainty in those whose function it is to keep uncertainty to a minimum. Playing safe within the rules, that reward not making waves, increases alienation among individual employees. The alternative approach reduces alienation, based on the development of commitment to the group and to the company, and of trust and openness both horizontally and vertically.

Organizational systems theory and philosophy

In order to move the organization from a traditional hierarchy to a flatter cooperative style of operation, one has to employ methods to bring about change processes. Techniques have to be used to keep the change process moving in the right direction. But there must be a firm basis for the activity in theory, or philosophy. Technical organizations tend to adopt procedures and give little attention to the reasons for adopting them. The word “philosophy” tends to generate some derision. However, activity that is not based on an understanding of the theory is merely ad-hoc intuitive trial and error. The traditional hierarchy itself is based on a solid foundation of the philosophy of management of the early part of this century. It was effective in effecting governmental policies through a bureaucracy, and this was later adopted as the way of managing large industries. But it is ineffective in managing the large complex corporations that are information intensive, that have a sophisticated and educated work-force, and that must react and change quickly in the face of turbulence and uncertainty.

Quality Improvement Program (Ontario Hydro)
You may be aware of the organizational change programs that have been introduced in the North American nuclear power industry in recent years. Florida Power and Light not only adopted such a program, but were determined to have their program win the coveted Baldridge Award for the best organizational change and renewal program in the US. After winning the award, their internal program declined drastically. That is, they adopted a process that was aimed at winning an award, not at being more efficient in conducting their business. These should be the same thing, but not so. They did not follow the basic principles of changing attitudes and values amongst their employees. They only motivated the employees to carry out certain practices that met the Award’s criteria. There was no philosophy other than winning.

Ontario Hydro bought the FPL program. They adopted it in a modified form, called Quality Improvement Program (QIP), to meet Ontario Hydro’s traditional culture and hierarchy. QIP did not challenge the hierarchical management system of Ontario Hydro, nor did it address the boundary issues in operations. It was imported as a process without any prior analysis of the Ontario Hydro problems that it was to address, so it was a shot-gun approach. Employees were led to believe that they would be more empowered to address their local problems and collectively, to address the wider organizational issues. They were very disillusioned eventually when they realized that this was not part of the QIP plan. Recommendations were developed by the teams, and were then submitted to management for approval in principle. If approved they were sent through the normal line management hierarchy for implementation. But they were lowest priority since they the local manager had no ownership of that issue. Only trivial local issues were approved for implementation by the teams themselves.

Conclusion.
We are now looking at our licensee organizations in the 1990s that are using organizational structures and functions that were developed historically to meet the needs of industry during the growth of industrialized society. That was a relatively slow-changing society based on a fixed class system. Spin-offs from technological change were limited to the elite, with only trivial incidental benefits for the employee population. We are now in an age when demands on organizations are being reshaped constantly in response to developing understandings and new perceptions of opportunities. Employers, employees, customers and the general public are well informed, evaluative and more self-assertive than previously. These organizations have to be able to produce a wide variety of responses appropriate to constant turbulence and uncertainty. They have to be self-regulating systems. They are not, and many of the technical

problems that the industry, and the regulator, has to address are the results of an organization that has not learned how to react to change and uncertainty.

The Change-resistant Organization

One author has recently discussed the change-resistant organization. This view sees the demand, or pressures, on an organization arising from the community, the customers or from internal employees, as changing more rapidly than is generally assumed. Some organizations are sensitive to changing demands and make small adjustments on a continuing basis to meet them. Others do not see the changes, remaining complacent on their conviction that they have been successful by doing what they do, so will continue to be so. Pressure builds up in response to a growing discrepancy between stimulus and response. At some point, the change in demands is recognized and there is an understanding that something must be done. Actions taken then tend to be convulsive and designed to give an appearance of decisive action in the face of an emergency. A step-change implemented by management is seldom appropriate, since there is seldom a real insight into what has gone wrong. Some superficial symptoms are addressed and the main problem may remain, creating subsequent problems.

So what makes an organization change resistant? The main answer is that the organization clings to the classic features of the traditional bureaucracy. This bureaucracy is required to coordinate many specialist activities in design, operation, and maintenance of the NPP. Bureaucracies can only function effectively by developing standard operating procedures, so that each organizational component follows set sequences of activity, which when pulled together, fulfill the extended goals of the organization. Each organizational unit has only limited discretion to vary its responses to meet situational demands. In this way the coherence of the overall function is maintained. Because of this;

1 it is difficult to shape output to meet specific demands. General efficiency is achieved at the cost of addressing individual situations effectively;
2 inertia is inevitable. The demands change, the environment changes, but the SOPs can only be changed slowly and with difficulty and high cost;
3 control is hierarchical with a top down information flow.

These 3 factors work against a sensitivity to change or to abnormal conditions. Stability is necessary, and organizational personnel guard the autonomy that they have achieved through their labours. They are usually anxious about the amount of responsibility and authority that they actually have, however, and about the ambiguities that they face in their position. (Baum, “The Invisible Bureaucracy” Oxford Press 1987). So change is threatening and is resisted.

How do we alert bureaucracies to the increasing gap between their activities and the demands of their environment? Some alerting groups that have been set up as alerting mechanisms have suffered from “bureaucratic capture” where the alerting group develops the same tendencies as the bureaucracy it is supposed to protect. The confusion in recent years : AECB bureaucratic capture : AECB upsizing and changing personnel/focus: OH downsizing : increasing mismatch and no new rules.

