

# The Safety of Ontario's Nuclear Power Reactors

## **A SCIENTIFIC AND TECHNICAL REVIEW**

Vol. 2 Appendices

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Ontario Nuclear Safety Review  
Toronto, Ontario  
29 February, 1988

# **THE SAFETY OF ONTARIO'S NUCLEAR POWER REACTORS**

## **Appendices**

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**ONTARIO NUCLEAR SAFETY REVIEW**  
**TECHNICAL REPORT**  
**APPENDICES**

**Preface**

These appendices contain seven detailed elaborations of matters covered more superficially in the Technical Report. They have been written by well-known authorities, or by the professional staff of the Review. They are essential supplements to the condensed material of the Technical Report.

Several of the appendices contain detailed recommendations. Some of these have been incorporated into the Review's overall conclusions and recommendations. Others stand alone, as the opinions of the appendices' authors. I am in broad agreement with most of them, but have preferred to leave them within the authors' material. I hope that they will be given detailed study by appropriate bodies, especially Ontario Hydro and the Atomic Energy Control Board.

**F. Kenneth Hare**  
Commissioner

## Appendix I

### Ontario Hydro's CANDU Nuclear Stations: An Outline of Safety-related Design Aspects

by  
D.A. Meneley

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## A. Introduction

1. Development of the existing nuclear-electric energy system in Ontario took place over a period of approximately 30 yr. This development was a leading contributor to evolution of the mature technology of the Canada Deuterium Uranium (CANDU) systems that exists today. As is to be expected in any evolutionary process, each operating station shows evidence of the stage of the development at which major decisions were made for that particular station.

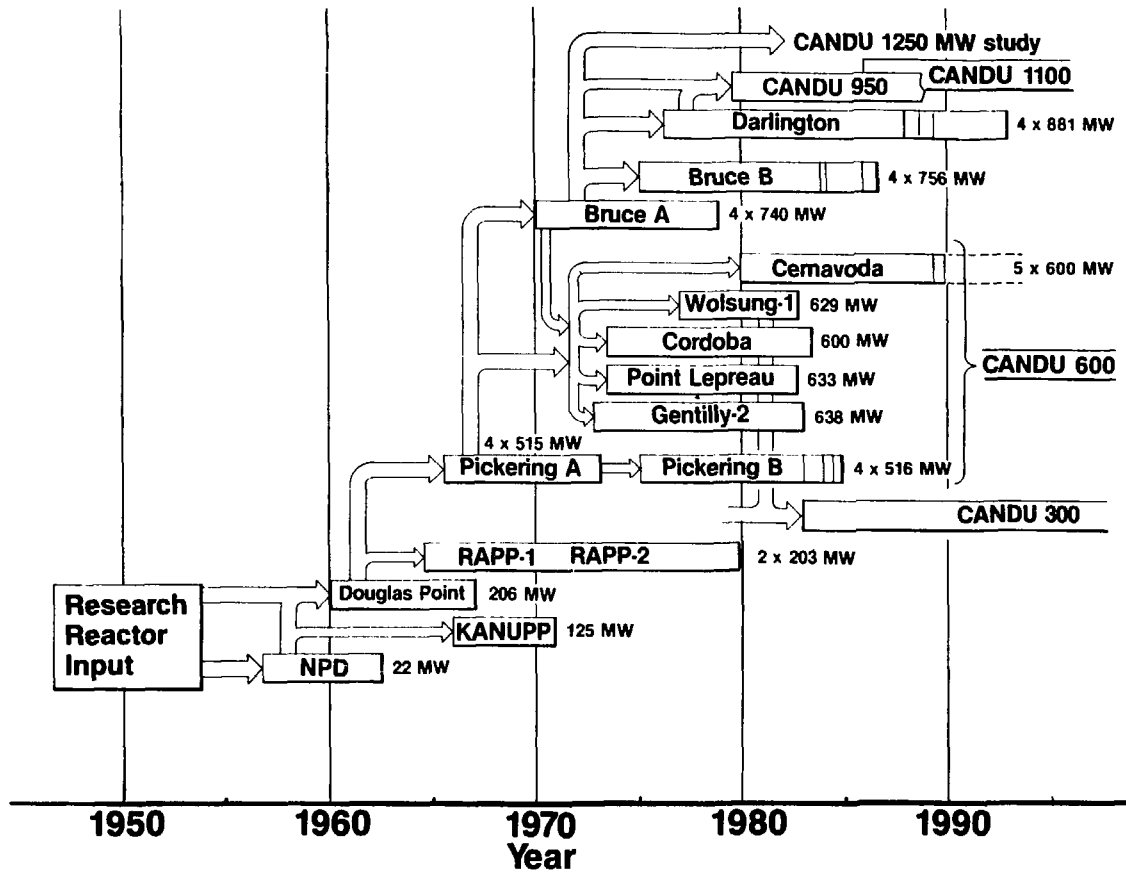
2. This document provides a general outline of the main features of Ontario's CANDU stations and describes the similarities and differences between individual stations. Only general concepts are described; the intent is to provide a brief introductory document to aid in understanding the most important safety features of Ontario's nuclear stations. Using Pickering B and Bruce B as reference stations, the text and diagrams illustrate the major safety-related aspects of the five existing stations--Pickering A, Pickering B, Bruce A, Bruce B, and Darlington. A brief comparison of the CANDU system with the US light-water reactor (LWR) system concludes this appendix.

## B. General Concepts of Multi-unit CANDU Plants

### *(a) Design evolution*

3. Evolution of the CANDU energy system (Figure I-1) began with the 22-MWe (megawatt electric, net output) plant designated Nuclear Power Demonstration 2, or NPD2. NPD2 succeeded an earlier design, NPD1, which utilised a pressure vessel. All subsequent CANDU reactors are derived from the NPD2 design. The NPD2 plant was designed with the clear recognition that the eventual need of Ontario Hydro was for a much larger plant; this was the reason for adoption of the pressure tube design. The inherent physical characteristics of a full-scale natural uranium fuelled, heavy water moderated power reactor result in it being a large size. The pressure vessel system used in LWRs was not practical for a large reactor--aside from the fact that such

Figure I - 1



The Development Sequence of the CANDU Nuclear Energy System

vessels could not be fabricated in Canada. (The first guiding principle of the development was that the whole system should be Canadian-made.) NPD2 entered service in 1962 and is now being decommissioned because of material degradation and unfavourable economics.

4. The second step in CANDU evolution was a scale-up in power by a factor of 10 to the 206-MWe prototype unit at Douglas Point, on the Bruce Nuclear Power Development site. This plant incorporated most of the features of the current Ontario Hydro plants, except that it was built as a single unit. A great deal was learned from this plant about the systems that would be needed to operate full-scale nuclear stations economically and safely. First in service in 1967, Douglas Point is now being decommissioned because of material degradation and unfavourable economics.

5. The first commercial stations were the four 515-MWe Pickering A units, again representing a scale-up by a factor of 10 in total power. The unit size was dictated by the electrical grid limitation on the maximum loss of generation and by the judgement that this size of reactor represented a prudent maximum given the knowledge base existing at that time. The choice of four identical units was determined by the power needs and the important economic savings achieved by series construction relative to single-unit systems. At this point, a new containment concept was developed in response to concerns by the nuclear regulatory agency, the Atomic Energy Control Board (AECB),\* over proximity of the Pickering site to Metropolitan Toronto. The vacuum containment system provided added assurance of containment integrity at the same time as allowing a reduction in the peak design pressure of the containment buildings relative to that necessary in single-unit containment systems. (For example, the design pressure of the Atomic Energy of Canada Limited (AECL) CANDU-600, single-unit station containment building is 224 kPa compared with 141 kPa in Pickering.) The choice of a vacuum building design was basically the result of an economic optimisation; with four units being built, the extra cost of the

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\*In this appendix, "AECB" refers to the organisation; "Board" refers to the five-member board.

vacuum building was more than compensated by the savings in the structural costs arising from the lower building design pressure.

6. The next major step was development of the 740-MWe Bruce concept (later uprated to the equivalent of about 840 MWe). This concept was chosen in response to the desire to reduce capital costs to compete with coal-fired generation, as well as to recognise the larger unit size allowance of the expanding Ontario Hydro grid. An additional factor was the need to reduce radiation doses to station staff, which, while within regulatory limits, were unacceptably high at the Douglas Point unit at that time. This led directly to the close-in design of containment, in which as much of the equipment as possible is arranged outside the main containment envelope for easier access during maintenance and emergencies. (The careful attention paid during this same period to reduced worker exposure through materials selection, maintenance procedures, arrangement of equipment, system chemistry control, and other innovations reduced the dose rates to operating staff by a factor of at least five in all newer CANDU stations.) The close-in design of containment also allowed separation of the heavy-water systems from the ordinary-water/steam systems, as well as recovery of heavy water that leaks from the heat transport system without mixing with ordinary water. The heat transport system design was modified to permit smoother reactor power manoeuvring during operation, the need for which was anticipated by Ontario Hydro's power system planners. Another factor that led to adoption of the Bruce design was the desire for some degree of diversity in CANDU stations. Although confidence in the concept was not yet backed up by operating experience, very large commitments were essential to meet the rapidly increasing electrical load.

7. These two station designs provided the basis for subsequent developments in the Ontario Hydro system. The Pickering B station was originally conceived as a repeat of Pickering A, and the Bruce B station as a repeat of Bruce A. The Darlington design originated under the designation of "Improved Bruce." Several incremental improvements were made to the heat transport system, control systems, fuel handling, and used-fuel storage. A third concept was developed but never built: this was to incorporate 1250-MWe units. An

improved Pickering concept also was developed but not built. A complete genealogy of CANDU development is shown in Figure I-1, including the relationship between Ontario Hydro stations and other CANDU units.

8. In parallel with, and during the same period as, this rapid design evolution, safety design concepts also evolved rapidly. The 1972 AECB Siting Guide, which was first applied to licensing of Pickering A, required that a complete set of accident analyses be done under the assumption that a serious process system failure (such as a loss of coolant accident [LOCA]) had occurred, and that one of the three special safety systems (shut-down, emergency coolant injection, containment isolation) had not responded. For the case of a large LOCA plus failure of a shut-down system, this logic required analysis of a rapidly rising power level followed by shut-down due to fuel movement and subsequent mixing of fuel and moderator water.

9. Such analysis was found, in the course of the preparations for Pickering A, to be difficult to quantify with a high level of assurance. The analysis was speculative. Even though Pickering A was judged to be acceptably safe by the AECB, designers made the decision at this time to include two shut-down systems in all future plants. The position argued by designers was that, provided these two systems could be designed to be independent and diverse, the Siting Guide logic would not then require analysis of accidents involving loss of reactor shut-down. This position was accepted by the AECB. Their decision showed that the underlying philosophy of the Siting Guide recognises the low probability of occurrence of, for example, a loss of coolant combined with simultaneous failure of two shut-down systems--with a probability in the range of one per billion years. Bruce A was the first station to incorporate two fully capable and independent shut-down systems. Analysis recently completed by Ontario Hydro staff and corroborated by the Safety Division of Argonne National Laboratory shows that loss of shut-down following a major LOCA does not result in containment failure.

10. The second major development in safety systems design during this period arose from experimental work initiated to establish the functional capability of

emergency coolant injection systems (ECISs). These experiments showed that, under extreme conditions (the so-called "stagnation" pipe break), water could be prevented from refilling some fuel channels quickly enough to prevent failure of fuel sheaths and consequent release of fission products to the containment space. Such a release indicates loss of one of the barriers of the "defence-in-depth" strategy used in safety system design. These findings were obtained very late in the licensing process for Bruce A; Ontario Hydro subsequently made a commitment to upgrade the design of ECISs in all its nuclear stations. The upgrade involves installation of high-pressure, high-flow injection systems. Units 3 and 4 of Pickering A are the only ones still awaiting this upgrade, which will be completed in the near future.

11. Several other minor changes in requirements were introduced during this period, such as a separate reliability requirement on reactor regulating systems and on electrical power systems, and much more extensive examination of the response of process and safety support systems during accidents. None of these changes represented any fundamental alteration of the Siting Guide logic, but rather a progressively stricter interpretation of its provisions. Many of the differences in detail in present Ontario Hydro stations can be traced to this evolution in safety system requirements; many system upgrades and backfits to operating stations have been incorporated in recent years. These hardware changes were accompanied by a gradual tightening of the specific requirements on design, construction, and operation of all stations. For instance, the quality control and inspection requirements on high-pressure piping are much more stringent today than they were at the beginning of the programme, in recognition of the potentially serious consequences of sudden failure of such piping.

12. The general features of the Pickering and Bruce designs will be described, using Pickering B and Bruce B as reference stations. The Darlington design will be referenced only in cases in which it differs significantly from Bruce B.

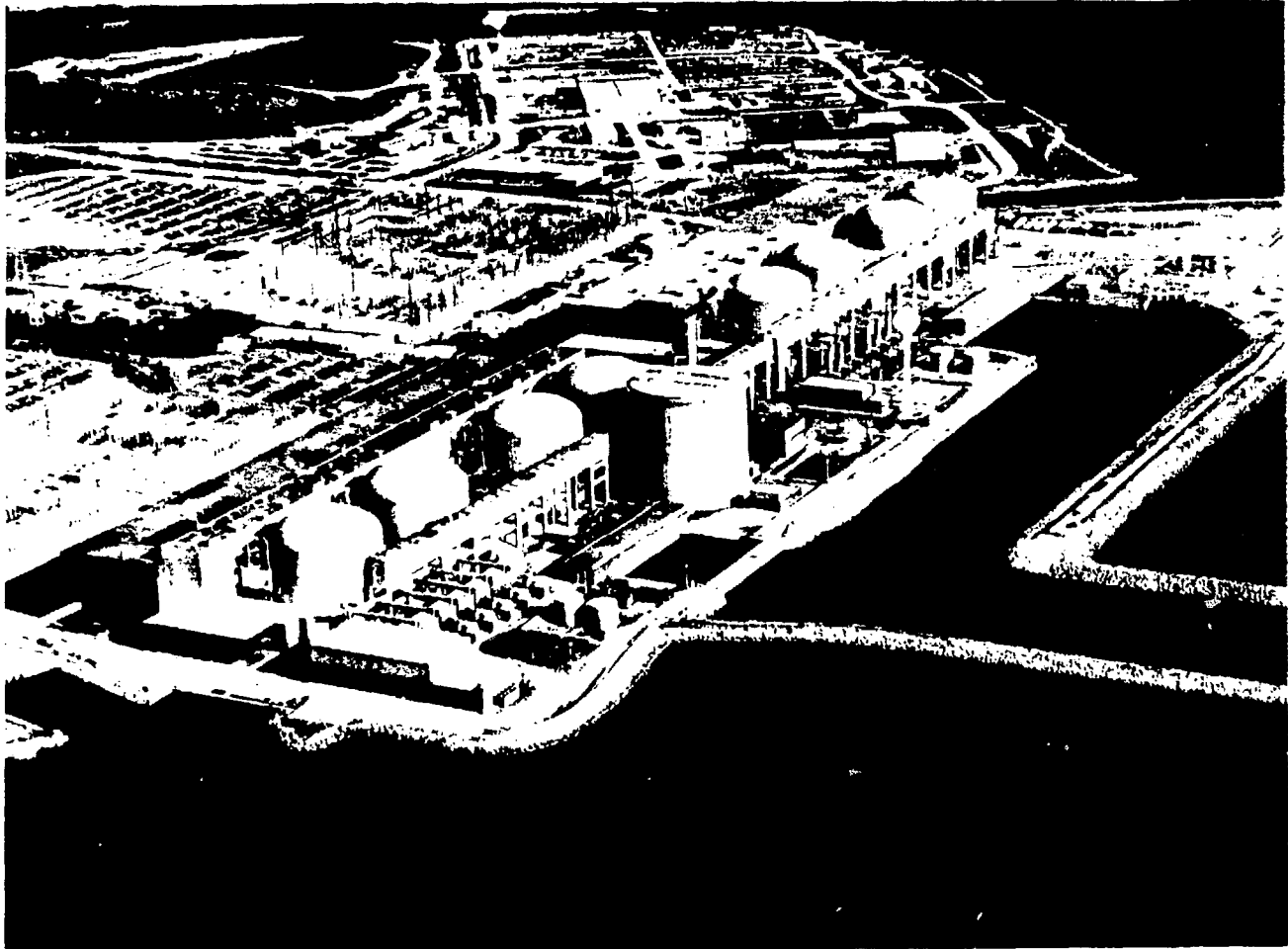
### **(b) The Pickering concept**

13. The design of Pickering was derived directly from the Douglas Point station. Other than the increase in unit power, the main innovations were (i) use of a vacuum building common to all units; (ii) a common control room for the four units; (iii) two independent reactor heat transport circuits; and (iv) a common used-fuel storage bay, with fuel transfer mechanisms to move used fuel from each unit to the storage bay. This reactor concept was developed at the same time as the AECB Siting Guide, and Pickering A was the first station licensed according to its rules. The general features of the Pickering station as it exists in 1987 are shown in Figure I-2.

### **(c) The Bruce concept**

14. This plant originally was designed with four units, each with electrical output of 740 MWe. Output was later increased by adoption of a new fuel design, so that steam could be supplied to heavy-water plants on the Bruce site. Major innovations relative to Pickering were (i) fuelling machines that travel on rails under the reactors and service all units, carrying fuel to and from the common central fuelling area; (ii) a single-circuit heat transport system utilising feedwater pre-heaters separated from the steam generators; and (iii) adoption of the close-in containment design. Bruce A was the first station to use a completely independent second safety shut-down system. The general features of the Bruce stations are identical; Bruce A and Bruce B are shown in Figure I-3.

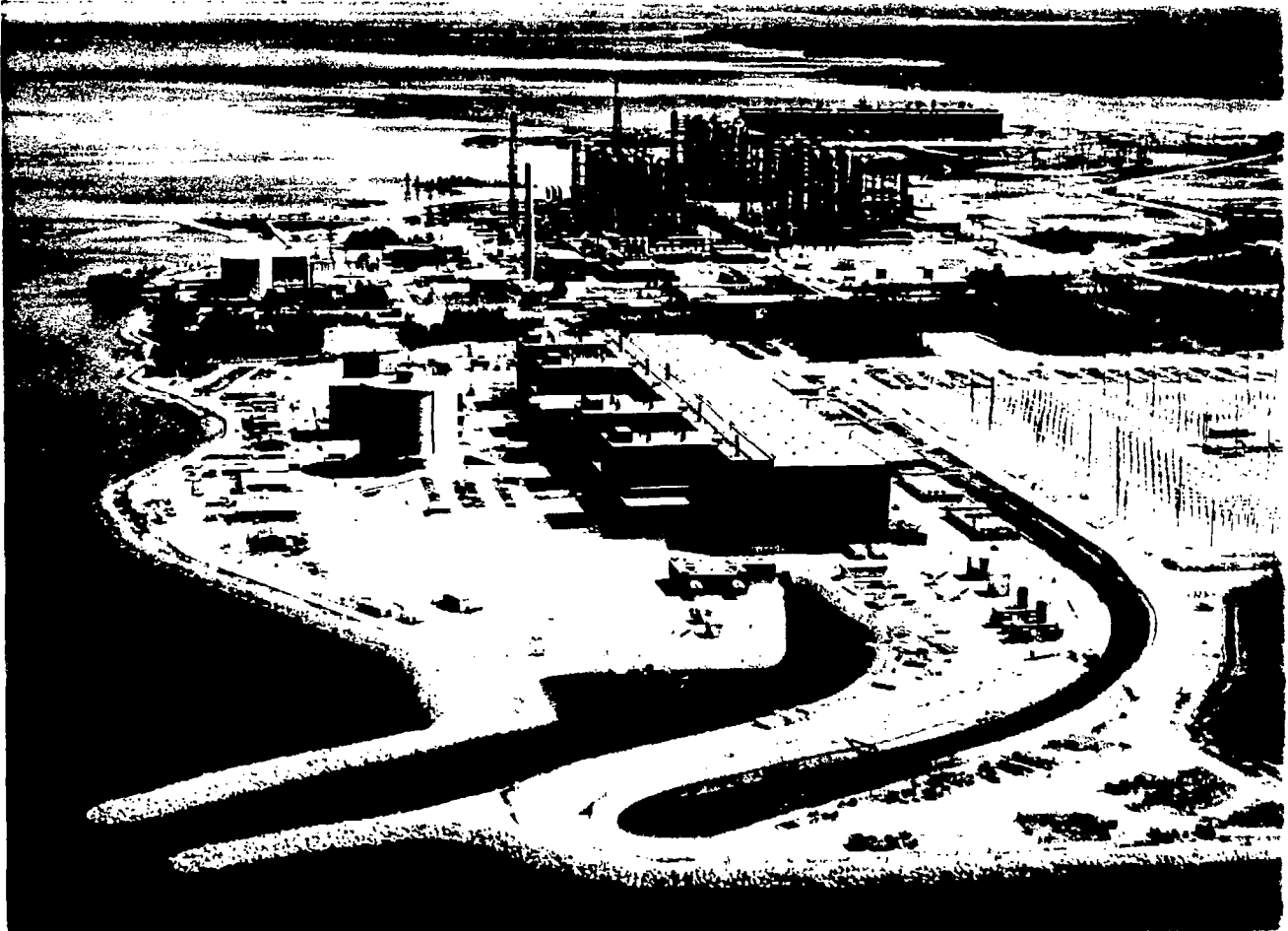
15. Darlington retained most of the major design features of Bruce B, with the exception of the primary heat transport system, which reverted to the two-loop layout of Pickering (and of the AECL 600-MWe units) with four large pumps and steam generators, as shown in Figure I-4. The general features of the Darlington station are shown in Figure I-5.



The Pickering Nuclear Station Site on Lake Ontario

AECL

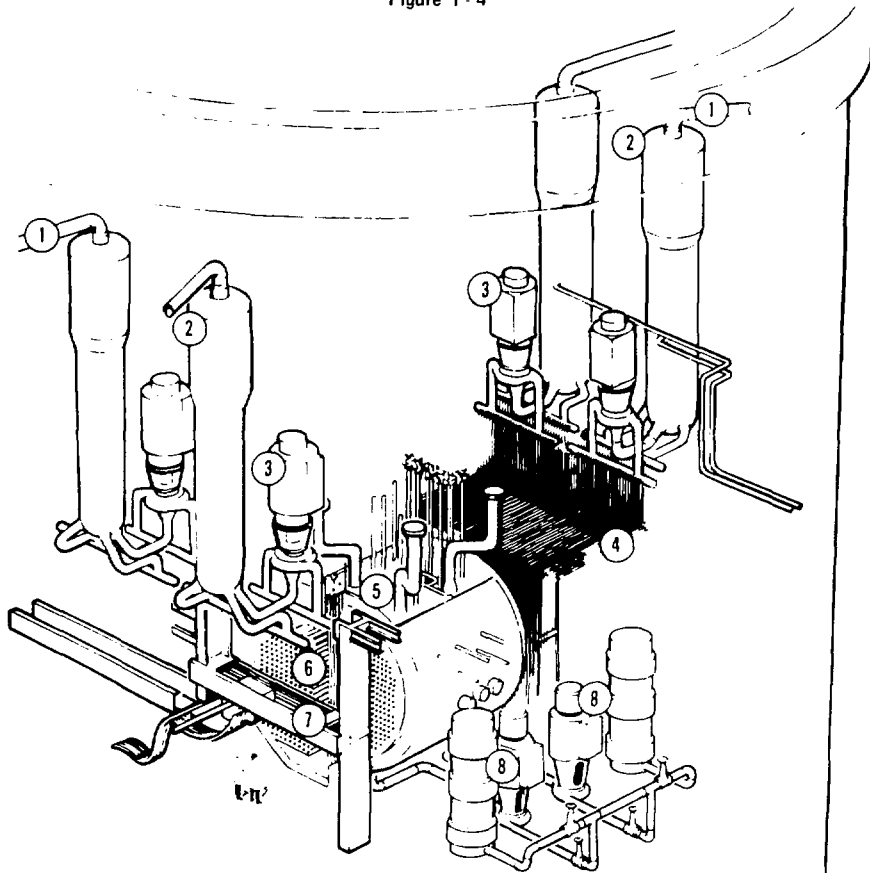




The Bruce Nuclear Station Site on Lake Huron  
- Bruce B in Foreground, Bruce A in Background

AECL

Figure I - 4

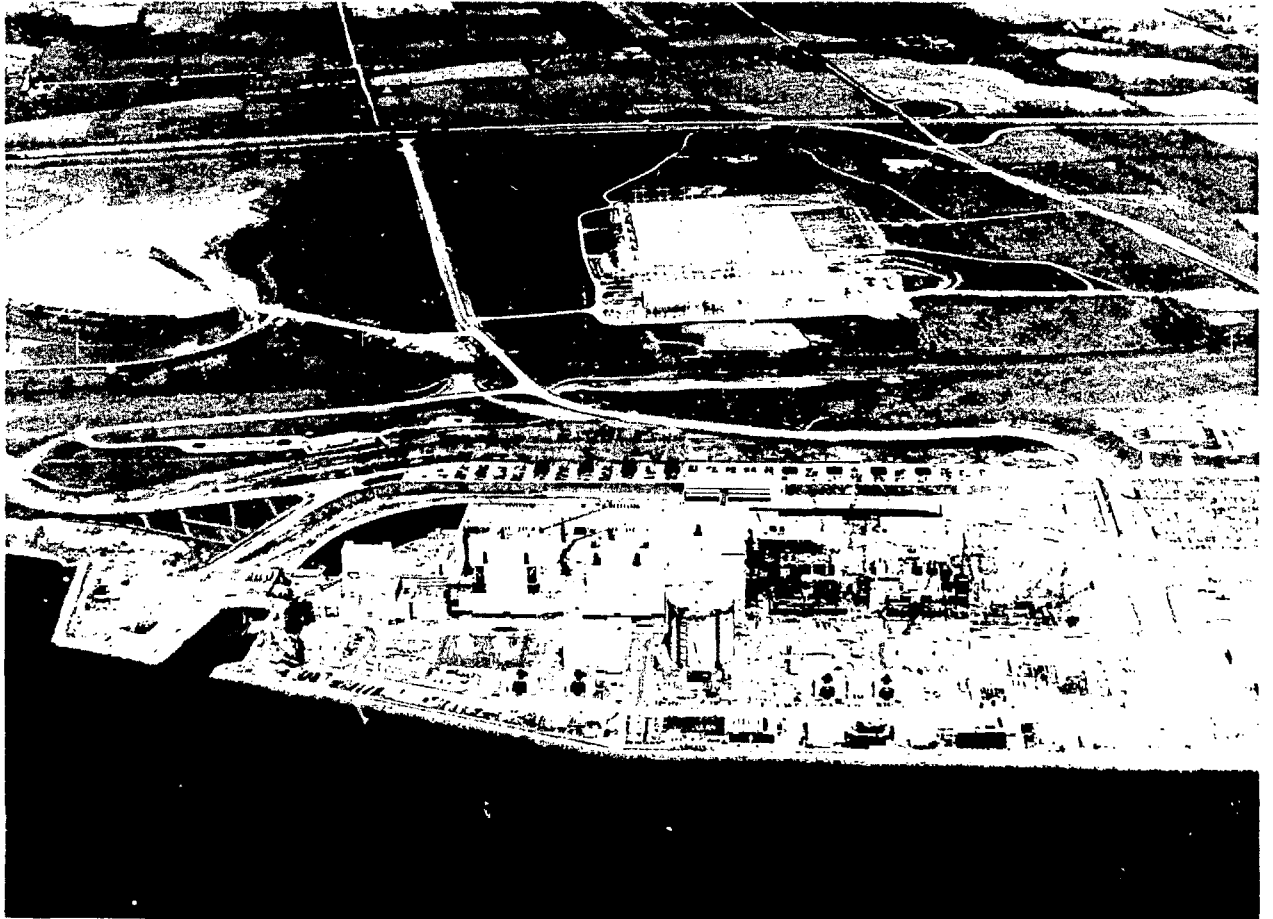


- 1 MAIN STEAM SUPPLY PIPING
- 2 STEAM GENERATORS
- 3 MAIN PRIMARY SYSTEM PUMPS
- 4 FEEDERS
- 5 CALANDRIA ASSEMBLY
- 6 FUEL CHANNEL ASSEMBLY
- 7 FUELLING MACHINE BRIDGE
- 8 MODERATOR CIRCULATION SYSTEM

Typical Layout of Reactor and Steam Generating Systems

AECL

Figure I - 5



The Darlington Nuclear Station Site on Lake Ontario

Ontario Hydro

### C. Nuclear Steam Supply System

16. The thermal fission reactor known as CANDU contains only two basic components--natural uranium fuel and heavy-water moderator. These components are arranged so as to produce a self-sustaining chain reaction in the uranium fuel. It is this reaction that produces the heat energy that is the *raison d'être* of a power reactor. The heat is used to boil water so that steam can be supplied to a conventional steam turbine.

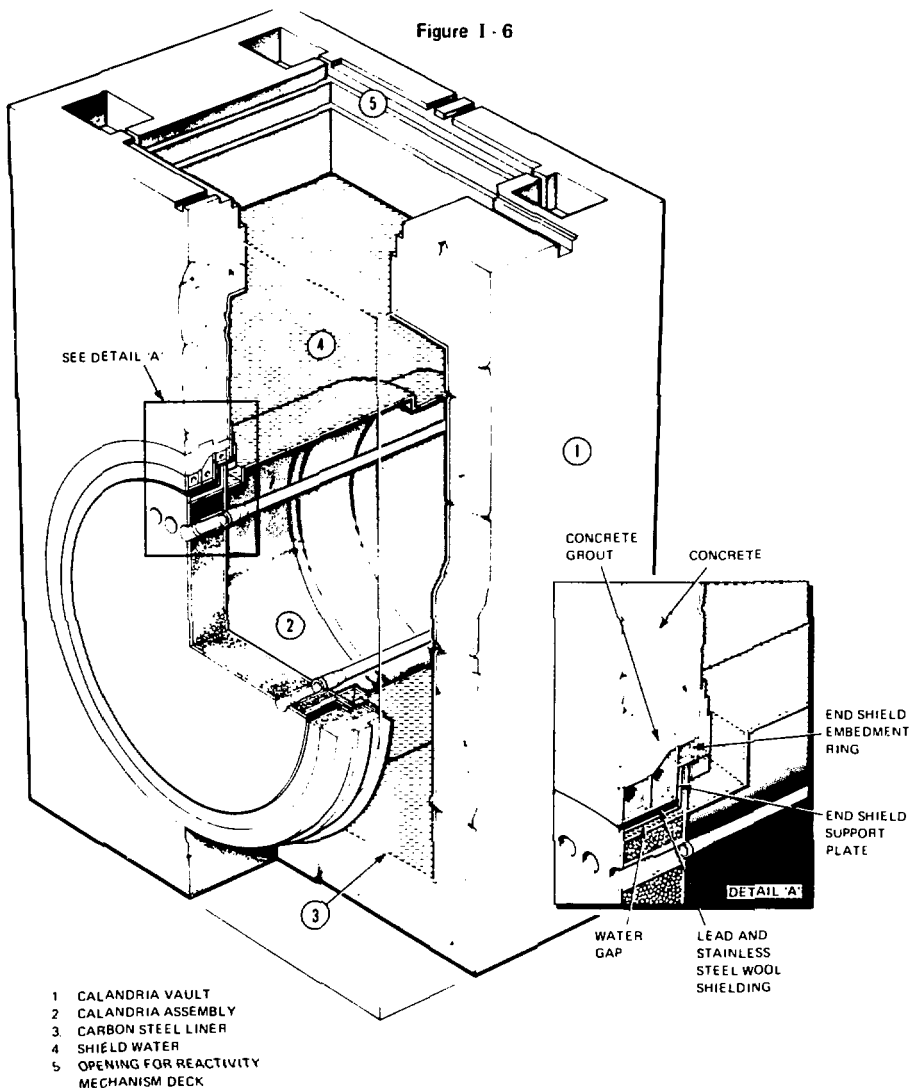
17. In order to control the process of heat generation, all reactors must have a reactor regulating system. To remove the heat from the reactor, a heat transport system is required. To generate steam to spin the turbine in a CANDU reactor, a steam generator system is required. Finally, to replace fuel as it is used, a fuel handling system is required. All of these systems are subsystems of the Nuclear Steam Supply System.

#### (a) Reactor and auxiliaries

18. The reactor consists of a horizontal cylindrical tank about 7 m in diameter and 6 m in length, as shown in Figure I-6. Zircaloy-2 (an alloy of zirconium with 1.5% tin, 0.12% iron, 0.1% chromium, and 0.05% nickel) calandria tubes are attached to a tubesheet at each end of the reactor, thus forming a square array of through-tubes about 14 cm in diameter, on a 28.6-cm spacing. The array of calandria tubes defines the region into which fuel is placed--the so-called core of the reactor. Each individual calandria tube is called a fuel channel. The moderator water fills the calandria tank and surrounds the outside surfaces of the calandria tubes.

19. During reactor operation, some heat is produced in the moderator water by collisions between the water atoms and both neutrons and gamma rays. The moderator water is circulated through heat exchangers to remove this heat and maintain the operating temperature at about 70°C. Ion-exchange columns are used to maintain water purity and acidity level (because water purity slowly degrades as a result of corrosion of piping and tanks in the moderator circuit).

Figure I - 6



Typical Layout of Calandria, Tube Sheets and Calandria Tubes

Ontario Hydro

The helium cover gas on the moderator tank is circulated through catalytic recombiners to prevent build-up of deuterium and oxygen produced by neutron radiolysis of moderator water.

20. Surrounding the calandria tank is one form or other of shielding material, whose purpose is to stop the escape of neutrons and gamma rays from the reactor and so to protect the operating staff outside the shield. The shields are sufficiently thick to allow staff access to the faces of the reactor during shut-down.

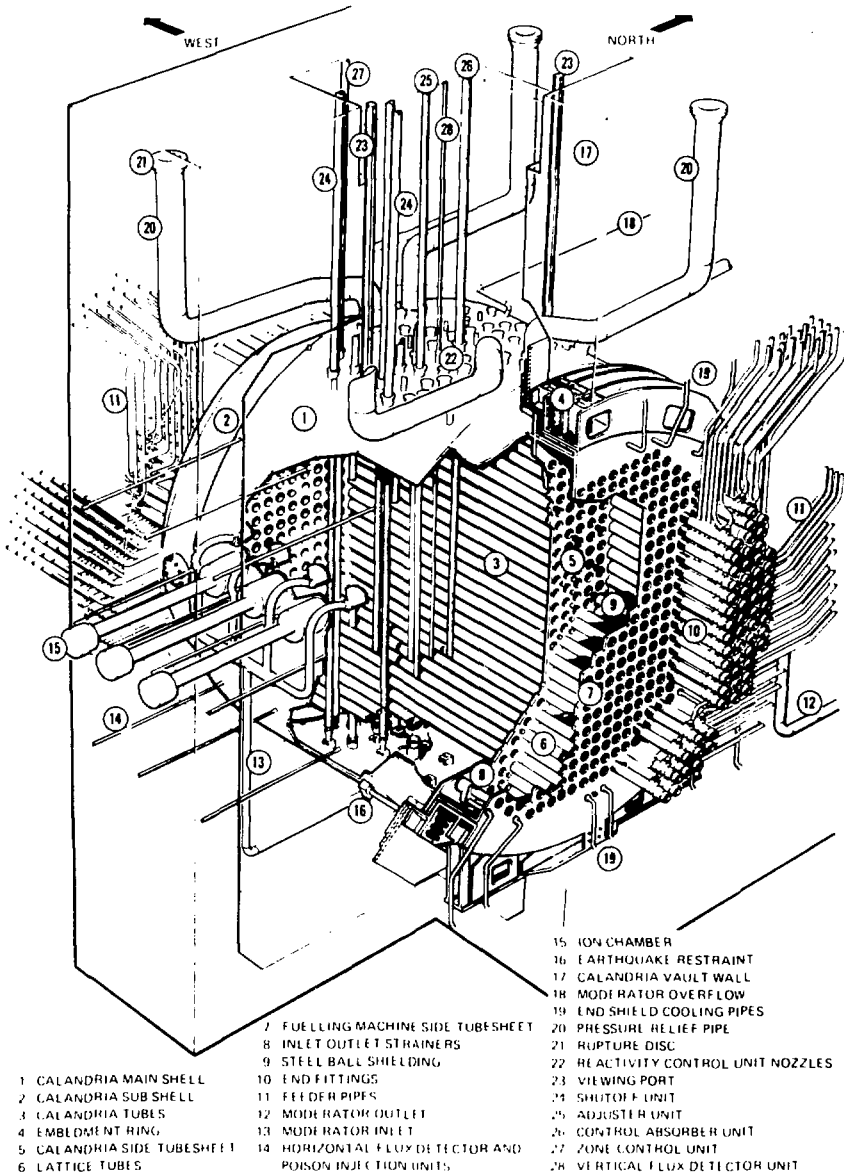
21. Pickering B has 380 horizontal fuel channels. There is a channel-free zone inside the calandria tank around the outer perimeter of the core. This zone is called the radial reflector. It reflects thermal neutrons back into the fuelled zone to increase the fission rate near the core outer boundary.

22. The Pickering A reactors contain 390 channels compared with 380 in Pickering B--an instance of the additional design margins used in this earlier plant because of uncertainties in the heat removal capability of the fuel, which were later resolved. The major difference between the Pickering A and Pickering B reactor designs is that the earlier version uses moderator dump as part of the emergency shut-down system instead of the poison injection system adopted for Pickering B. In fact, the Pickering B reactor is the same as that of the AECL 600-MWe design, with the exception of its fuel.

23. The Pickering B and Pickering A reactors are illustrated in Figures I-7(a) and I-7(b). The large tank under the Pickering A reactor is the moderator dump tank.

24. The Bruce reactors are identical to those at Pickering in channel spacing and reflector geometry, but the reactor core contains 480 channels. The major difference between the Bruce A and Bruce B reactors is that the earlier design utilises booster rods that are inserted to overcome xenon poisoning following power reduction. Booster rods contain 93% uranium-235. Bruce B and all other Ontario Hydro reactors use absorber rods called adjusters that are normally in

Figure I - 7(a)



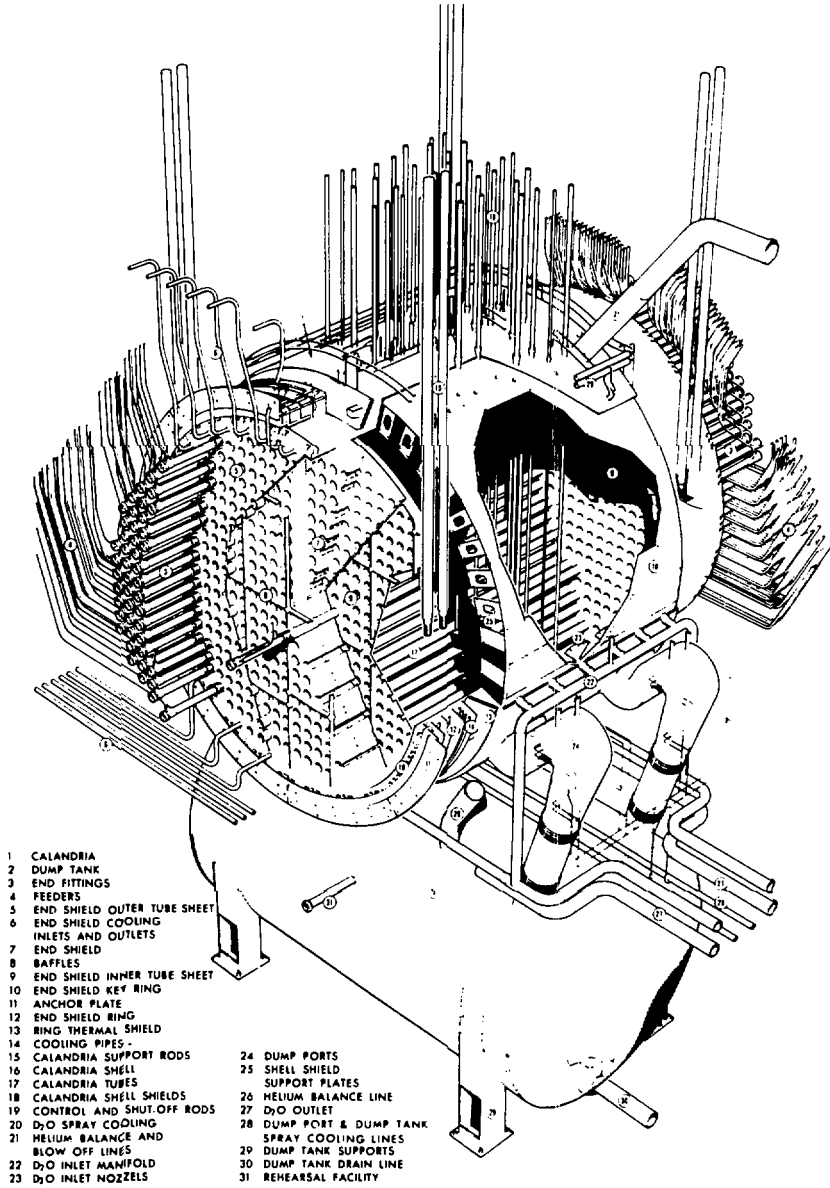
- 1 CALANDRIA MAIN SHELL
- 2 CALANDRIA SUB SHELL
- 3 CALANDRIA TUBES
- 4 EMBLEMENT RING
- 5 CALANDRIA SIDE TUBESHEET
- 6 LATTICE TUBES

- 7 FUELLING MACHINE SIDE TUBESHEET
- 8 INLET/OUTLET STRAINERS
- 9 STEEL BALL SHIELDING
- 10 END FITTINGS
- 11 FEEDER PIPES
- 12 MODERATOR OUTLET
- 13 MODERATOR INLET
- 14 HORIZONTAL FLUX DETECTOR AND POISON INJECTION UNITS

- 15 ION CHAMBER
- 16 EARTHQUAKE RESTRAINT
- 17 CALANDRIA VAULT WALL
- 18 MODERATOR OVERFLOW
- 19 END SHIELD COOLING PIPES
- 20 PRESSURE RELIEF PIPE
- 21 RUPTURE DISC
- 22 REACTIVITY CONTROL UNIT NOZZLES
- 23 VIEWING PORT
- 24 SHUTOFF UNIT
- 25 ADJUSTER UNIT
- 26 CONTROL ABSORBER UNIT
- 27 ZONE CONTROL UNIT
- 28 VERTICAL FLUX DETECTOR UNIT

Pickering B Reactor Assembly

Figure I - 7(b)



Pickering A Reactor Assembly,  
Showing Arrangement of Dump Ports and Dump Tank

Ontario Hydro



the reactor and that are removed to overcome xenon poisoning. In Bruce B and most other CANDU units, the adjuster rods consist of cobalt-59 pencils arranged in bundles on a central rod. The cobalt-59 absorbs a neutron, producing cobalt-60, a radioactive isotope.

### **(b) Primary heat transport**

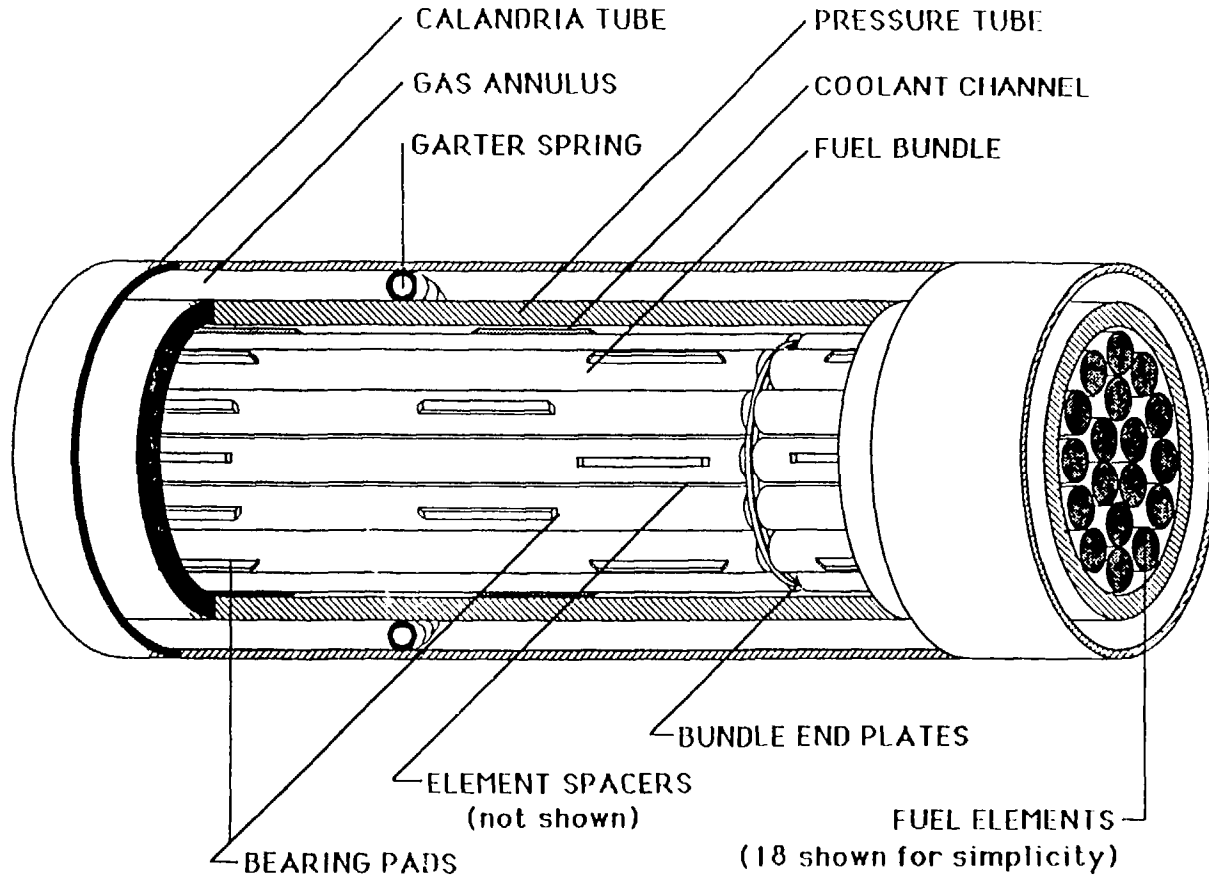
25. Pressure tubes are inserted inside each calandria tube and insulated from the calandria tubes by a gas gap containing low-pressure carbon dioxide, as shown in Figure I-8. Pressure tubes are made from an alloy of zirconium with 2.5% niobium. The pressure tube internal diameter is about 10 cm, and the wall thickness is about 4 mm. This wall thickness is sufficient to contain a water pressure of 10-11 MPa (1400-1500 pounds per square inch) inside the pressure tubes, with a tube stress less than one-third the ultimate strength of the material. In this pressure range, the saturation temperature of heavy water is 290-310°C, sufficiently high to produce 4.0- to 4.7-MPa steam for the turbine.