Categories of hazard and strategies of safety control

There is an assumption that underlies most of the analysis of accidents that has recently been challenged by several of the leading analysts. The assumption is that there is a continuum that starts with the simplest and most frequent accidents, such as slips and falls, all the way through to the accidents of the scope of Bhopal or Chernobyl. This diagram illustrates an alternative viewpoint. Jens Rasmussen (the man who devised the now famous categorization of human behaviour as skill based, rule based and knowledge based) prepared this diagram to indicate that one can isolate three different types of accidents.

The first is the simple type that I mentioned. These are high in frequency and there is very little change occurring between instances. Safety control is effected through epidemiological analysis of past accidents. Because there have been many instances of automobile accidents due to driver inattention to the road when using hand-held communication devices, their use is discouraged and manufacturers are now promoting hands-free operation devices. So this category has many instances that allows rapid assessment of the variables and a fast response in the way of redesign or change in user habits.

The second type is less frequent than the first, but still frequent enough to allow feedback of experience to benefit continued operations. Aircraft accidents are an example. When an aircraft crashes, there is an intensive investigation that isolates both the detailed design features and the system level complex problems that have been involved in the causal chain. So we have evolutionary safety control, with findings from a few but quite large accidents feeding back to change types of accidents.

The third type is the one that mostly concerns us. Nuclear power plants accidents are extremely rare, the frequency of CANDU accidents being zero as you so well know. Control of accidents sequences is based on predictive analysis, that should be updated with actual data when parts of the accident sequence actually occur. A major feature of this type of accident is that the rate of technological change between accidents can be quite large, opening the possibility, or maybe a high probability of fighting the last war instead of the next. The thinking of some industry people is still fixated on the sequence of events at TMI2 to an extent that is almost pathological. The mistakes that were made were very much a feature of that stage in the development on the North American nuclear power industry. The rate of change is high and the shape of any likely accident will be dramatically different from TMI2. But many will still be applying the lessons learned from that long ago event. The x-axis is important.

Slide 21 TMI2
But let us look at the results of the investigation of TMI2, and of Chernobyl, from a point of view that is not the dominant one. The investigation of TMI2 included an analysis performed by the Human Factors Society of America for the USNRC. Their findings and recommendations were published in a 3-volume NUREG. These included many fundamental issues, and to an extent conventional wisdom on how to manage the technology of people in organizations. This slide notes some of the points raised. In both these accidents there was a discrepancy between the capability and the responsibilities assigned to operators. That is, when a task or job is assigned to a type of employee, they have to be provided with the knowledge and skills to do that job, as defined by a proper task analysis. This was not done. This is still not done.

Both investigations noted unsatisfactory management and technical staff performance. The HFS report to the USNRC made recommendations that would have addressed this deficiency had they been implemented. They were not. There are still many examples to support the position that this criticism is still valid. In the TMI2 analysis, the design of the interfaces between the plant systems and the operators who had responsibility for them were inadequate in several ways. This was an area that could be addressed without threatening anybody too much. Improvements were made to control room design, but only in terms of the actions recommended. As I noted earlier, with no understanding of the theory behind the actions, there is no lasting improvement in this area. It is a one-shot upgrade that is followed by a reversion to bad practice. That is what we have seen, e.g. Darlington, advance CR designs.

Many of the recommendations that should have been implemented after TMI2 were lost in the deluge of technical detail that was produced by almost every group that wrote on the accident. The wood disappeared in the obsession with studying the knots in the trees.

Chernobyl is difficult because most of what we know about that accident derives from badly written and almost illegible reports that were processed by our analysts. Closure is a common human cognitive function, and I suspect that there has been a great deal of closure at work in the analysis of Chernobyl. Closure refers to the filling in of gaps in a story or findings from an analysis according to what seems to be highly probable as the intervening pieces. It is the basis of gossip, and of the party game of a story being passed from individual to individual until a quite unrecognizable story returns to the originator. However, accepting what we are given, there seems to have been a large organizational and managerial component to the causal chain. The culture emphasized production over safety. Requirements to approach central agencies for approval to take certain actions resulted in delays and subsequent confusions in operating control. Head office and field site personnel responsibility and authority levels were confused and inappropriate to the situation. But the analysis in this case appear to have been lacking in detail. All that we see is the generalizations and system level issues, unsupported by adequate detail and analysis.

The Technocratic Bureaucratic Paradigm
suitable for characteristics of industrial society

The technological imperative
- man as an extension of the machine
- man as an expendable spare part

Maximum task breakdown
- simple narrow skills
- poor handling of uncertainty
- need for external controls
- tall, pyramidic organization
- autocratic management style
- vertical communication dominant

Competition, gamesmanship produce rewards
- intentionally selfish (organization)

Alienation
- low risk-taking
- low trust, suspicion

The New Organizational Paradigm
suitable for characteristics of information society

Joint optimization principle
- man as complementary to machine
- man as resource to be developed

Optimum task groupings
- multiple broad skills
- uncertainty handled by flexibility
- self-regulation / internal controls
- flat organization
- participative management style
- vertical and horizontal communication

Collaboration, negotiation produce rewards
- wider purposes (members, society)

Commitment
- innovation
- high trust, openness

The Evolution of Socio-technical Systems"
E.Trist. Ontario Ministry of Labour,,June 1981