26. Heavy water is pumped through each pressure tube via a header and inlet feeder pipe arrangement and collected through an outlet feeder pipe connected to an outlet header. The heavy water then passes through the steam generator tubes, where heat is transferred across the tube wall to ordinary water at a lower pressure, causing it to boil. The heavy water is then returned to pump suction pipes and inlet headers and distributed to the inlet of fuel channels through individual inlet feeder pipes. The temperature at the reactor inlet is about 250-270°C.

27. The heat transport heavy water is heated by the fuel bundles located inside the pressure tubes in the reactor core region by forced convective heat transfer. Most of the heat produced in fission is released within the uranium dioxide fuel, inside the 1.2-cm-diameter fuel elements. The remainder of the fission heat is released in moderator water and structural materials.

28. Pressure in the heat transport system is maintained either by controlling the rate of inflow and outflow of heavy water (Pickering) or by controlling

Figure 1 - 8



Arrangement of Components of a Single Fuel Channel

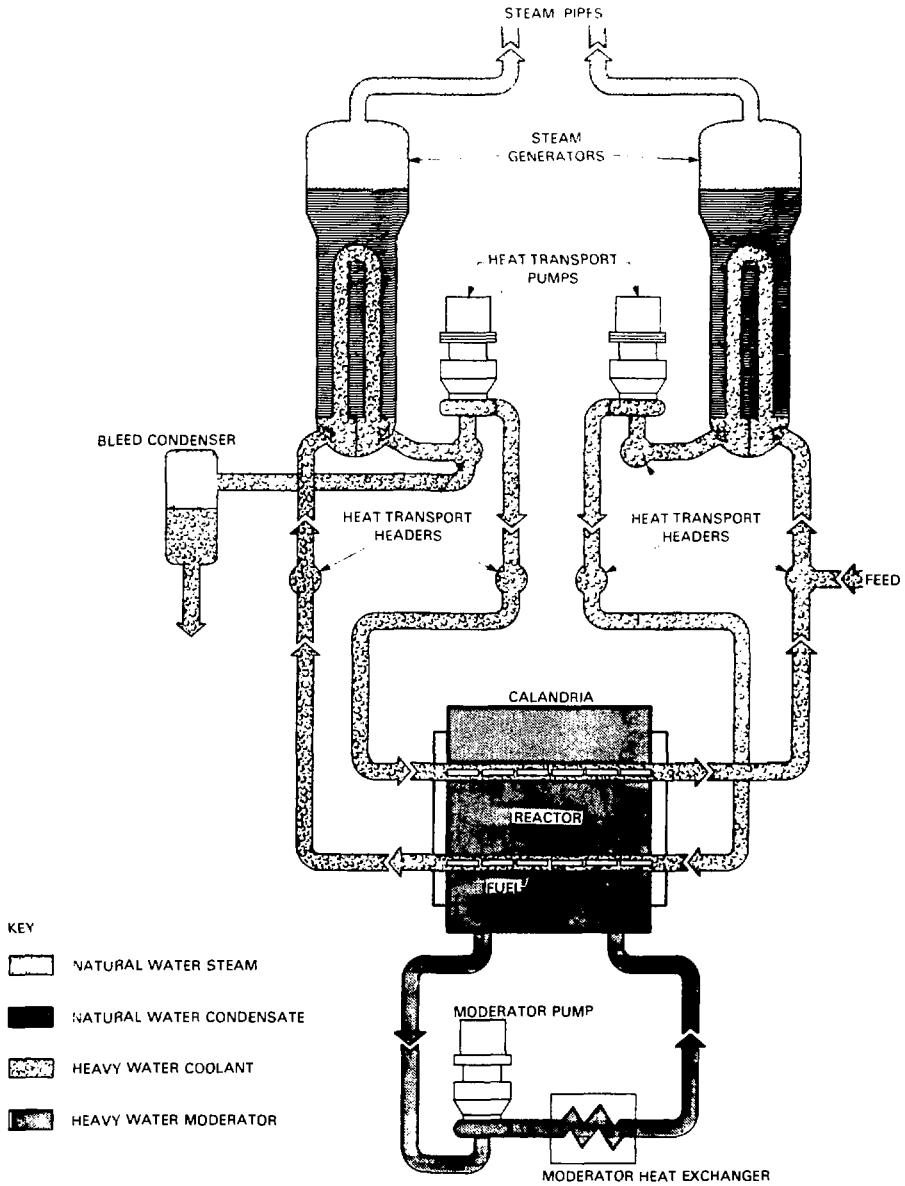
steam pressure and water level in a closed tank called a pressuriser, which is connected to the heat transport piping (Bruce). Leakage from this high-pressure system is compensated by addition of heavy water from a storage tank; most of the leaked water is recovered in liquid and vapour recovery systems. Ion-exchange columns maintain water purity; hydrogen is added in small amounts to ensure a low free-oxygen concentration (to prevent corrosion), and lithium hydroxide is added if necessary to maintain alkaline conditions in the heat transport water for the purpose of corrosion protection in all parts of the heat transport system.

29. The large pumps that force heavy water around the closed heat transport loop are driven by electric power produced by the unit generator or from a separate transformer connected to the station switchyard.

30. The Pickering heat transport system consists of two separate figure-eight loops, each connected to 190 fuel channels. Channels on each side of the vertical centreline of the reactor face are connected to one loop. A single pass of the figure-eight loop consists of three pumps in parallel (plus one stand-by pump), inlet header, 95 channels connected to the inlet header with coolant flowing in one direction through the reactor, an outlet header, and three steam generators with integral pre-heaters. The second pass of the loop is identical, with coolant flow through the remaining 95 channels of the loop in the opposite direction. (The original reason for this piping arrangement was to minimise the heavy-water inventory in the loop; in large reactors, this advantage essentially disappears.) In each loop, the channel flow direction alternates in a checkerboard pattern on the reactor face. The two loops are interconnected at the outlet headers by small-diameter piping to maintain a pressure balance between the loops. The general concept of the figure-eight arrangement is shown in Figure I-9; Figures I-10(a), I-10(b), I-10(c), and I-10(d) show the main components of the Pickering heat transport system in more detail.

31. The first two reactor units of Pickering A originally used Zircaloy-2 pressure tubes. All subsequent CANDU units use the zirconium - 2.5% niobium

Figure 1-9



Heat Transport Systems – General Arrangement

Ontario Hydro

Figure 1 - 10(a)

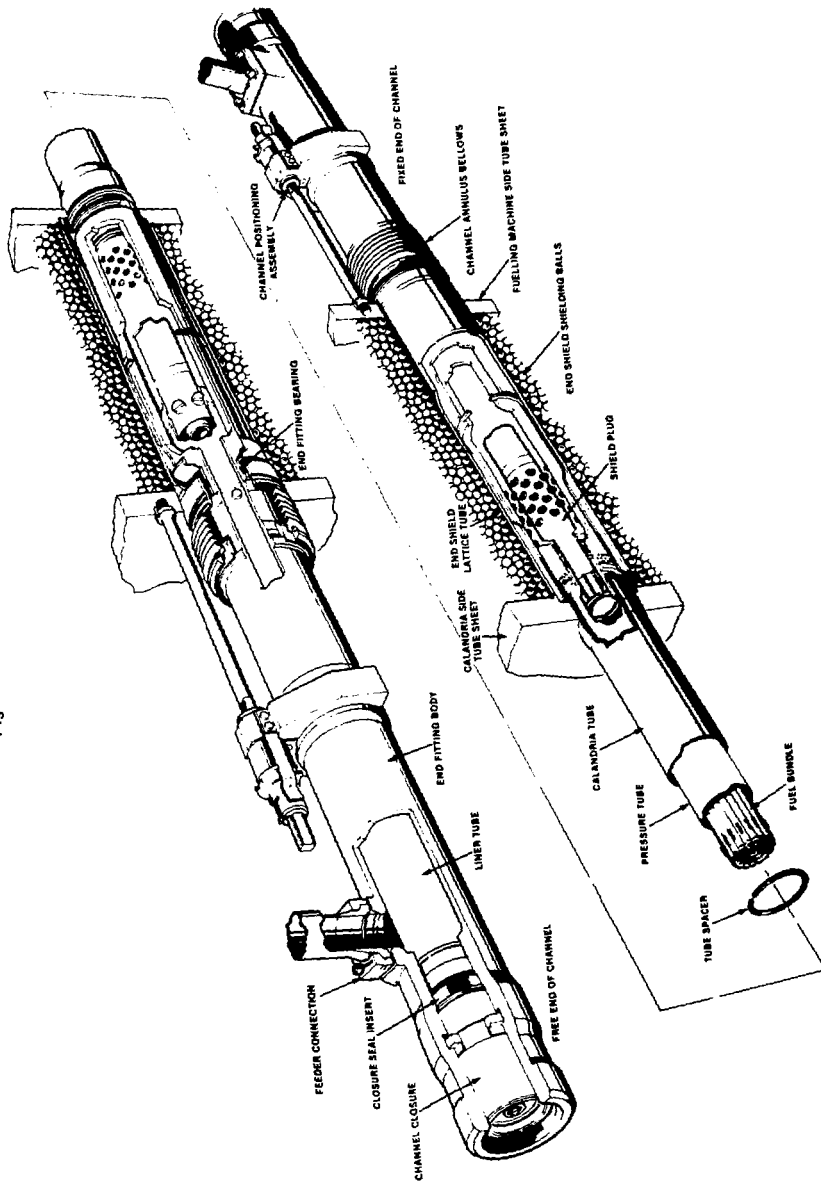
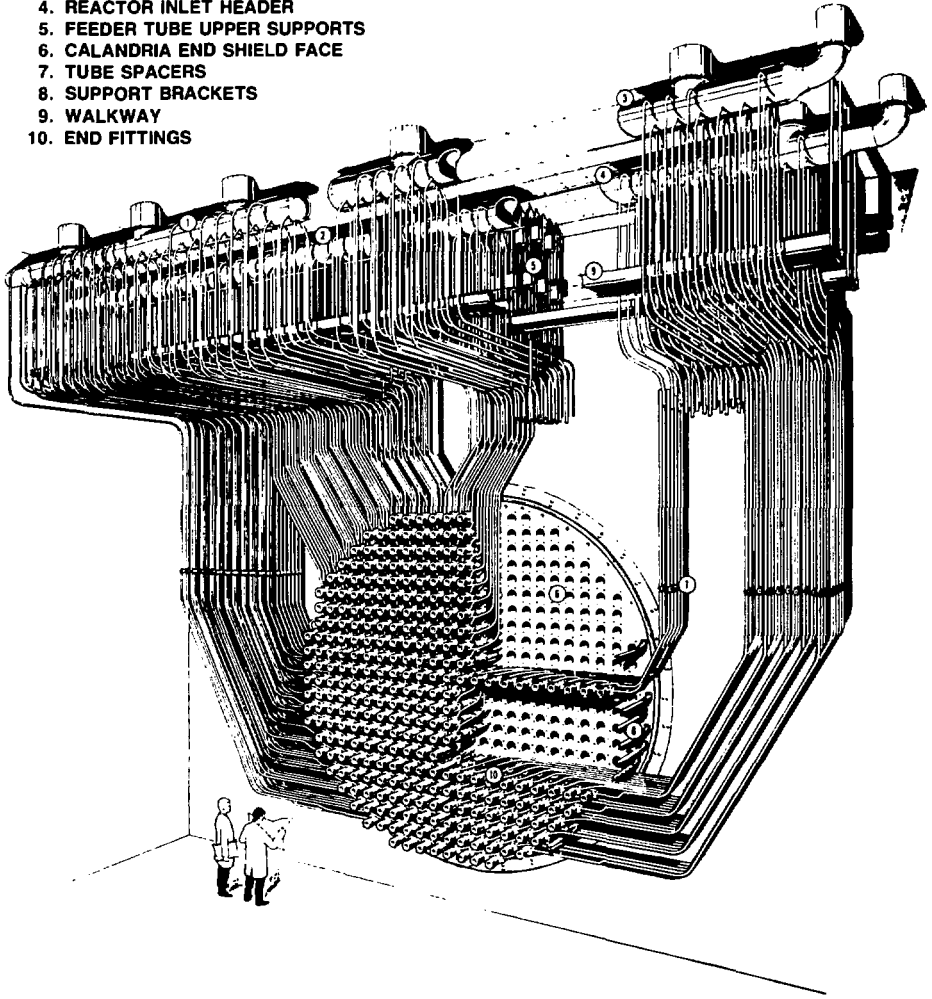


Figure 1 - 10(b)

1. REACTOR OUTLET HEADER
2. REACTOR INLET HEADER
3. REACTOR OUTLET HEADER
4. REACTOR INLET HEADER
5. FEEDER TUBE UPPER SUPPORTS
6. CALANDRIA END SHIELD FACE
7. TUBE SPACERS
8. SUPPORT BRACKETS
9. WALKWAY
10. END FITTINGS



Pickering B Feeder Layout on Reactor Face

Ontario Hydro

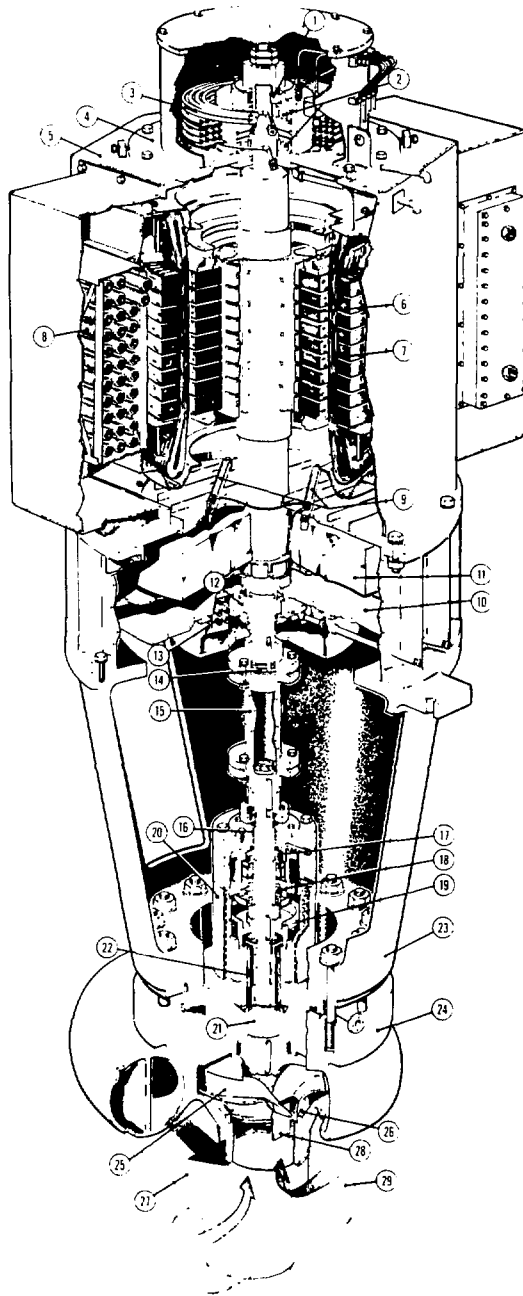
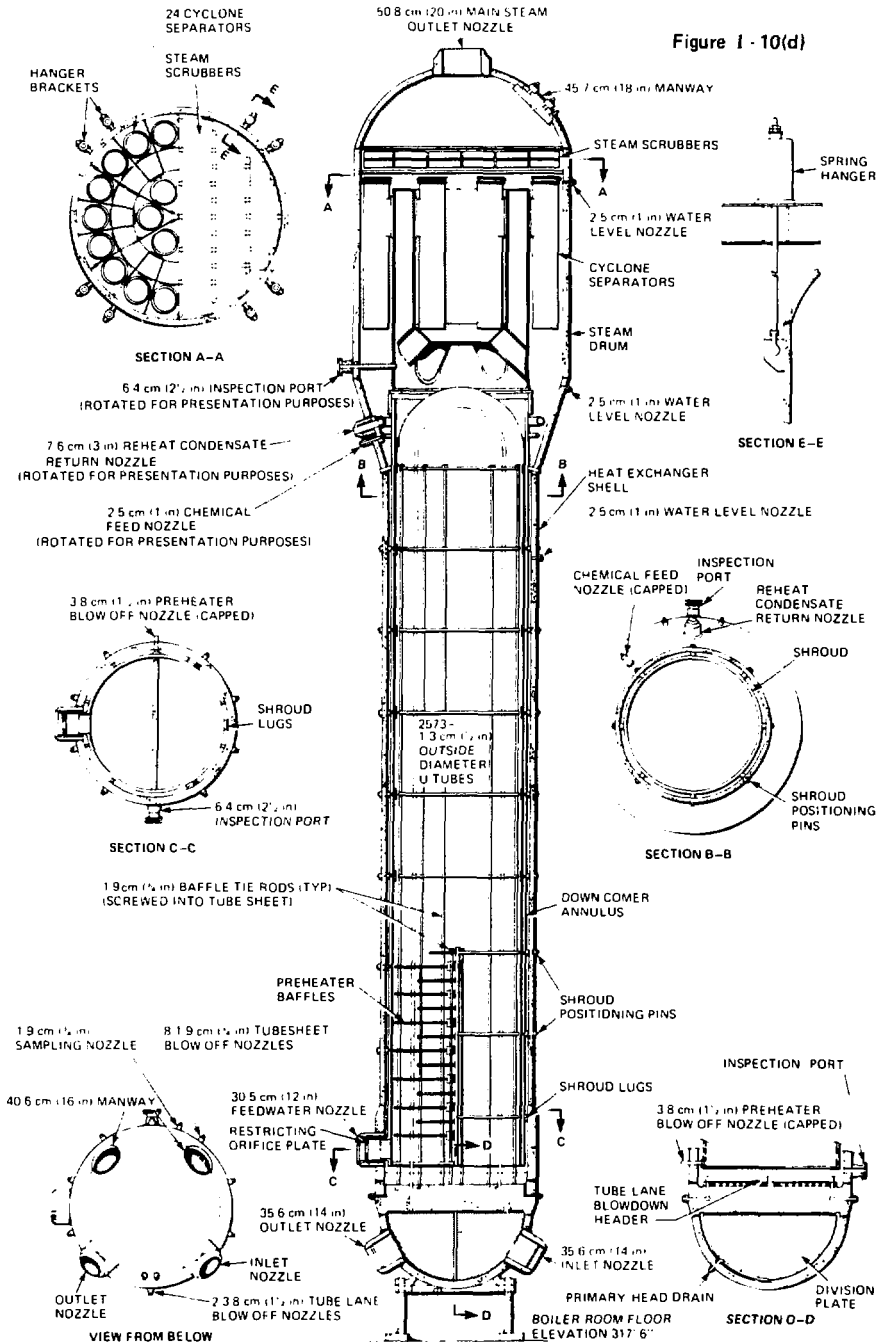


Figure 1 - 10(c)

- 1 UPPER GUIDE BEARING
- 2 THRUST BEARING
- 3 BEARING COOLING COILS
- 4 UPPER BEARING BRACKET
- 5 MOTOR FRAME
- 6 ROTOR
- 7 STATOR
- 8 STATOR COOLERS
- 9 SPACE HEATER
- 10 LOWER BEARING BRACKET
- 11 FLYWHEEL
- 12 LOWER GUIDE BEARING
- 13 BEARING COOLING COILS
- 14 THRUST DISC
- 15 SPACE COUPLING
- 16 VAPOUR CONTAINMENT SEAL
- 17 SECONDARY MECHANICAL SEAL
- 18 PRIMARY MECHANICAL SEAL
- 19 RECIRCULATION IMPELLER
- 20 PUMP COVER
- 21 THROTTLE BUSHING
- 22 PUMP BEARING
- 23 MOTOR STAND
- 24 PUMP CASE
- 25 PUMP IMPELLER
- 26 CASE WEAR RING
- 27 PUMP DISCHARGE ELBOW
- 28 ANTI-ROTATION VANE
- 29 SUCTION TEE

Pickering B Heat Transport Pump

Ontario Hydro



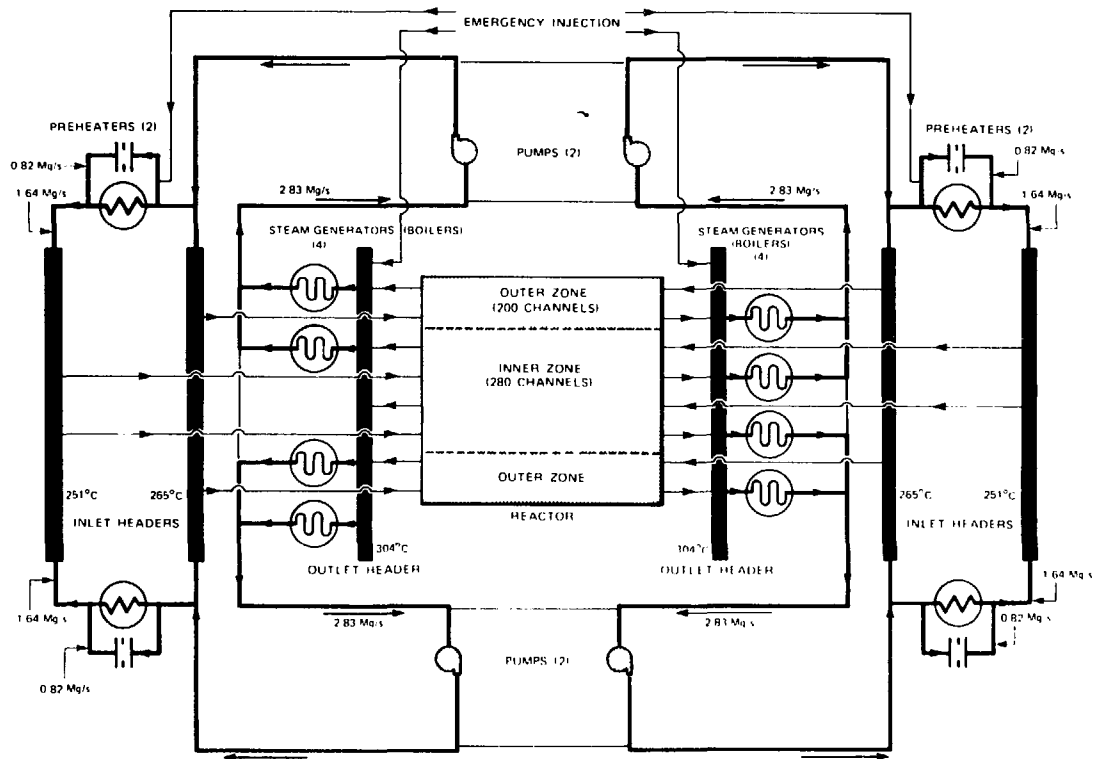


alloy. This newer alloy was not sufficiently proven for general use at the time of commitment of Pickering A units 1 and 2. Tests done at Chalk River Nuclear Laboratories established its suitability for service in time for commitment to Pickering A units 3 and 4. The advantages of zirconium - 2.5% niobium are higher tensile strength and improved corrosion resistance. Following the failure of the Zircaloy-2 pressure tube in Pickering Unit 2 in August 1983, both units 1 and 2 have been refitted with pressure tubes made of zirconium - 2.5% niobium. Unit 1 has now been returned to service; Unit 2 is expected to restart in 1988.

32. The Bruce heat transport system was originally conceived as a ring header system, in which an inlet header encircles the core. All 480 fuel channels are connected to this header. Four heat transport pumps deliver water from the outlet of the steam generator to the inlet header. The inlet header distributes cooling water to half the reactor channels. The other half of the channels are fed from the ring header at the opposite end of the reactor core. A second unique feature of this system is that the pump outlet flow is split: half of the flow is directed through a separate pre-heater and inlet header. The pre-heater raises the temperature of ordinary water entering the steam generator. The cooler water exiting from this pre-heater is used to cool the highest-power channels at the centre of the reactor. The other half of the pump flow is directed to an inlet header to which the lower-power channels around the periphery of the core are connected. The Bruce design evolved during the detailed design phase of Bruce A; the ring header concept was abandoned by capping the pipes that originally connected the two pump inlets and outlets at each side of the reactor. In addition to these piping changes, a pressuriser was added to the heat transport system to improve its capability for power manoeuvring.

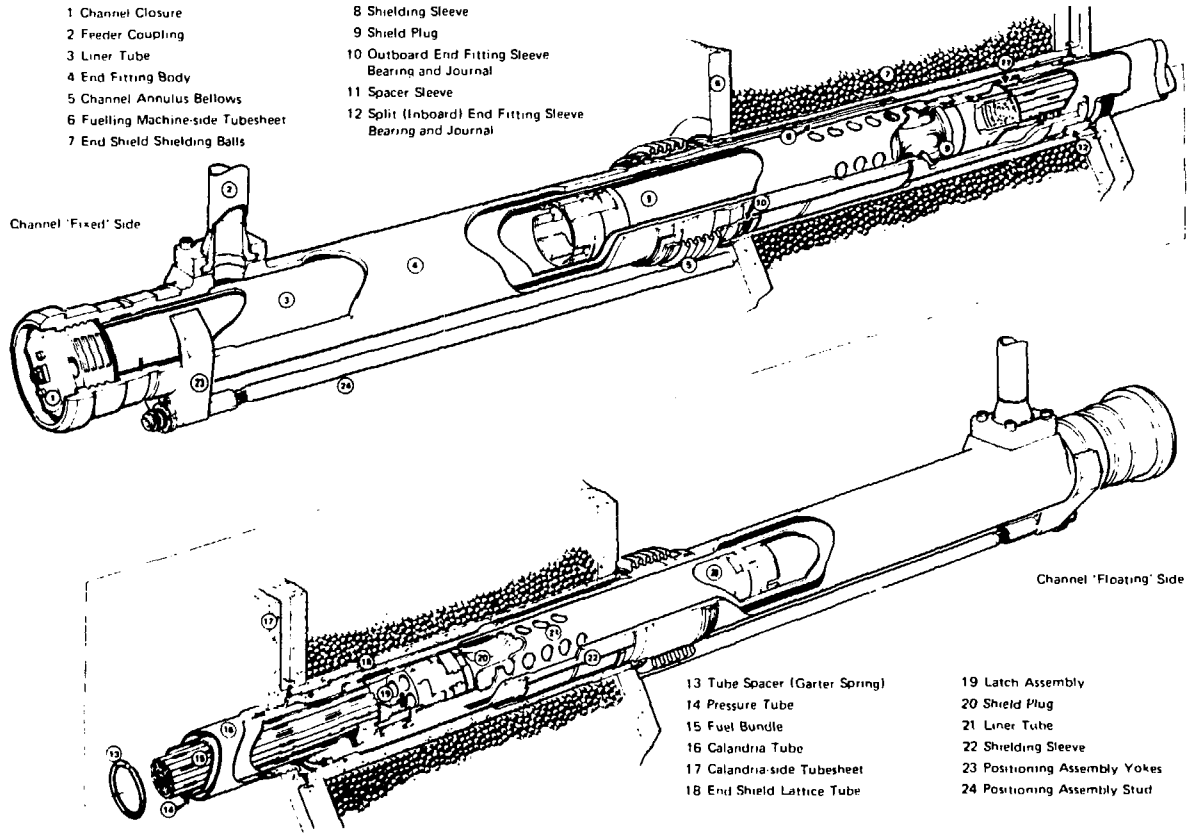
33. The general layout of the Bruce heat transport system is shown in Figure I-11; components of this system are shown in more detail in Figures I-12(a), I-12(b), I-12(c), and I-12(d).

Figure I - 11



\*The flows indicated are based on the reference design conditions.

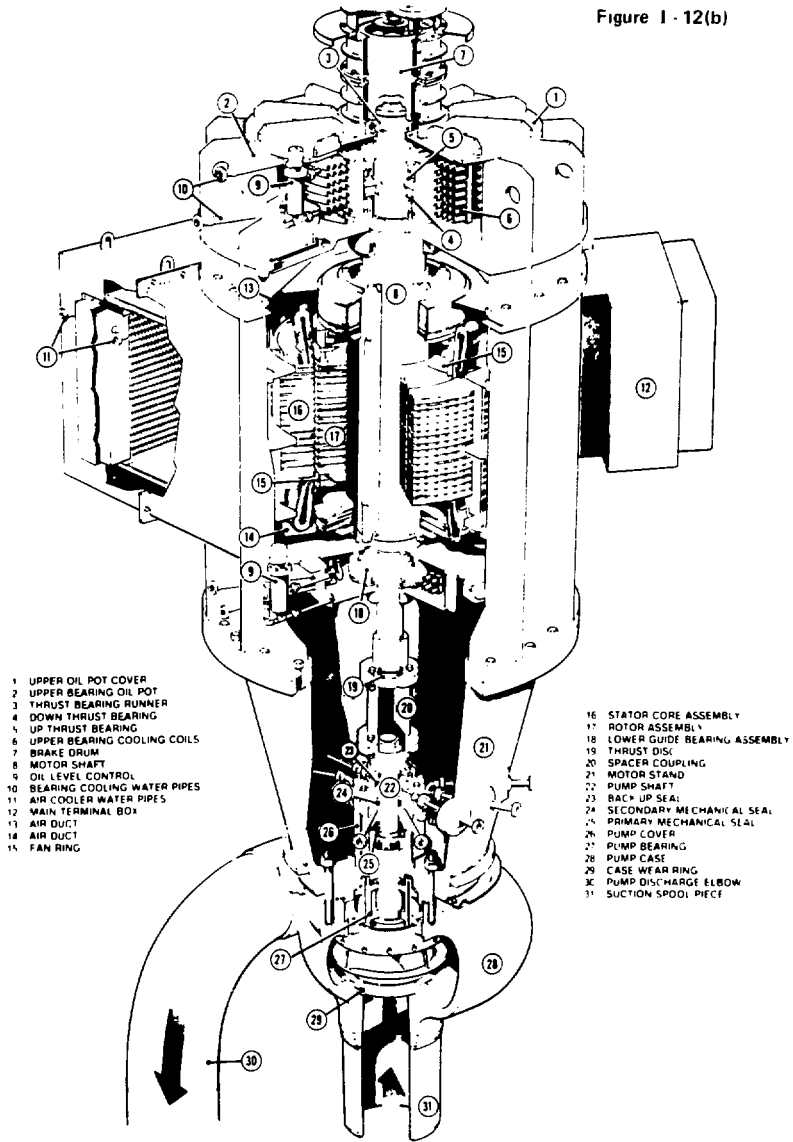
Figure I - 12(a)



Bruce B Fuel Channel Assembly

37

Figure I - 12(b)



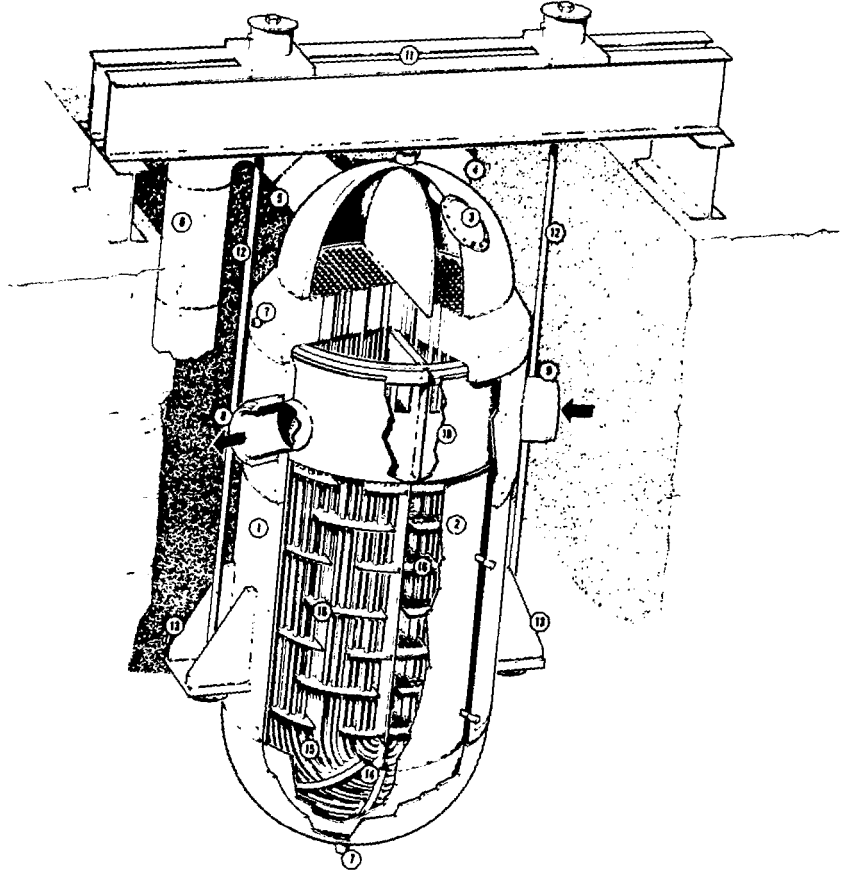
- 1 UPPER OIL POT COVER
- 2 UPPER BEARING OIL POT
- 3 THRUST BEARING RUNNER
- 4 DOWN THRUST BEARING
- 5 UP THRUST BEARING
- 6 UPPER BEARING COOLING COILS
- 7 BRAKE DRUM
- 8 MOTOR SHAFT
- 9 OIL LEVEL CONTROL
- 10 BEARING COOLING WATER PIPES
- 11 AIR COOLER WATER PIPES
- 12 MAIN TERMINAL BOX
- 13 AIR DUCT
- 14 AIR DUCT
- 15 FAN RING

- 16 STATOR CORE ASSEMBLY
- 17 ROTOR ASSEMBLY
- 18 LOWER GUIDE BEARING ASSEMBLY
- 19 THRUST DISC
- 20 SPACER COUPLING
- 21 MOTOR STAND
- 22 PUMP SHAFT
- 23 BACK UP SEAL
- 24 SECONDARY MECHANICAL SEAL
- 25 PRIMARY MECHANICAL SEAL
- 26 PUMP COVER
- 27 PUMP BEARING
- 28 PUMP CASE
- 29 CASE WEAR RING
- 30 PUMP DISCHARGE ELBOW
- 31 SUCTION SPOOL PLECT

Bruce B Heat Transport Pump

Ontario Hydro

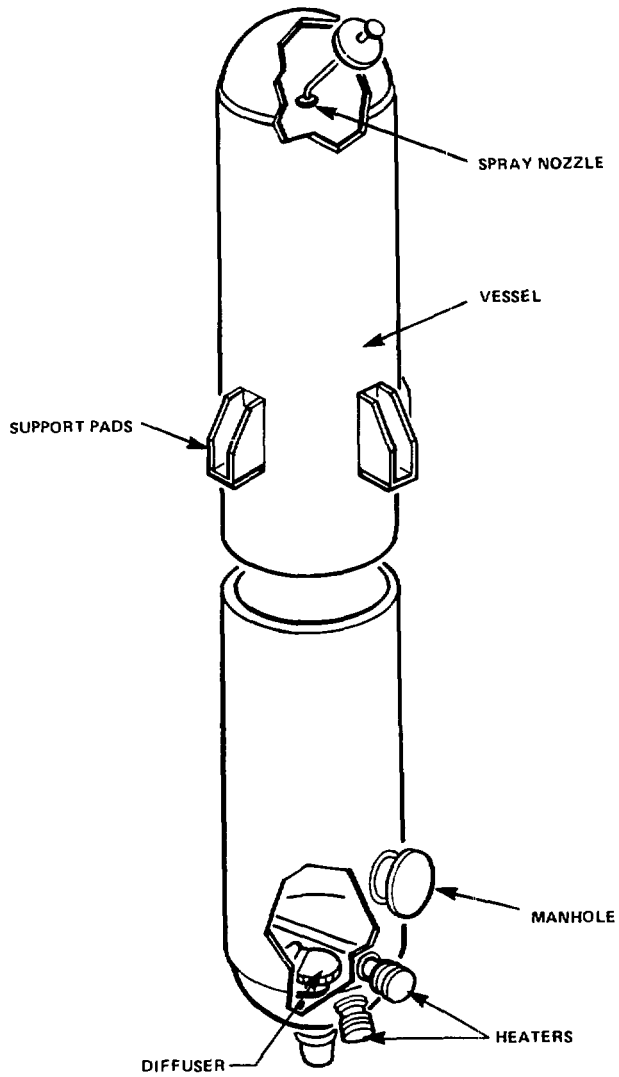
Figure 1 - 12(c)



- |                             |                           |
|-----------------------------|---------------------------|
| 1 PREHEATER OUTER SHELL     | 9 FEED INLET NOZZLE       |
| 2 SHROUD                    | 10 INLET FLOW DISTRIBUTOR |
| 3 MANWAY                    | 11 SUPPORT BEAMS          |
| 4 HEAVY WATER OUTLET NOZZLE | 12 SUPPORT ROD            |
| 5 HEAVY WATER INLET NOZZLE  | 13 SUPPORT BRACKET        |
| 6 HEAVY WATER INLET PIPE    | 14 TUBE BEND SUPPORTS     |
| 7 DRAIN NOZZLE              | 15 U-TUBES                |
| 8 FEED OUTLET NOZZLE        | 16 BAFFLE                 |

Bruce B Preheater

Figure 1 - 12(d)



Bruce B Pressuriser

34. The Darlington design adopted a two-loop heat transport system very similar to that of Pickering, but with a pressuriser attached to the interconnect line between the outlet headers. The major differences appear in the pump and steam generator arrangement: Darlington has only one heat transport pump and one steam generator in each pass of each heat transport loop. These changes were adopted to reduce the capital cost and complexity of the heat transport system.

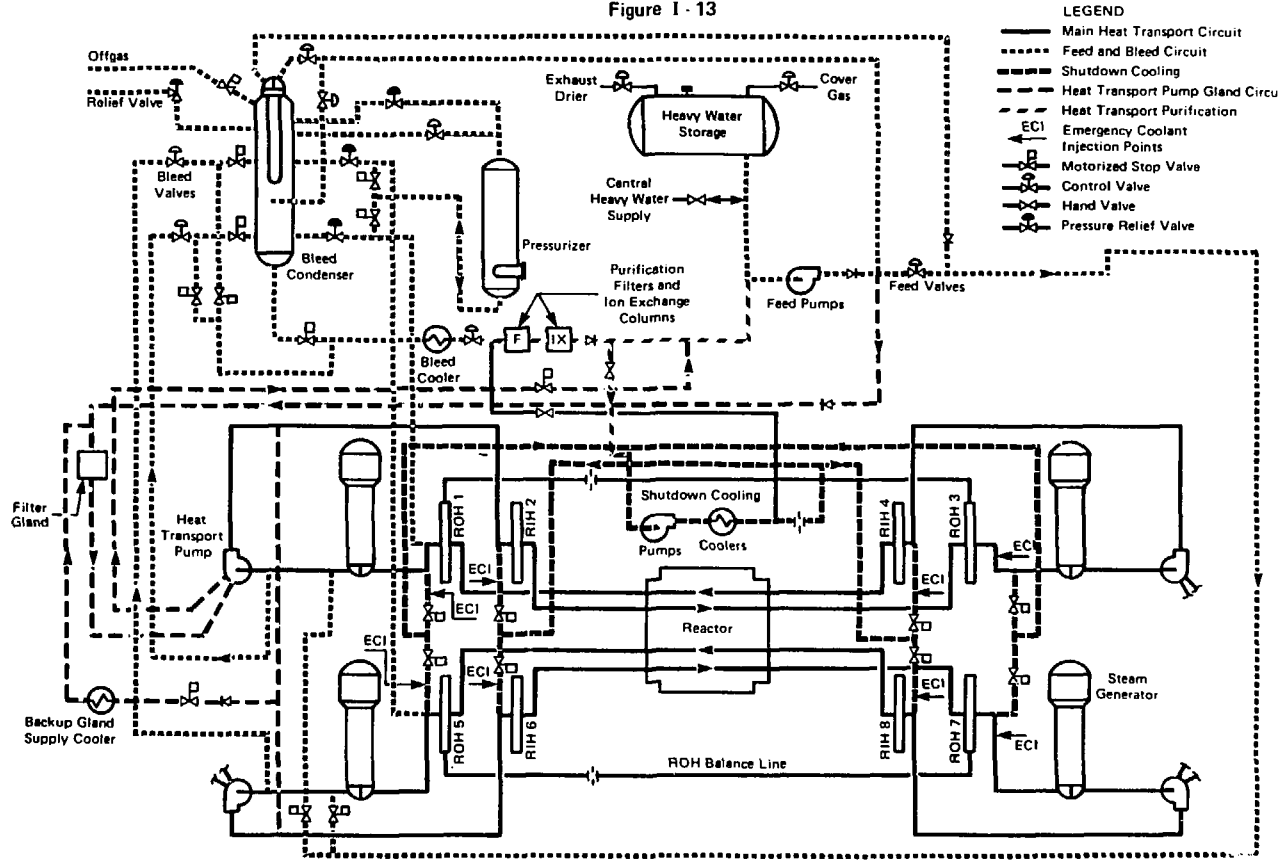
35. The general layout of the Darlington heat transport system is shown in Figure I-13. Components are shown in more detail in Figures I-14(a), I.14(b), and I.14(c).

### (c) Steam generation

36. Each steam generator consists of a large tank of ordinary water through which pass several thousand small-diameter tubes containing the heavy water of the heat transport system. The pressure of the ordinary water is maintained at about 5 MPa. Heat transferred through the steam generator tubes from the 10-MPa heavy water inside them boils the ordinary water outside the tubes. The upwelling of steam bubbles inside the tube bundle carries a large flow of mixed steam and water upwards. Steam is collected in a steam drum above the tube bundle, where steam separators are located to remove the entrained water. The water is returned via an annular shroud outside the tube bundle, but inside the steam generator shell, back to the bottom of the tube bundle. The mass flow of recirculated water is five to seven times the mass flow of steam. The purpose of this high recirculation flow is to mix the boiling water thoroughly to prevent local build-up of deposits that may cause corrosion and to improve the heat transfer characteristics on the outer surface of the steam generator tubes. Figure I-15 is a cutaway view of a typical steam generator, showing the main internal parts.

37. The main steam line is connected to the steam generator shell at the top. Saturated steam at about 4.7 MPa, containing less than 0.5% water, is sent to the turbine. This steam passes through the turbine and is eventually collected

Figure I - 13



- LEGEND**
- Main Heat Transport Circuit
  - - - Feed and Bleed Circuit
  - · - Shutdown Cooling
  - · · Heat Transport Pump Gland Circuit
  - · · · Heat Transport Purification
  - ← ECI Emergency Coolant Injection Points
  - ⊗ Motorized Stop Valve
  - ⊕ Control Valve
  - ⊘ Hand Valve
  - ⊙ Pressure Relief Valve

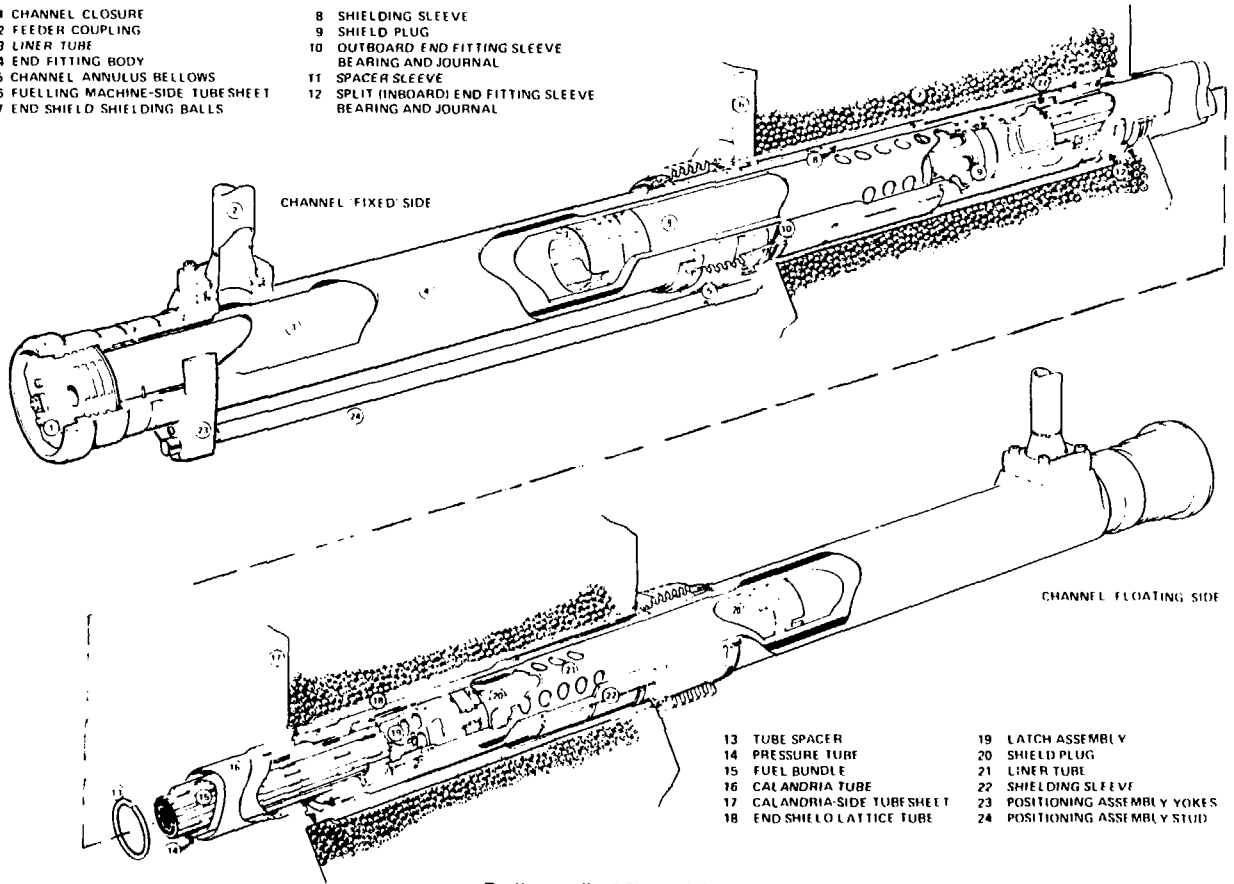
Darlington Heat Transport System – General Layout



Figure I - 14(a)

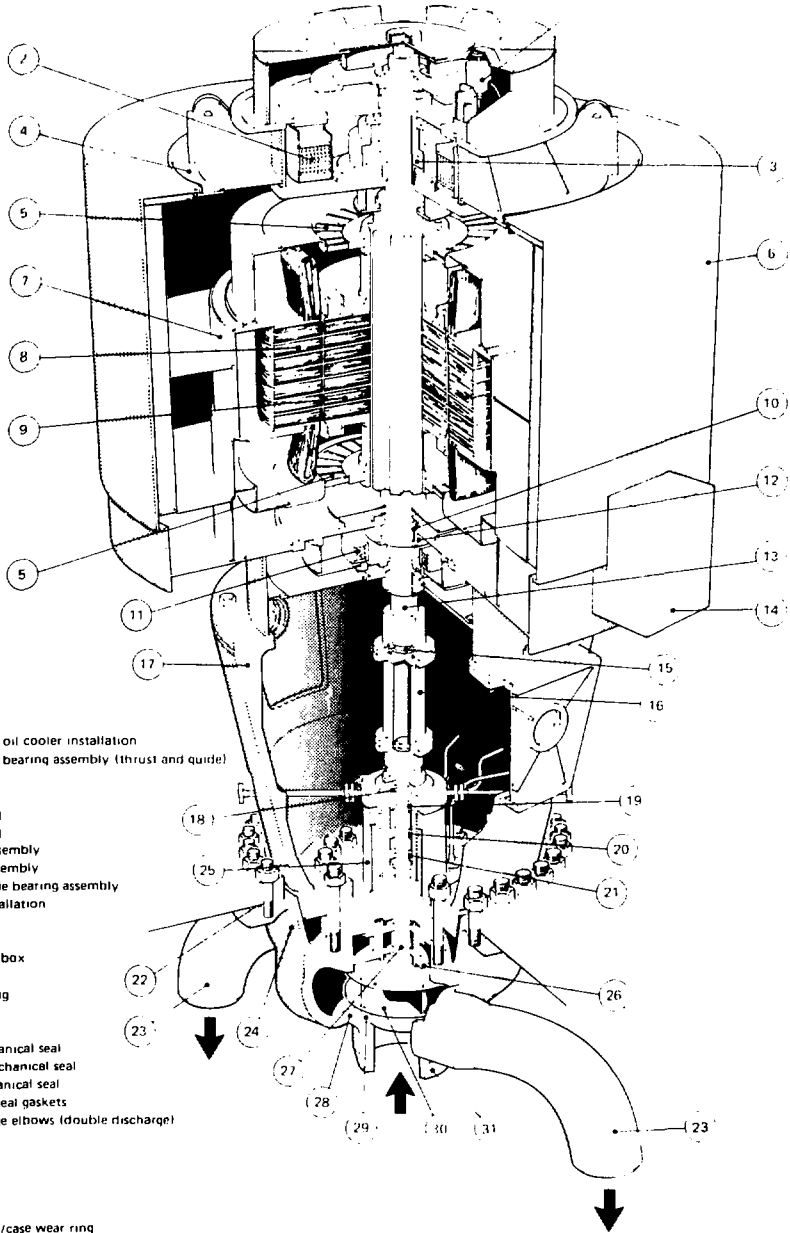
- 1 CHANNEL CLOSURE
- 2 FEEDER COUPLING
- 3 LINER TUBE
- 4 END FITTING BODY
- 5 CHANNEL ANNULUS BELLAWS
- 6 FUELLING MACHINE-SIDE TUBESHEET
- 7 END SHIELD SHIELDING BALLS

- 8 SHIELDING SLEEVE
- 9 SHIELD PLUG
- 10 OUTBOARD END FITTING SLEEVE BEARING AND JOURNAL
- 11 SPACER SLEEVE
- 12 SPLIT (INBOARD) END FITTING SLEEVE BEARING AND JOURNAL



- 13 TUBE SPACER
- 14 PRESSURE TUBE
- 15 FUEL BUNDLE
- 16 CALANDRIA TUBE
- 17 CALANDRIA-SIDE TUBESHEET
- 18 END SHIELD LATTICE TUBE
- 19 LATCH ASSEMBLY
- 20 SHIELD PLUG
- 21 LINER TUBE
- 22 SHIELDING SLEEVE
- 23 POSITIONING ASSEMBLY YOKE'S
- 24 POSITIONING ASSEMBLY STUD

Darlington Fuel Channel Assembly



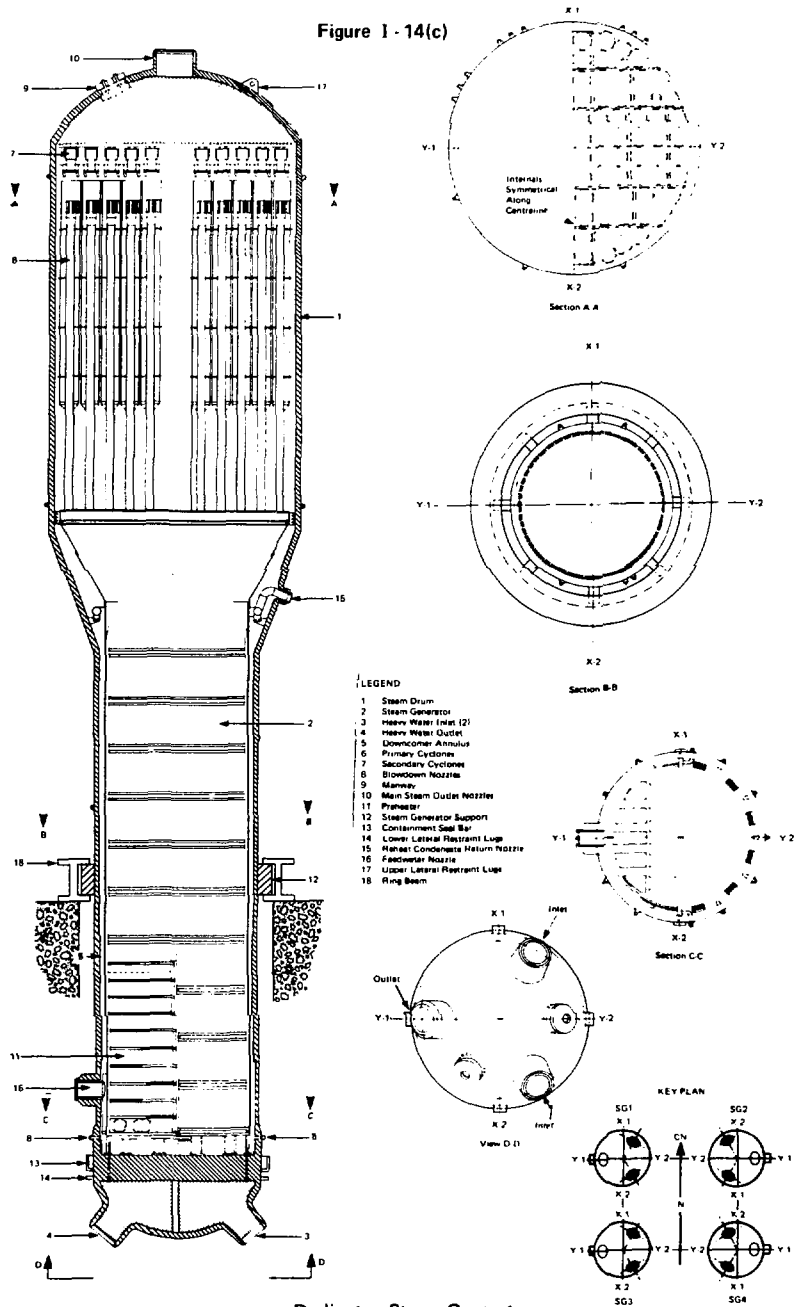
**LEGEND**

- 1 Brake
- 2 Non-drive end oil cooler installation
- 3 Non-drive end bearing assembly (thrust and guide)
- 4 Motor hood
- 5 Fan
- 6 Motor housing
- 7 Stator housing
- 8 Stator core assembly
- 9 Rotor core assembly
- 10 Drive end guide bearing assembly
- 11 Oil cooler installation
- 12 Labyrinth seal
- 13 Motor shaft
- 14 Main terminal box
- 15 Thrust disc
- 16 Spacer coupling
- 17 Motor mount
- 18 Back-up seal
- 19 Tertiary mechanical seal
- 20 Secondary mechanical seal
- 21 Primary mechanical seal
- 22 Containment seal gaskets
- 23 Pump discharge elbows (double discharge)
- 24 Pump support
- 25 Pump cover
- 26 Pump bearing
- 27 Pump shaft
- 28 Pump case
- 29 Pump impeller/case wear ring
- 30 Pump impeller
- 31 Suction spool piece

**Darlington Heat Transport Pump**

Ontario Hydro

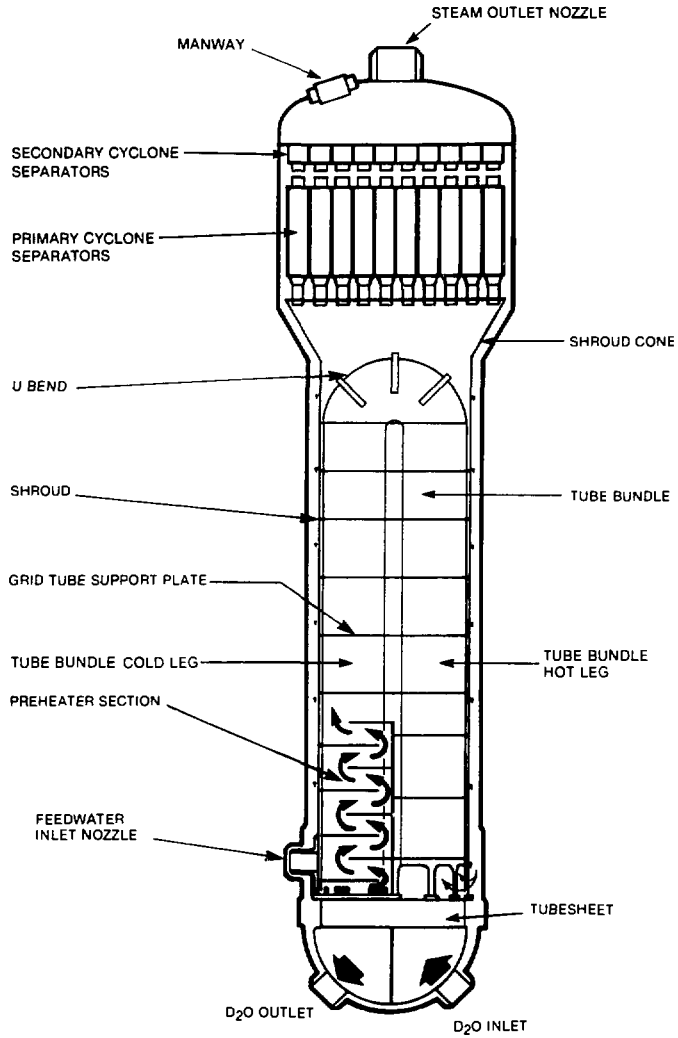
Figure I-14(c)



Darlington Steam Generator

Ontario Hydro

Figure I - 15



Typical CANDU Steam Generator

Ontario Hydro

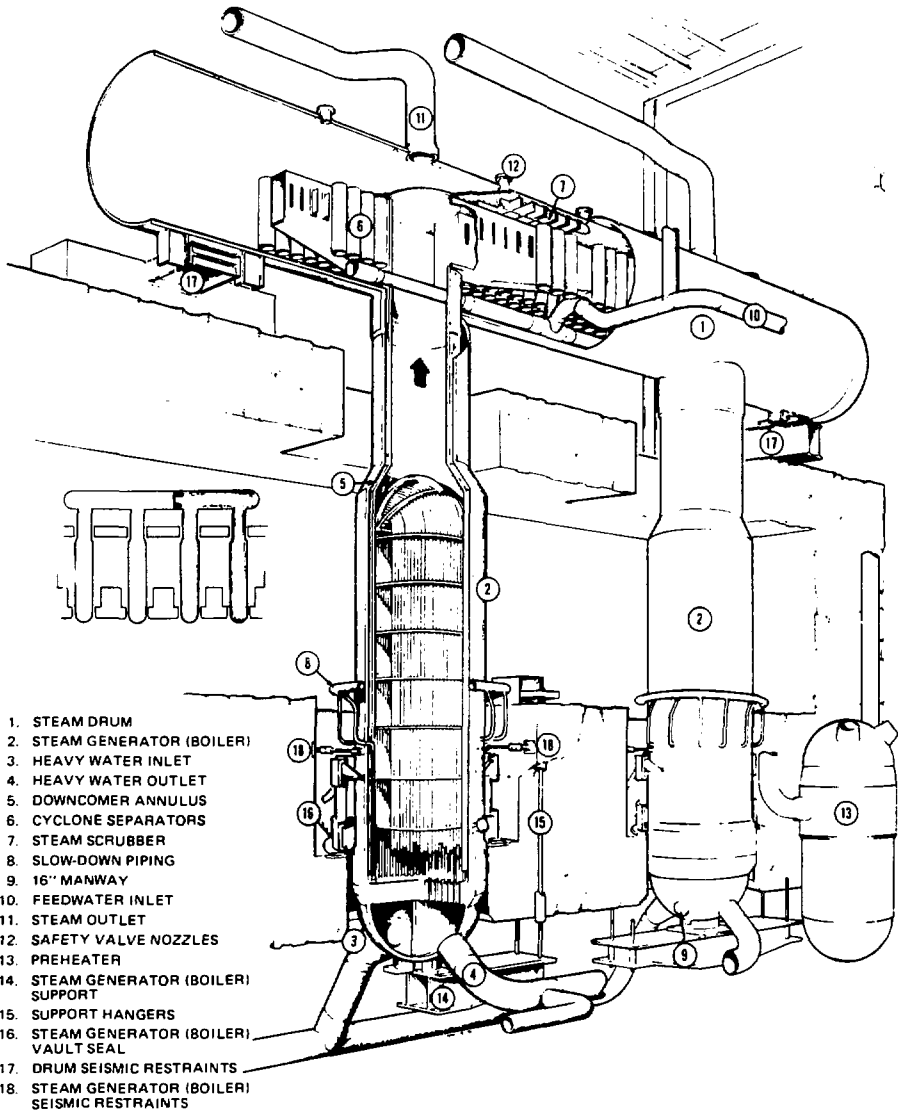
as water in the condenser, from which it is pumped through feedwater heaters back to the steam generator. The primary controls on the steam generator are water level and steam pressure. The level is maintained well above the top of the tube bundle by feedwater flow control, and the steam pressure is maintained either by adjustment of the reactor power or by adjustment of the opening of the steam governor valve at the inlet of the turbine.

38. The Pickering design utilises 12 small steam generators operated in groups of three. Steam generators can be individually valved out of the heat transport loop, as can the pumps. The reason for selecting small steam generators in the Pickering design was that this was the state of the art for steam generators in the mid-1960s. Large steam generators in other plants at that time had a history of tube failures, at least partly connected with the generator diameter and the difficulty of maintaining a sufficient recirculation flow in all parts of a large tube bundle so that corrosion products did not build up locally.

39. As has been mentioned, the Bruce design specification included a requirement for fairly rapid reactor power changes. This led to a need for a pressuriser on the heat transport system and to a large water volume in the steam generators. The design selected for Bruce A included four relatively large tube bundles in the steam generators at each end of the reactor, connected to a common horizontal steam drum, as shown in Figure I-16. This unique arrangement served to provide a large water inventory above the tube bundles. Unfortunately, the size and complexity of the steam generator outer shell posed very difficult problems of thermal distortion and stress during power manoeuvring in Bruce A. For this reason, the horizontal steam drum was abandoned in the Bruce B design. Each of the eight steam generators on Bruce B incorporates a conventional steam drum above the tube bundle, as shown in Figure I-17. Subsequent analysis, measurement, and control of Bruce A steam generators have largely overcome the early problems with this design.

40. The drive for simplicity in station layout led to selection of only four steam generators for the Darlington design. As a result, these steam generators are among the largest in the world. Even though the design is firmly based on

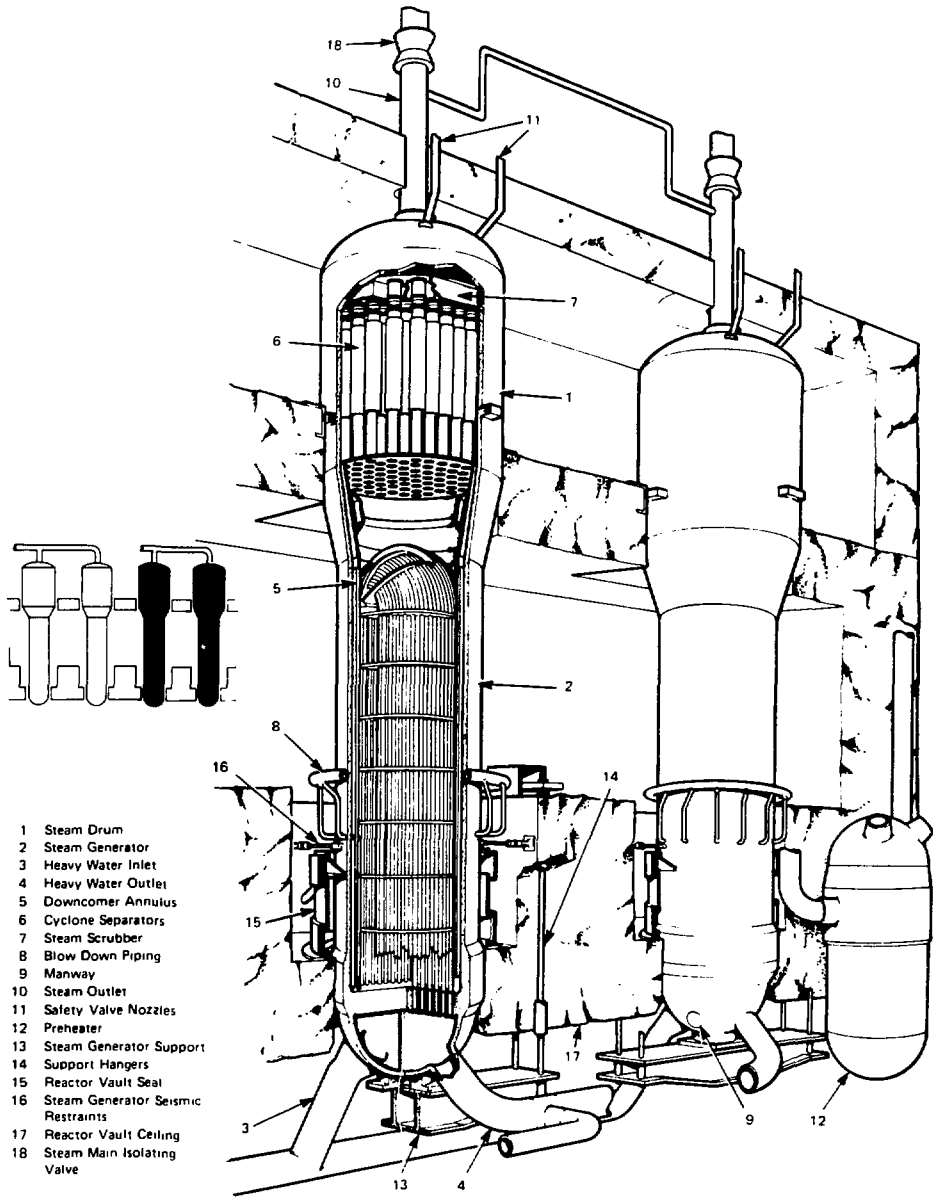
Figure 1-16



Bruce A Steam Generators and Common Steam Drum

Ontario Hydro

Figure I - 17



Bruce B Steam Generators and Preheater

Ontario Hydro

established practice and considerable operating experience, the design is unique in several respects.

**(d) Heat generation--the fission process**

41. A fission chain reaction can be initiated in naturally occurring or slightly enriched uranium fuel by distributing the fuel in correct proportions with a suitable moderator. The only other necessary condition for a chain reaction is that the assembly of fuel and moderator be large enough so that relatively few of the neutrons produced escape from the reactor. For each combination of fuel and moderator material (plus required control and structural material), there is an optimum mixture that produces the highest multiplication factor (the ratio of the number of neutrons in two successive generations). In a reactor *such as CANDU, the fuel is arranged in channels to facilitate heat removal.* The near-optimum ratio of fuel and moderator is achieved by selecting the desired spacing or pitch between fuel channels. The chain reaction releases heat energy.

42. Heat is produced primarily from fission of uranium-235 and plutonium-239. Most of this heat appears within the fuel material through collisions of the high-energy fission products with other atoms. Each fission produces about 2.5 fast neutrons. These neutrons are slowed down by collision, mainly with deuterium atoms. This slowing down is essential in order to increase the chance that a neutron will react with another fissionable atom to produce a further 2.5 fission neutrons. Each complete cycle can be considered as a neutron generation. When exactly 1.0 of the 2.5 original fast neutrons produces another fission, a self-sustaining chain reaction is established, and the reactor is said to be critical. When more than one of these neutrons produces a further fission, the reactor is supercritical, and the fission rate (and therefore power) in the reactor increases exponentially. When less than one fast neutron reacts in this way, the reactor is subcritical, and the fission rate reduces exponentially to zero. The rate of increase or decrease depends on the degree to which the reactor is supercritical or subcritical.



43. Prior to bringing a subcritical reactor to the critical condition, a source of neutrons (such as those produced by reaction of gamma rays with beryllium or deuterium) is first introduced into the reactor. In the presence of uranium-235 and a moderator, these neutrons produce a measurable number of fission chains. Because the reactor is subcritical, such fission chains die out after a few generations. If the reactor is being started up for the first time, the neutron source must be added from outside; sufficient neutrons usually are produced from internal reactions if the reactor already has been operated for a few days at a power level above 1% of full power.

44. The next step in starting up a reactor is to remove neutron-absorbing material gradually to increase the chance that a fission neutron will produce another fission after it slows down. When the reactor is critical, this continued fission becomes independent of the external neutron source, because each fission produces exactly one fission in the next generation. The absorbing material concentration is then reduced slightly, so that the number of neutrons in the reactor (and therefore the amount of heat produced per second) increases slowly until the desired power level is reached. At this point, the concentration of absorbing material is again increased slightly so that the number of neutrons, and therefore the power level, remains constant.

45. The process of fast-neutron production, slowing down, and subsequent fission is extremely rapid: in CANDU, less than 1 ms is required to complete a prompt neutron generation. Power increase is controllable only because approximately 0.6% of the fast neutrons are emitted from fission products rather than during the primary fission event. The half-life of fission product decay and neutron emission varies according to the particular fission product involved; the longest half-life of these delayed neutrons is about 1 min. This delayed-neutron fraction provides an adequate control band within which power manoeuvres can be done using conventional control systems.

46. If the number of excess neutrons in a generation exceeds the delayed-neutron fraction, the increase in neutron density  $N(t)$  and therefore power becomes completely independent of delayed neutrons. The increase rate is

exponential with time, inversely proportional to the prompt neutron generation time  $\Lambda$ , and proportional to the amount by which the reactivity exceeds the delayed-neutron fraction ( $\rho - \beta$ ), as shown in the equation:

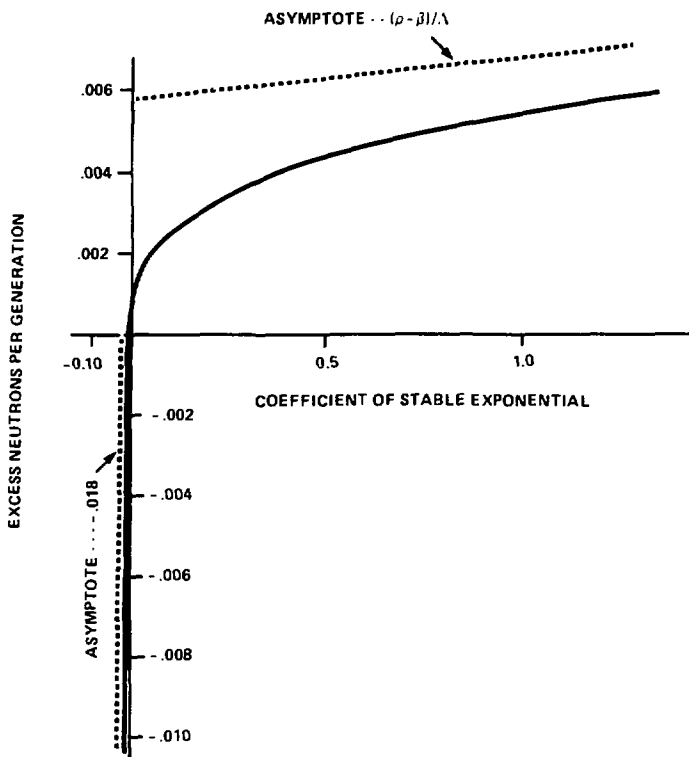
$$N(t) = N(0) \exp[(\rho - \beta)t/\Lambda]$$

The reactor is then said to be in a prompt-critical state: i.e., the reactor is critical on prompt neutrons alone. Figure I-18 indicates the behaviour of the time constant in a CANDU reactor as a function of the number of excess neutrons in each neutron generation. It can be seen that power control is much simpler when the number of excess neutrons per generation is small.

47. The first objective of a reactor regulating system is to ensure that power manoeuvres are carried out at slow and controlled rates when the reactor is supercritical. Power reduction can be carried out at a faster rate because this manoeuvre is in a safe direction.

48. Another factor that is important to the neutron balance on a time scale of the order of minutes to hours is that some of the fission products have very large absorption cross-sections for neutrons. The most important of these fission product chains involves the isotopes iodine-135 and xenon-135. About 0.06 atoms of iodine-135 are produced for each fission; these decay with a half-life of 9.2 h. The isotope xenon-135 absorbs a large number of neutrons at high reactor power (and is transmuted by this absorption). When reactor power is reduced after a period at high power, the production of xenon-135 continues via iodine-135 decay, whereas the amount destroyed in neutron absorption decreases as a result of the decreased level of neutron flux. The increased amount of xenon-135 absorbs neutrons and shuts off the neutron chain. This effect--often called xenon poisoning--must be compensated by either a temporary reduction of the concentration of some other absorbing material in the reactor, or an increase in the concentration of fissionable material in order to maintain a critical reactor condition.

Figure I - 18



Coefficient of Exponential Power Change as a Function of Excess Multiplication

49. The neutron density distribution in a large power reactor is determined by a transport process. Neutrons migrate freely through the materials by collision with various atoms in the reactor. This process can be approximated by assuming that the movement obeys Fick's law of diffusion: i.e., the neutron current is proportional to the negative gradient of the neutron density. The equations governing the distribution in this approximation are identical in form to those describing heat conduction in a medium with internal heat generation and zero absolute temperature at the outer boundaries. Using the heat conduction analogy, it can be seen that the local temperature in a large medium will be quite sensitive to changes in the local heat production rate. The local neutron density in a large nuclear reactor behaves in the same way--a small change in local neutron multiplication leads to a large change in local neutron density and therefore in local power density. Under these conditions, the neutron distribution is said to be loosely coupled. Distributed control elements are required to maintain the local neutron density within a narrow range around the desired distribution.

50. The temperature and density of materials in a reactor influence the neutron balance to some degree. Increases in fuel temperature usually have a negative effect on neutron multiplication, whereas increases in coolant temperature, moderator temperature, and coolant density may have either positive or negative effects. These effects are smaller in the CANDU reactor than in other designs. With all effects included, the reactor has a small negative power coefficient: i.e., the neutron multiplication rate decreases slightly following a given increase in reactor power. This is a very useful characteristic from the point of view of control systems design, because it results in a small negative feedback term that stabilises the control system. Small negative feedback is easier to accommodate in reactor design than is a large negative value--less total control movement is required to bring the reactor up to full power, so that the control system has less potential for initiating a large power excursion if it malfunctions and adds positive reactivity inadvertently. Also, the potential for a large increase in reactivity by cooling the fuel (e.g., by mixing of fuel and cooling water as in the Chernobyl accident) is significantly reduced.

51. On a longer time scale, measured in days, neutron multiplication capability decreases as a result of a reduction in the concentration of uranium-235 via fission, and as a result of an increase in the concentration of fission products that absorb neutrons. These changes necessitate replacement of used fuel bundles with fresh fuel bundles.

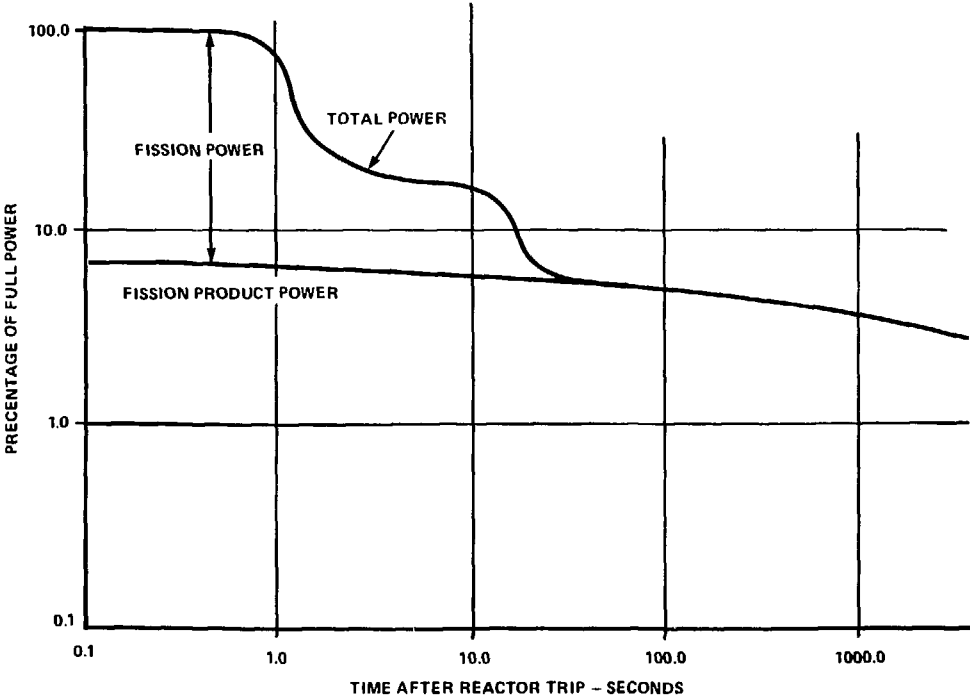
52. The basic physics of the Pickering and Bruce reactor designs are nearly identical. Slight differences exist because of the use of a different fuel bundle, the larger size of the Bruce reactor, the strategy of in-reactor fuel management, and the method of overcoming the xenon-135 negative reactivity. Guide tubes for neutron detectors and control elements are also arranged in a slightly different way: the absorption of neutrons in the zirconium structural components results in minor differences in the neutron density distribution across the reactor. Because it is larger, the Bruce reactor is somewhat more loosely coupled and so requires slightly larger amounts of distributed control capability.

#### (e) Heat energy sources

53. Of the total heat energy produced in fission, about 93% is released almost at the same time as the fission event occurs. The bulk of this energy is carried by the fission product fragments as they recoil away from the fission event. The remaining 7% of the heat is released through decay of fission products; this heat release is delayed until some time after the fission event. Figure I-19 shows the time dependence of heat release following shut-down of a CANDU reactor. This is an important characteristic with regard to safety: the fission chain reaction can be stopped very quickly (in about 1.5 s), but the decay heat must be removed for a long time after shut-down in order to prevent fuel overheating.

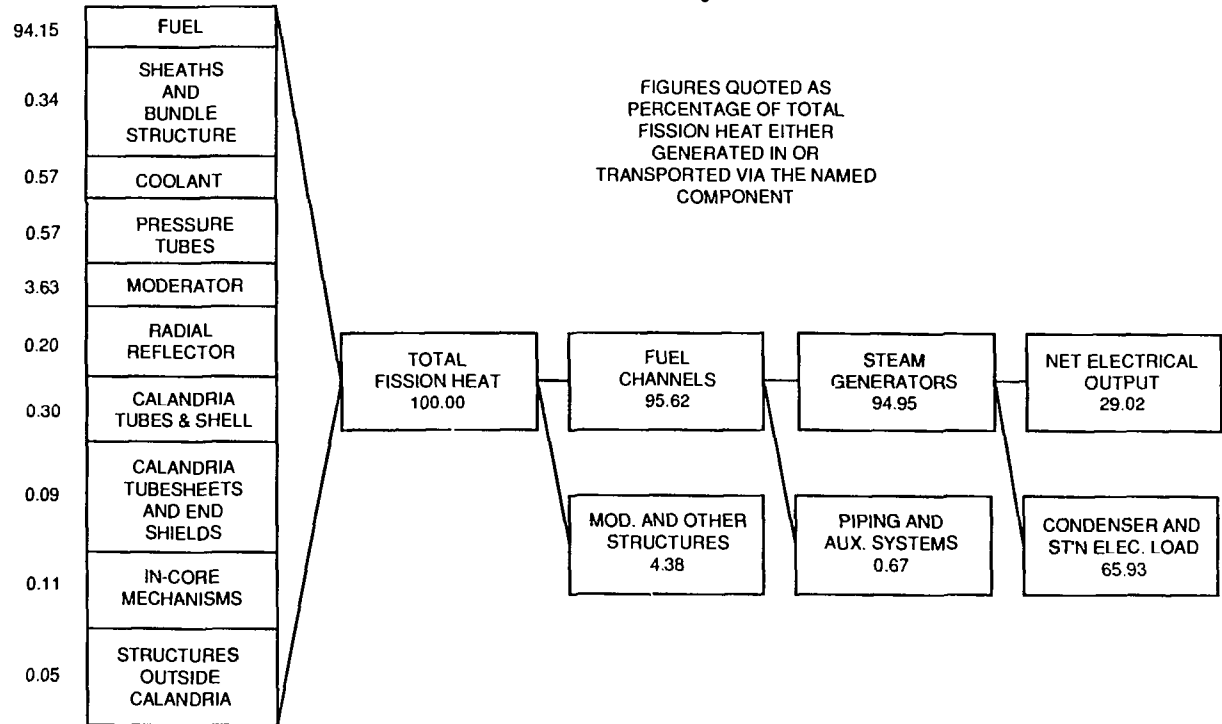
54. Another important consideration is the location of the heat release. As can be seen in Figure I-20, fission heat is deposited in fuel, coolant, structural materials, and moderator water. Outside the fuel, mainly neutrons and gamma rays deposit this energy.

Figure 1 - 19



Reactor Power Trace After Shutdown

Figure I - 20



Heat Deposition and Heat Transport in a Typical CANDU Unit

## (f) Reactor regulating system

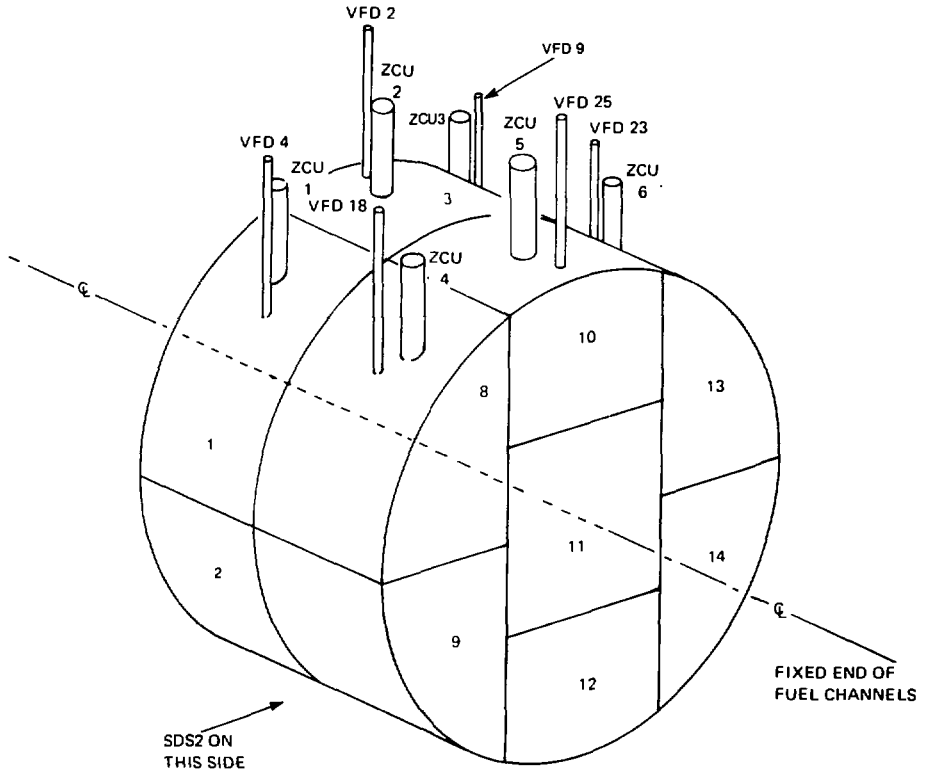
55. The CANDU regulating system is operated by a series of programs in a digital control computer. Its functions are (i) to control the total reactor power to the demand level; (ii) to control to pre-set values the relative power among 14 separate zones of the reactor; (iii) to reduce reactor power linearly with time in response to a number of different faults in process systems; and (iv) to rapidly reduce reactor power in response to more serious imbalances between heat production and heat removal rates.

56. The regulating system operates the primary power control devices, called the zone control compartments. The layout of these compartments in the reactor is shown in Figure I-21. These compartments contain ordinary water, which absorbs thermal neutrons much more than does heavy water. The water levels in the compartments are raised when the actual power is higher than the demand level and lowered when the actual power is below the demand level. Levels also may be raised or lowered to adjust the zone power to the desired level. When, for example, one of the zones of the reactor has a relatively high power, as detected by instruments located in each zone, the water level in that zone is raised to reduce the power relative to the remaining reactor zones. The external parts of the zone control system are shown in Figure I-22.

57. The control map for the zone control system is shown in Figure I-23. Within the normal control range (dark shading), the light-water zone control compartments are filled or emptied to match the demand power. When these compartments approach the upper limit of their control range, indicating that neutron multiplication is too high, the mechanical control absorbers are inserted to reduce the multiplication so as to bring the zone controllers back into their normal range. When the zone controllers approach the lower limit of their control range, indicating that neutron multiplication is too low, adjuster rods are removed from the core to increase the multiplication so as to bring the zone controllers back into their normal range.

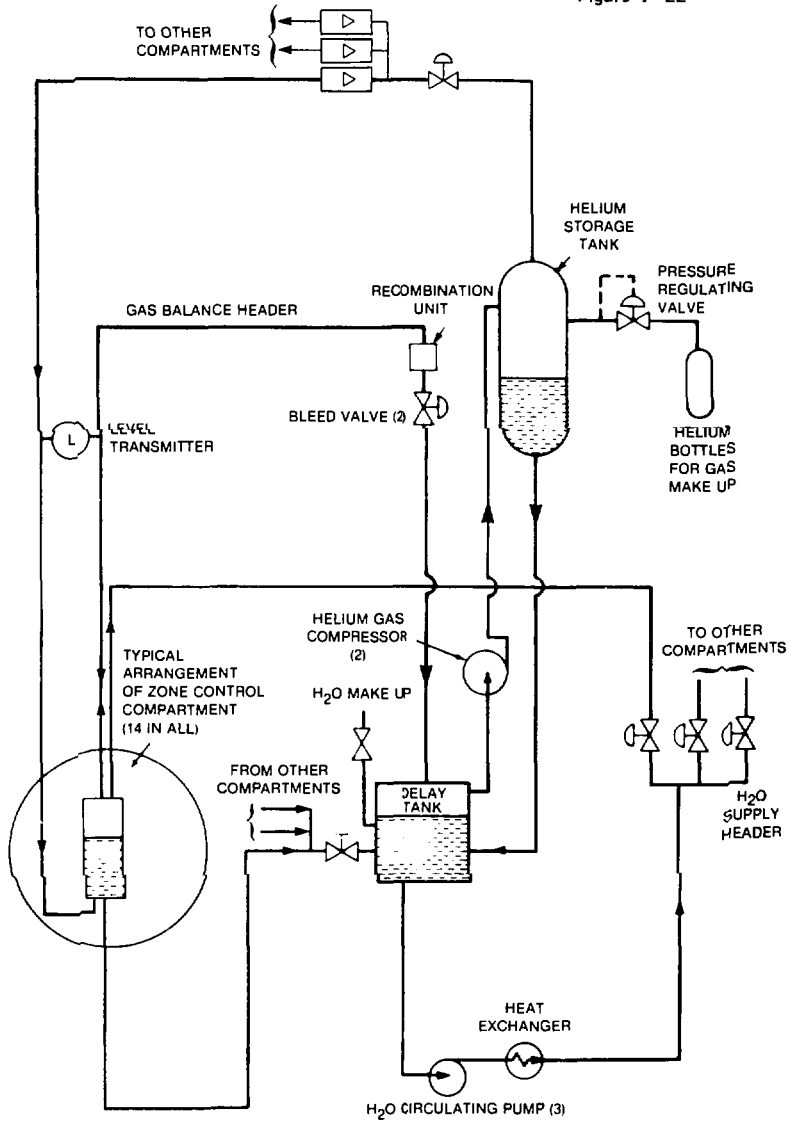


Figure I - 21



Layout of Zone Control Units

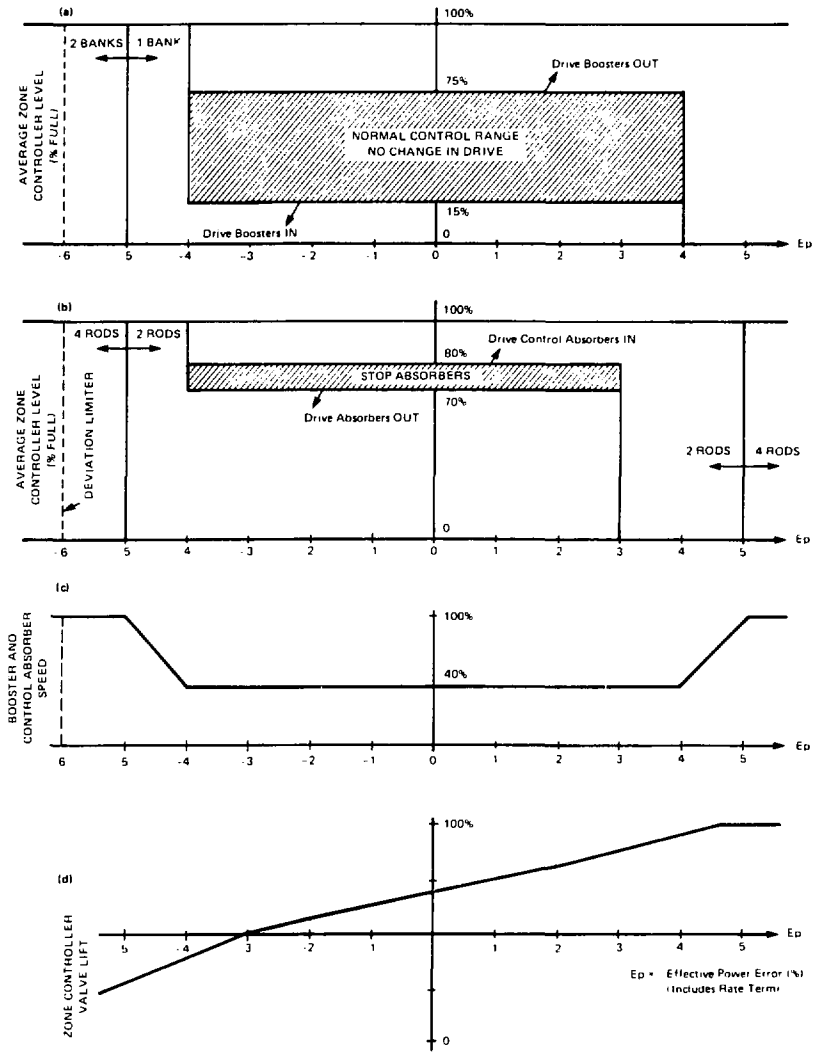
Figure I - 22



Systems for Zone Control Unit Filling and Draining

Ontario Hydro

Figure I - 23



Control Map Used by Zone Control System

58. A separate set of control routines monitors several operating parameters and reduces reactor power when the pre-set limit of any of these parameters is exceeded. Power can be reduced either slowly, for slightly abnormal conditions, or rapidly, for more severe malfunctions. A series of hardware interlocks prevents the control computers from increasing the neutron multiplication at greater than a pre-set safe maximum rate. Such interlocks are essentially switches, which are turned off when one control mechanism is being moved so that no other mechanism can be moved at the same time.

59. A second control computer operates in stand-by mode (receiving inputs but delivering no control output) at all times. If a fault condition is sensed on the first machine, control is transferred to the stand-by computer. If the second machine malfunctions, the reactor is shut down.

60. The basic control concepts of Pickering and Bruce are identical. The hardware and software devices used for control are somewhat different: the trend in newer reactor designs is toward more digital and fewer analogue control systems. The Darlington design utilises all-digital systems.

61. Bruce A utilises enriched uranium booster rods (normally located out of the core) to compensate for low neutron multiplication, instead of the more conventional adjuster rods in all other plants. Bruce B and Pickering A use adjuster rods containing cobalt-59 instead of stainless steel. Darlington is designed for stainless steel adjuster rods. These two rod types have exactly the same function, except that the cobalt-59 rods produce radioactive cobalt-60 when irradiated with neutrons. Cobalt-60 is a valuable product for medical therapy and other industrial uses.

### **(g) Fuel and fuel handling**

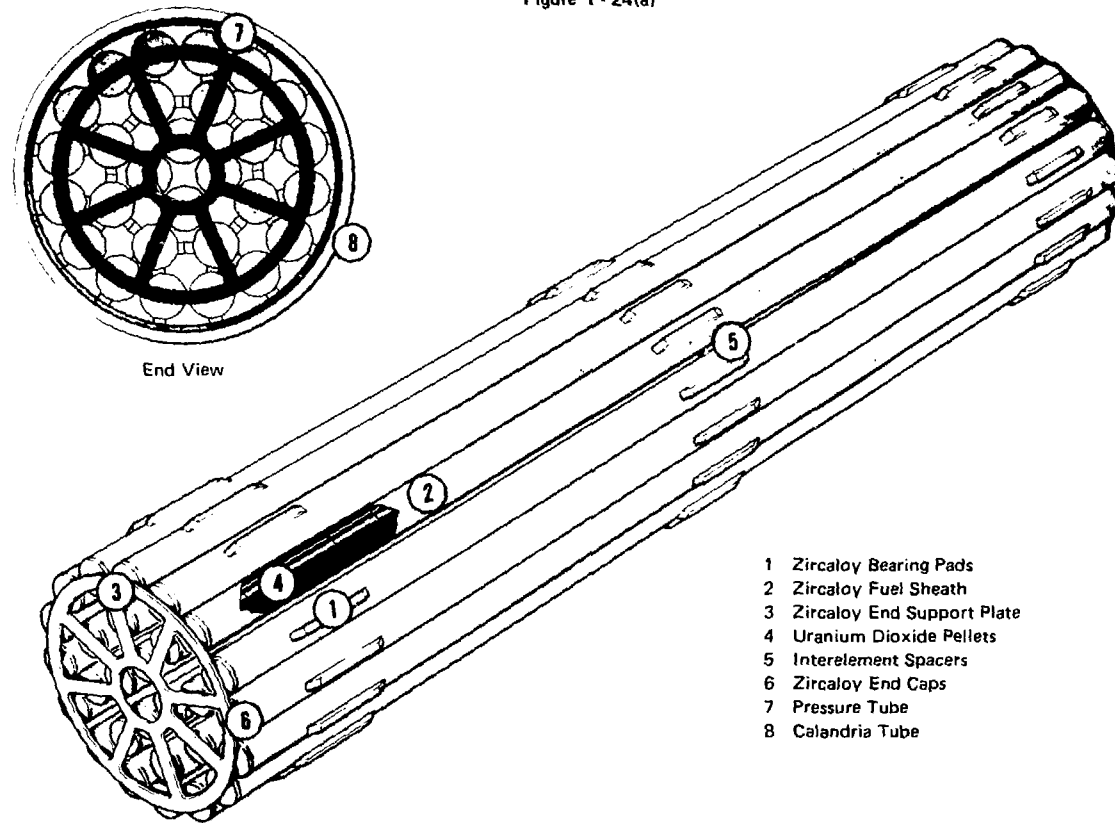
62. The CANDU fuel element consists of a tube (0.5 m long, 6 mm inner radius) of Zircaloy-4 (zirconium alloyed with 1.5% tin, 0.18% iron, and 0.1% chromium) sealed at both ends. It is filled with natural uranium dioxide pellets. The elements are grouped into fuel bundles with 28 elements to a bundle in

Pickering and 37 elements to a bundle in Bruce. A new fuel bundle contains about 20 kg of uranium. Either 12 bundles (Pickering) or 13 bundles (Bruce and Darlington) are loaded into each fuel channel. The in-core residence time is 12 to 18 months, during which the fuel produces an average of about 180 thermal megawatt-hours per kilogram of uranium. Approximately 1% of the original uranium is fissioned to produce this energy. At the end of the period, depletion of fissionable atoms plus accumulation of fission products in the fuel material result in a slightly negative contribution of the bundle to the overall neutron multiplication in the reactor, and the bundle must be discharged. The two fuel bundle types are shown in Figures I-24(a) and I-24(b).

63. While the fuel is in the reactor, the uranium-238 atoms are bombarded by neutrons; some of these neutrons are absorbed by uranium-238. After a short decay chain, the result is production of plutonium-239. This is a very important reaction, because plutonium-239 fissions following absorption of one more neutron and releases energy in the same way as does uranium-235. In a real sense, the non-fissionable uranium-238 atoms are converted to fissionable material, which replaces the uranium-235 atoms in the uranium. Nearly half the energy produced during the useful lifetime of a CANDU bundle is produced by fission of plutonium-239.

64. Used fuel is discharged and new fuel added while the reactor is at full power, using a semi-automatic fuelling system. Under computer control, two fuelling machines move from the new fuel port (where one has been loaded with eight new fuel bundles) to opposite reactor faces. One machine locks on to each end of a single channel; after seal testing, the machines are raised to the same temperature and pressure that exist inside the channel. Tools inside the machines then remove the channel closure plugs and shield plugs at the end of the channel and store them in neighbouring barrels of rotating drums inside the machines. The drum of the machine containing new fuel is then rotated successively to the positions where the new fuel is located, two bundles in each drum position. As these bundles are pushed into the reactor, the used-fuel machine at the other end receives used fuel into its corresponding drum positions. When the operation is complete, the shield and closure plugs are replaced,

Figure I - 24(a)



Pickering 28 – Element Fuel Bundle

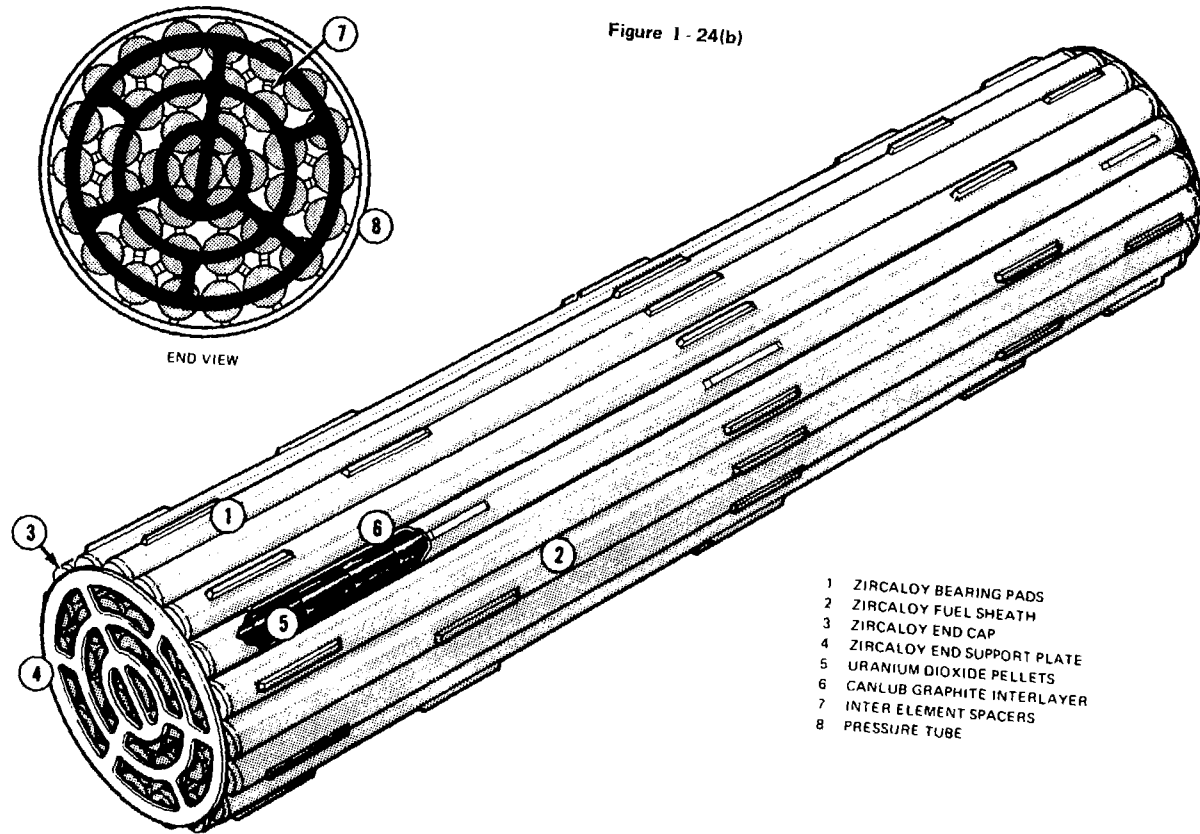


Figure 1 - 24(b)

Bruce and Darlington 37 Element Fuel Bundle

various leak tests are conducted, the fuelling machines are cooled down and depressurised, and the machines are unlocked from the channel. The machines are then moved on rails to the used-fuel port, and the used bundles are transferred to the receiving bay of the used-fuel storage pool. The general concept of fuel replacement is shown in Figure I-25.

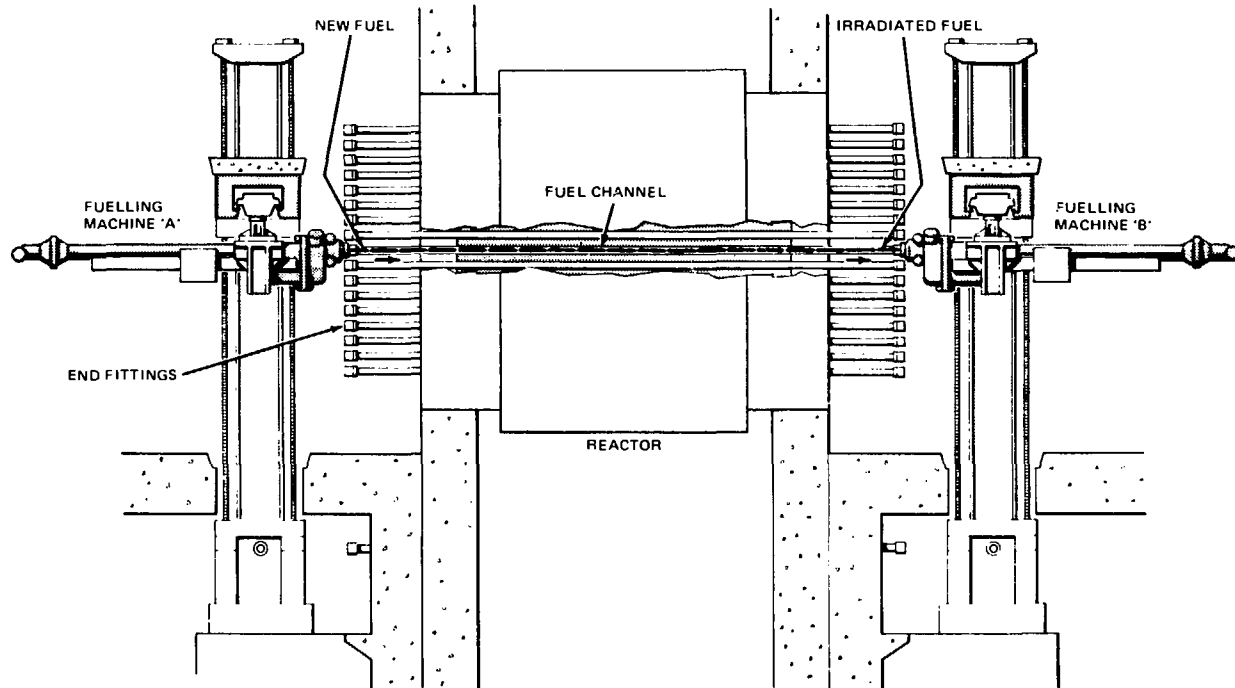
65. The fuel handling systems at CANDU stations, although admittedly complex, have functioned remarkably well over the lifetime of the systems. The enormous economic advantage conferred by on-line fuelling fully justifies the high cost of these systems. There have been a few instances in which fuel has been damaged in the process of unloading, and some radioactive materials have been released inside containment buildings. These mishaps have all been overcome with only a small loss of time, and the radioactive material has been successfully recovered.

66. The fuelling machines at Pickering and Bruce are somewhat different in detailed design, but are identical in concept. The largest difference exists in the mechanisms for transfer of used fuel from the used-fuel port to the receiving bay. In Pickering, there are two fuelling machines dedicated to fuelling each unit; each machine is carried on a bridge that can traverse the whole reactor face, and the machine can be lowered through the reactor vault floor to reach the fuel discharge port and maintenance area. Fuel discharged from a fuelling machine is carried on an underwater conveyor system to the central receiving bay. Figure I-26 illustrates this fuelling concept.

67. In Bruce, a pair of fuelling machines can service any unit; the whole fuelling trolley moves on a rail system. On arrival at a unit, the machines are picked up onto the fuelling bridges to perform fuelling operations and are then returned to the trolley. In this concept (which was selected in order to reduce equipment costs and to reduce the chance of the loss of fuelling capability on any one unit as a result of outages of any single machine), the used fuel is transported to the receiving bay inside the fuelling machine and then transferred directly through a porthole into the bay. Figures I-27 and I-28 show the Bruce fuelling concept.



Figure 1 - 25

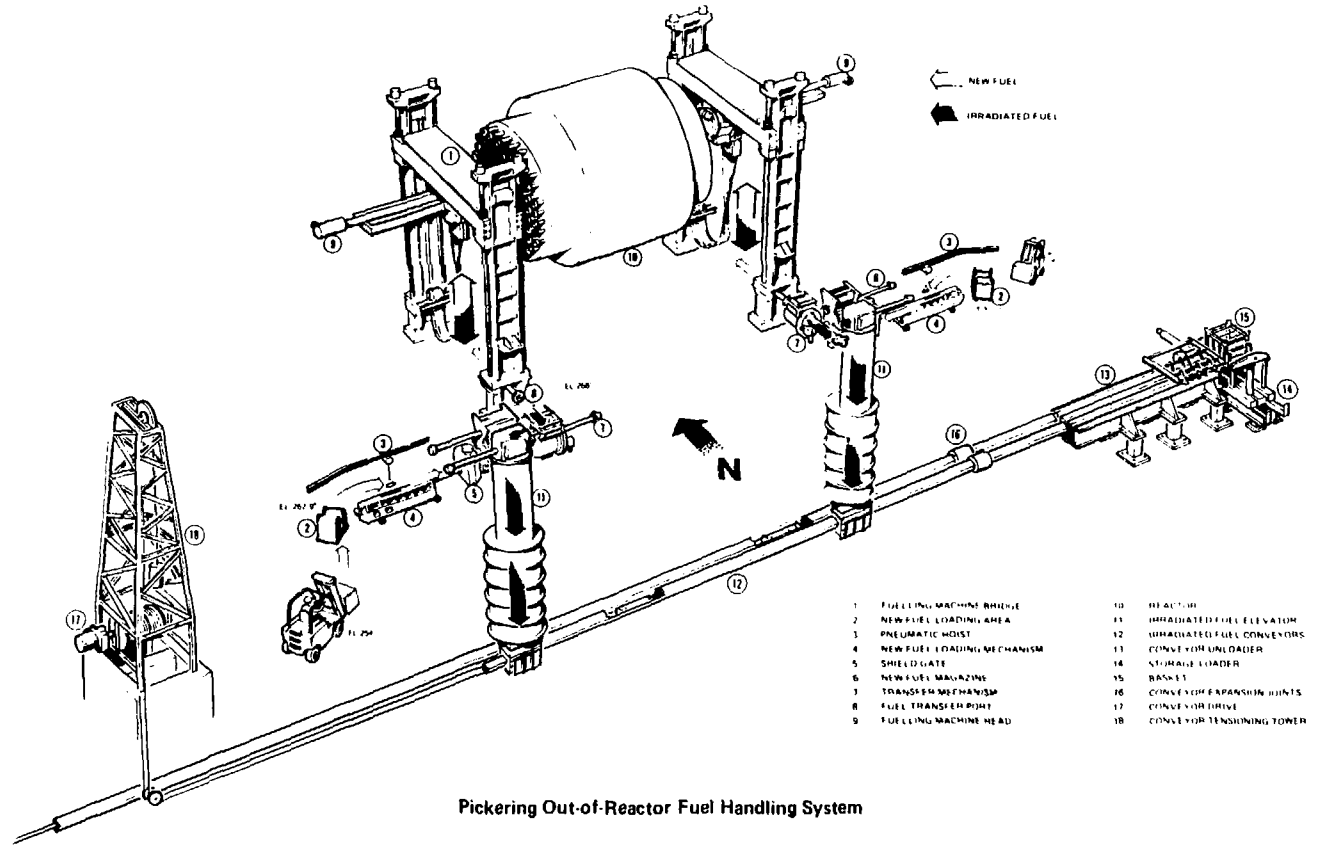


TWIN FUELLING MACHINES, CONTROLLED BY COMPUTERS, ARE USED FOR 'ON-POWER' REFUELLING THE MACHINES HOME ON A REACTOR FUEL CHANNEL, MAKE A PRESSURE TIGHT CONNECTION, REMOVE SEALING AND SHIELD PLUGS, INSERT AND REMOVE FUEL BUNDLES AND RECLOSE THE CHANNEL.

Arrangement of Fuelling Machines During Fuelling Operation

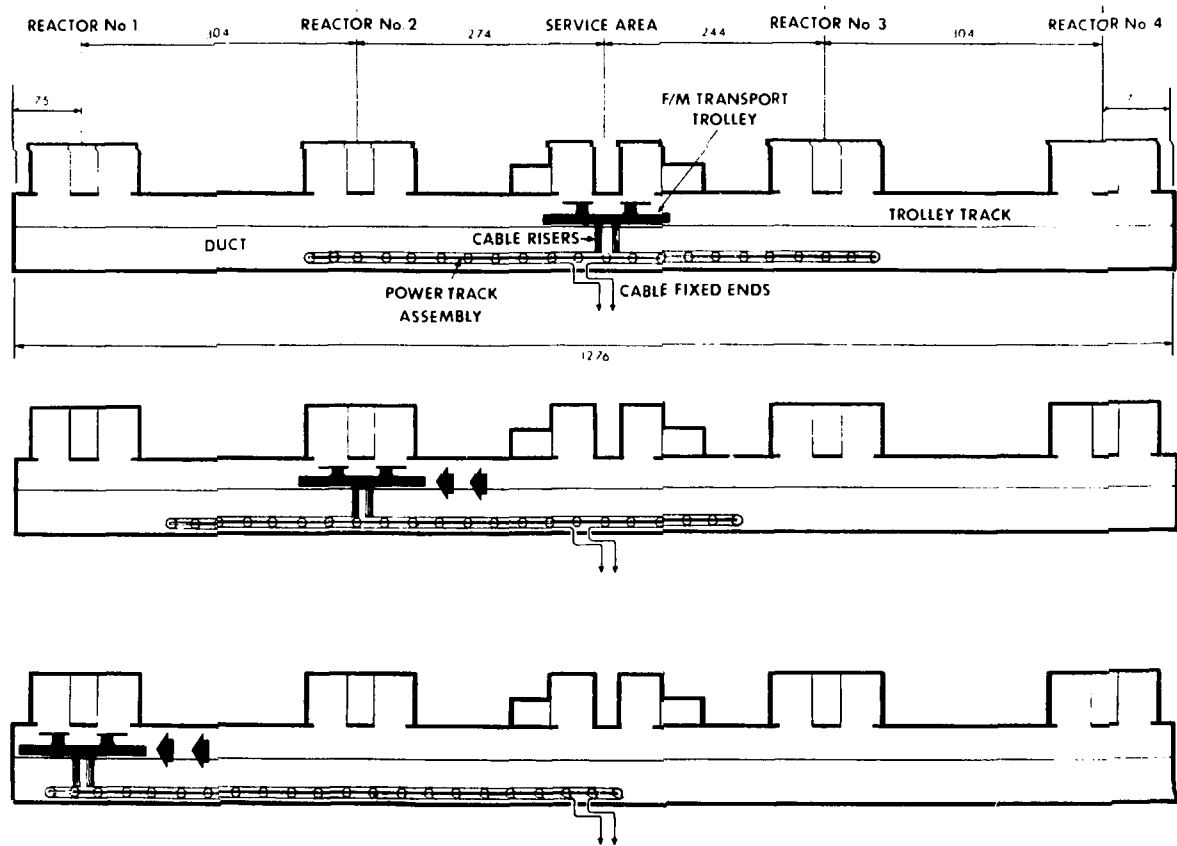
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Figure I - 26



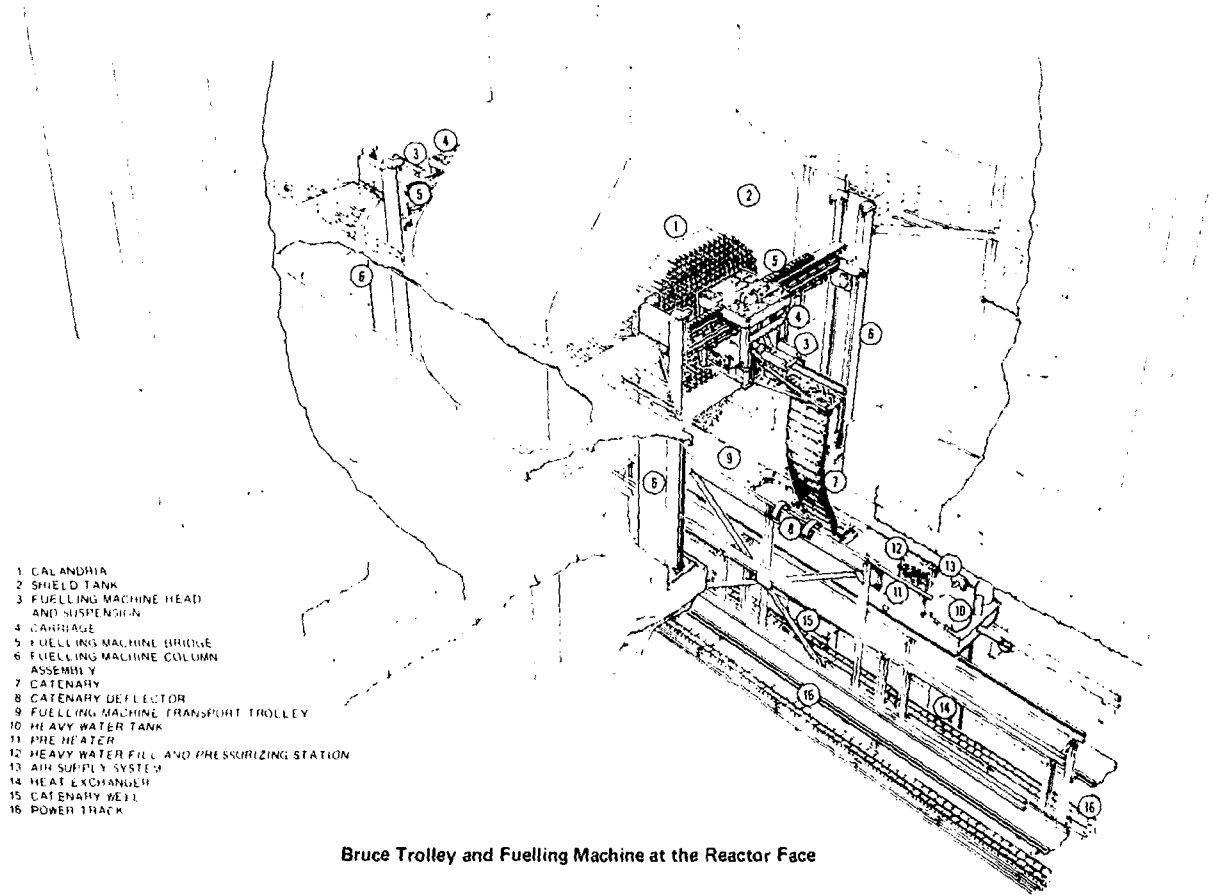
Pickering Out-of-Reactor Fuel Handling System

Figure I - 27



Bruce Power Track Fuel Handling System

Figure 1-28



- 1 CALANDRIA
- 2 SHIELD TANK
- 3 FUELLING MACHINE HEAD AND SUSPENSION
- 4 CARTRIDGE
- 5 FUELLING MACHINE BRIDGE
- 6 FUELLING MACHINE COLUMN ASSEMBLY
- 7 CATENARY
- 8 CATENARY DEFLECTOR
- 9 FUELLING MACHINE TRANSPORT TROLLEY
- 10 HEAVY WATER TANK
- 11 PRE HEATER
- 12 HEAVY WATER FILL AND PRE SCRUBBING STATION
- 13 AIR SUPPLY SYSTEM
- 14 HEAT EXCHANGERS
- 15 CATENARY WELL
- 16 POWER TRACK

Bruce Trolley and Fuelling Machine at the Reactor Face

68. Darlington retains the basic features of the Bruce fuelling system, with the exception of having two receiving bays, one at each end of the station. Fuelling machine maintenance facilities are located adjacent to the receiving bay in both the Bruce and Darlington designs.

#### **D. Special Safety Systems**

69. Special safety systems are those having no role in normal operation of the plant, but which are installed solely to control accident sequences and to mitigate the consequences of failures that may occur in the process systems (those used for normal operation) during any phase of operation. The special safety systems are independent of process systems and of one another.

70. A serious accident (with uncontrolled release of fission products from the station) at the National Research Experimental (NRX) reactor in 1952 (see Chapter II-C of the main report) was the catalyst for much of the present Canadian approach to reactor safety, including the requirement for separate and independent special safety systems. Overall, the lessons learned from this accident provided a great deal of guidance that set the Canadian power reactor safety design philosophy on a productive path to the future.

##### **(a) Protection concepts**

71. The reactor process systems are designed to maintain the process parameters (e.g., power, pressure, temperature) within a well-defined operating envelope during all stages of operation (start-up, steady-state operation, anticipated operating transients, shut-down). Because the reactor is largely operated by computer control, the operators' major duties during normal operation are to monitor the functioning of these automatic systems, detect incipient faults, correct faults where possible, conduct routine maintenance of process systems, and test special safety systems to ensure they are available in case they are needed in response to an unanticipated event.

72. The main control room operators also respond in cases in which the operating parameters approach the limits of safe operation defined by the safety system trip parameters; the operators' response is to move the plant operating state in the safe direction. The trip parameters are pre-set limits assigned to measured values of quantities such as pressure, temperature, and reactor power: shut-down, emergency cooling, and containment systems are activated automatically as appropriate if the limits are exceeded.

73. There is only one objective of any special safety action: to keep the fission products that are accumulated in the fuel elements from escaping to the environment. These fission products represent the only significant radiological danger from nuclear plant operation; public safety is assured if they are retained inside the plant. The preferred method of accomplishing this is to keep the fuel bundles cool so that the Zircaloy fuel sheaths remain intact. Other barriers to release, such as heat transport piping and the containment system, are provided as back-up.

74. In the short term, the first and most important step is to shut off the chain reaction to reduce the energy production rate. The second step is to ensure cooling of the fuel. To do this, the primary system coolant inventory must be maintained, either forced or natural circulation must be maintained or re-established, and a heat rejection route to the ultimate heat sink must be available. The third step is to close the containment structure so that any fission products not retained inside the fuel sheaths or primary heat transport system boundary are retained inside the containment system.

75. The shut-down system trip parameters can be thought of as a boundary enclosing a multi-dimensional operating envelope. The setpoints of these parameters (typically: high global reactor power, high local reactor power density, high positive rate of change of reactor power, either high or low primary system pressure, low primary coolant flow rate, high reactor building pressure, low steam generator level) are determined at the design stage by extensive analysis of many stylised accident scenarios. These accident scenarios are intended not to reproduce actual accident sequences, but to test the

capabilities of the special safety systems for a very wide range of initial conditions under which they might be called upon to protect the fuel.

76. Detectors are placed at appropriate locations for measurement of each parameter and connected to logic channels that are completely independent of the process system electronics. Each parameter is detected by three independent channels of logic. If any two values exceed the trip setpoint, the shut-down system is fully activated and the chain reaction is stopped in about 1 s. AECB requires that two different trip parameters be capable of achieving satisfactory shut-down for each accident scenario. In addition, the regulating system must be presumed to act normally if this results in greater demands on the shut-down system and to be inactive if its action results in lesser demands on the shut-down system. The other two special safety systems are similarly activated, as will be detailed in the next two subsections.

77. The philosophy guiding the nuclear power plant designer is one of defence-in-depth. Defences range from preventive measures, such as high-quality hardware and well-trained, highly motivated staff, to mitigative measures, such as those represented by special safety systems and establishment of an exclusion zone around each nuclear plant, within which permanent residence is not permitted. At each level of defence, care is taken in design and operation so that a single equipment failure will not defeat all barriers to release of fission products, and so that a failure will not propagate further failures.

78. The underlying reality in the design of any safety system, whether it be in homes, automobiles, aircraft, or nuclear power plants, is that the most severe possible accident can occur at some probability. The objective of the design is to make this probability acceptably low. It can never be zero. The value-judgement word is "acceptably": a real or perceived risk is accepted by society in return for some real or perceived benefit. The safety design engineer, therefore, always works in an indefinite framework of design requirements. Objective measures of safety can be compared easily with alternative ways of achieving the same benefit--but perceived risks and benefits vary considerably

with time and with the political mood of the society. Today's "acceptable" level of safety could be tomorrow's "not enough" and the next day's "too much." High levels of safety can be achieved, but at significant cost. The society must decide the acceptable balance between cost and benefit for regulated activities such as nuclear energy.

79. The most fundamental reality of nuclear power plant operation is that the maximum consequences of any severe accident are limited: they can never be large compared with other potential events, such as hydraulic dam failures or poisonous chemical releases. A nuclear power reactor absolutely cannot explode like a nuclear weapon, nor can it release enough energy or fission products to produce devastation equivalent to such an event.

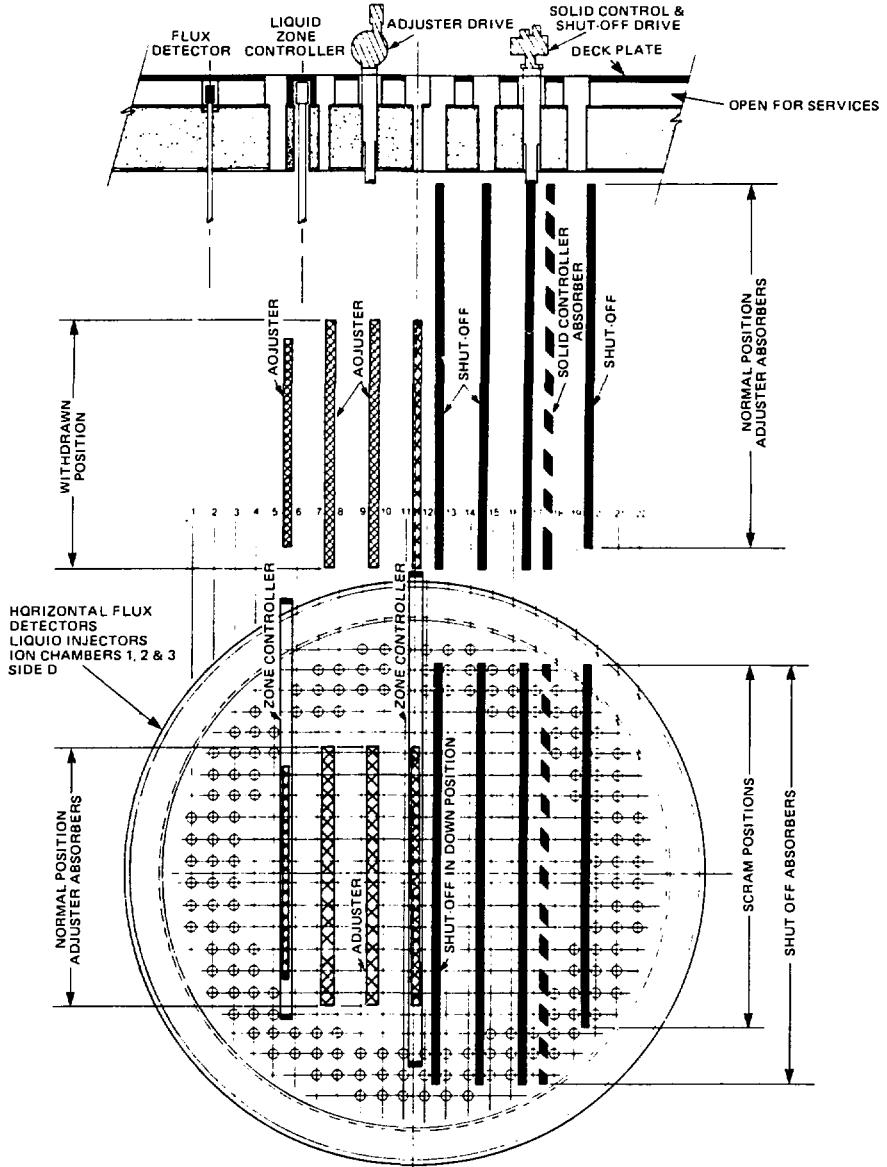
#### **(b) Shut-down system 1**

80. Shut-down system 1 (SDS1) consists of 28 stainless steel sheathed cadmium rods suspended vertically in guide tubes above the reactor, as shown in Figure I-29. These rods are known as shut-off rods. Each rod is restrained from dropping down its guide tube by a cable wound around a sheave and held by an electromagnetic clutch. The clutch voltage is supplied through circuits incorporating series relays operated by the trip parameter channels. The clutch voltage will be cut off by any combination of two out of three open-circuit trip channels, and the rods will drop into the core, with the first part of their travel being accelerated by coil springs. The time from cut-off of the clutch voltage to full rod insertion is about 1 s.

81. Two-out-of-three voting logic permits testing of one trip channel during operation. During testing, the channel is placed in the open-circuit state, so that if one more parameter is exceeded reactor shut-down will be initiated. Individual rods also can be tested by removing them from service one by one and dropping them into the core. The travel time is recorded and compared against the design specifications; the effect on the reactor can be seen via in-core instrumentation. By progressive testing of each part of the system and combination of the results using a standard system reliability model, the



Figure I - 29



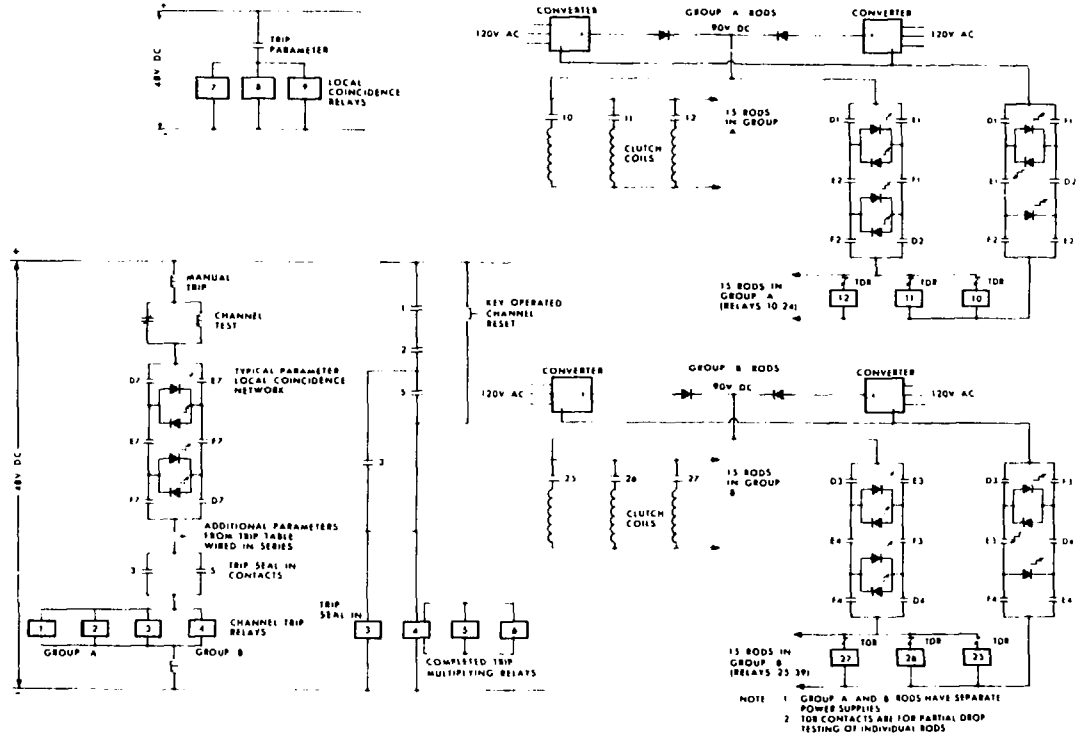
Location of Vertical Solid Reactivity Control Devices – Bruce Reactor

availability or readiness of the system to act on demand can be established. This derived availability must exceed 999 out of 1000 demands in order to satisfy requirements of the operating licence.

82. The Pickering B and Bruce B systems are identical in concept, with one exception. In Bruce B, SDS1 is activated by a local coincidence logic, in which two out of three channels of any single parameter must be exceeded in order to produce a reactor trip. In Pickering B, SDS1 is activated by a general coincidence logic, in which any two parameters exceeding the limits in two out of three channels will produce a reactor trip. The general structures of these logic systems are shown in Figures I-30 and I-31.

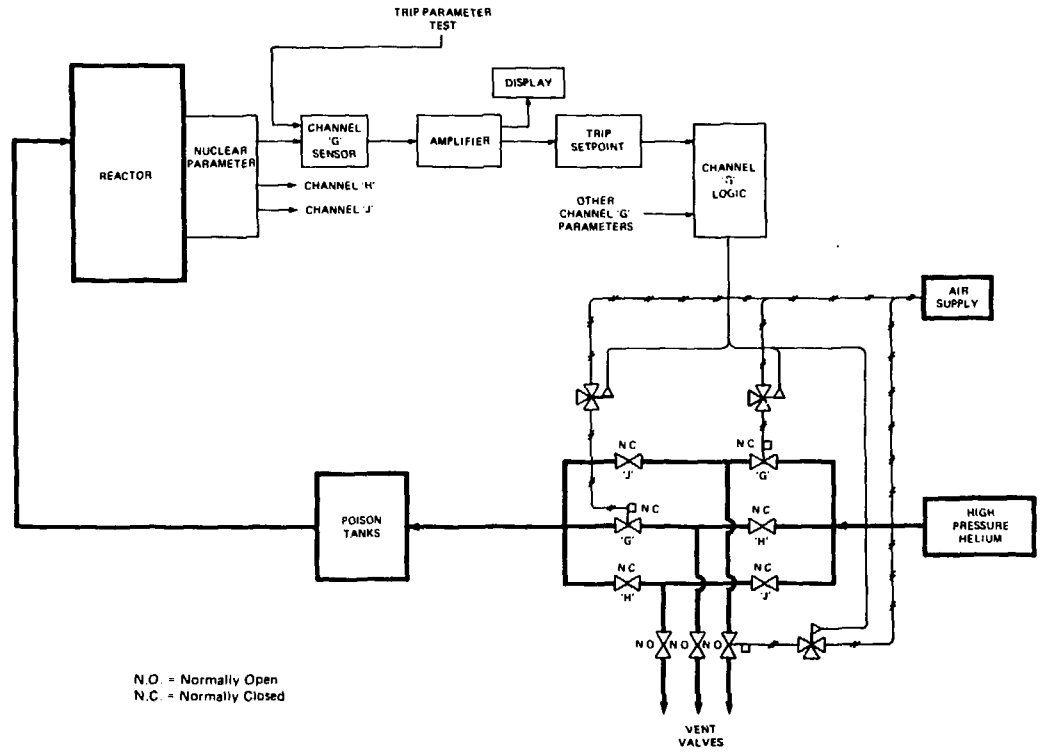
83. Pickering A incorporates a mixed general and local coincidence network to initiate an SDS1 trip; in this network, two out of three of the same parameter must exceed the limit for some of the parameters before a trip is initiated. Recalling that Pickering A was designed and constructed before the requirement for two independent shut-down systems was established, this is the only fast shut-down system installed in this station. When a trip signal has been issued, a dump arrest comparator begins to monitor the time dependence of the reactor power. If the shut-off rods do not reduce reactor power at least at the specified rate, a moderator dump is activated by connecting the helium volumes above and below the moderator tank, as shown in Figure I-7(b). Although somewhat slower in action than the shut-off rods, the moderator dump completely terminates the chain reaction in a few seconds. Because it is slower, the dump could not prevent fuel damage in the event of a very large pipe break in the primary system (large LOCA). For all other accident scenarios, it is fully capable of achieving safe shut-down. Even in the event of a large LOCA, several shut-off rods must be assumed to be unavailable before fuel damage would occur--long experience with reliability testing of Pickering A shut-down systems has shown that the rods achieve an availability of at least 9999 in 10 000 demands. Given independence of the two events, the estimated frequency of a large LOCA with simultaneous failure of shut-off rods is less than once in 10 million reactor-years. Allowance for unknown events, in which the failures might be related or consequential, suggests that about one

Figure 1 - 30



Local Coincidence Trip Logic System

Figure 1 - 31



N.O. = Normally Open  
N.C. = Normally Closed

General Coincidence Trip Logic System

order of magnitude should be discounted, leaving once in 1 million years as the frequency estimate for this event.

84. Bruce A systems are identical to Bruce B systems except for a slightly different set of initiating trip parameters. The Darlington trip parameter set and initiating logic is the same as that in Bruce B, except that the relay systems are replaced by digital computer routines that emulate the action of the relay systems installed in the other stations. The main advantages of the Darlington computerized shut-down system are improved reliability and ease of performance testing. In addition, given the fact that this station is expected to be operating up to about the year 2050, it is unlikely that the already-obsolete relays will be available through normal manufacturing channels for the full operating life of the station.

### (c) Shut-down system 2

85. The second shut-down system (SDS2) was originally considered as a means of providing a very fast shut-down following large LOCAs, as a back-up to the SDS1 system. During the licensing proceedings for Bruce A, however, the concept of SDS2 as a fully capable, independent shut-down system evolved from discussions between the designers and the AECB. Some backfitting was done to achieve this goal on Bruce A, and all subsequent stations meet this requirement. SDS2 is specific in the same manner as other special safety systems: i.e., it must be independent of process systems and other special safety systems. In addition, SDS2 must be independent from SDS1 and diverse in concept, design, construction, and operation.

86. SDS2 consists of several tanks containing a high concentration of gadolinium nitrate (a strong neutron absorber) in heavy-water solution. Each tank is connected by an open line to a perforated tube that runs horizontally through the reactor. The free interface between the concentrated neutron-absorbing solution and the clean heavy water of the moderator system is maintained by pressure-balancing lines connected between the top of the tank and the moderator cover gas. The general concept of SDS2 is shown in Figure

I-32; the arrangement of in-reactor gadolinium injection nozzles is shown in Figure I-33.

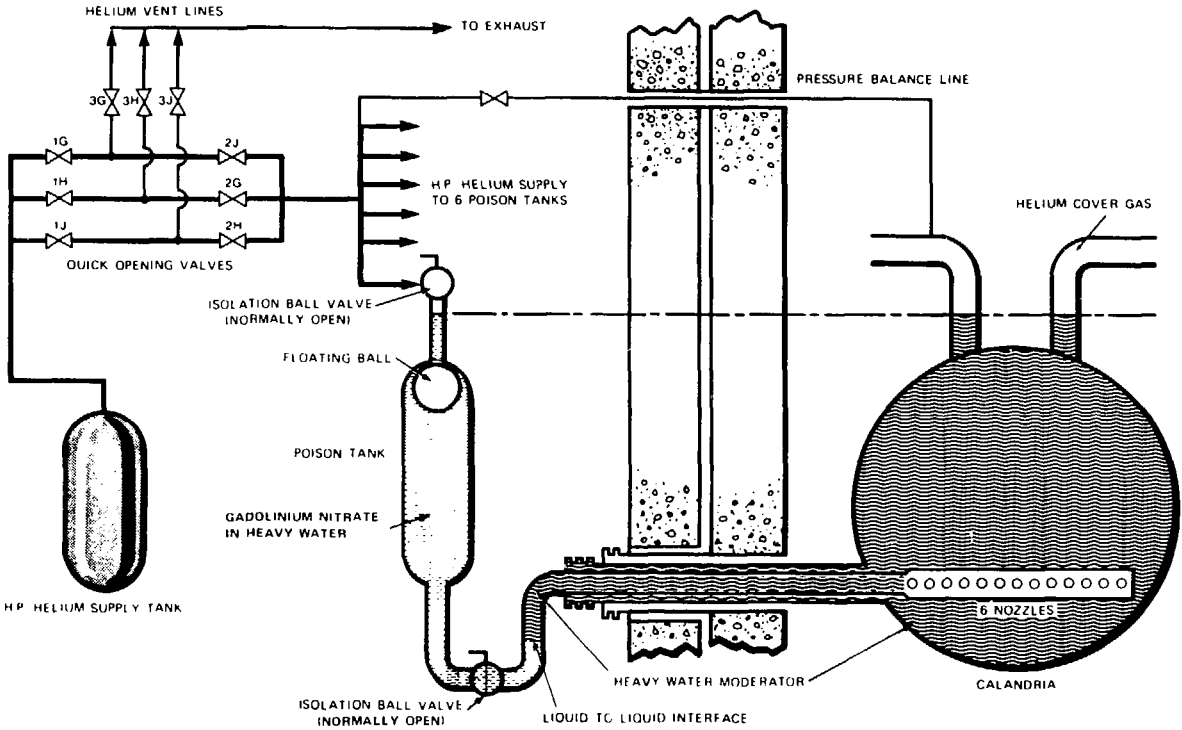
87. The system is activated by opening isolation valves between the tops of the gadolinium tanks and a tank of high-pressure (about 15 MPa) helium. These helium valves are arranged so that activation of two out of three trip channels will provide a helium flow path. The solution is forced into the perforated tubes and ejected out of the array of orifices into the moderator water. The detectors and trip logic that activate SDS2 are completely independent of those activating SDS1. The time from cut-off of clutch voltage to complete shut-down is about 1 s.

88. Testing of the system is carried out in much the same way as for SDS1, except that the individual tanks are not normally emptied into the reactor (this would shut down the reactor for about 40 h). Because the connection between the gadolinium tanks and the moderator water contains no closed valves, this particular test is unnecessary. The concentration of gadolinium in the tanks is tested periodically by sampling and laboratory analysis, and the tanks are refilled with fresh gadolinium solution at regular intervals.

89. Recent experience with operation of SDS2 shows that the system can achieve an availability of 9999 in 10 000 demands, similar to SDS1. Early experience in the Bruce A units was not good, because of the novelty of the system and some minor design weaknesses. These weaknesses have been detected and corrected. Both SDS1 and SDS2 are subjected to full-scale testing during commissioning of each reactor; the systems are tripped from high reactor power, and the rate of power run-down is measured at several locations. These measurements compare very well with calculated run-down characteristics. Typical reactor multiplication traces for both SDS1 and SDS2 are shown in Figure I-34.

90. The Pickering B reactor contains six injection tanks; because it is larger, Bruce B contains seven. The Bruce A reactors also contain seven injection tanks. The Darlington design incorporates eight injection tanks and so achieves

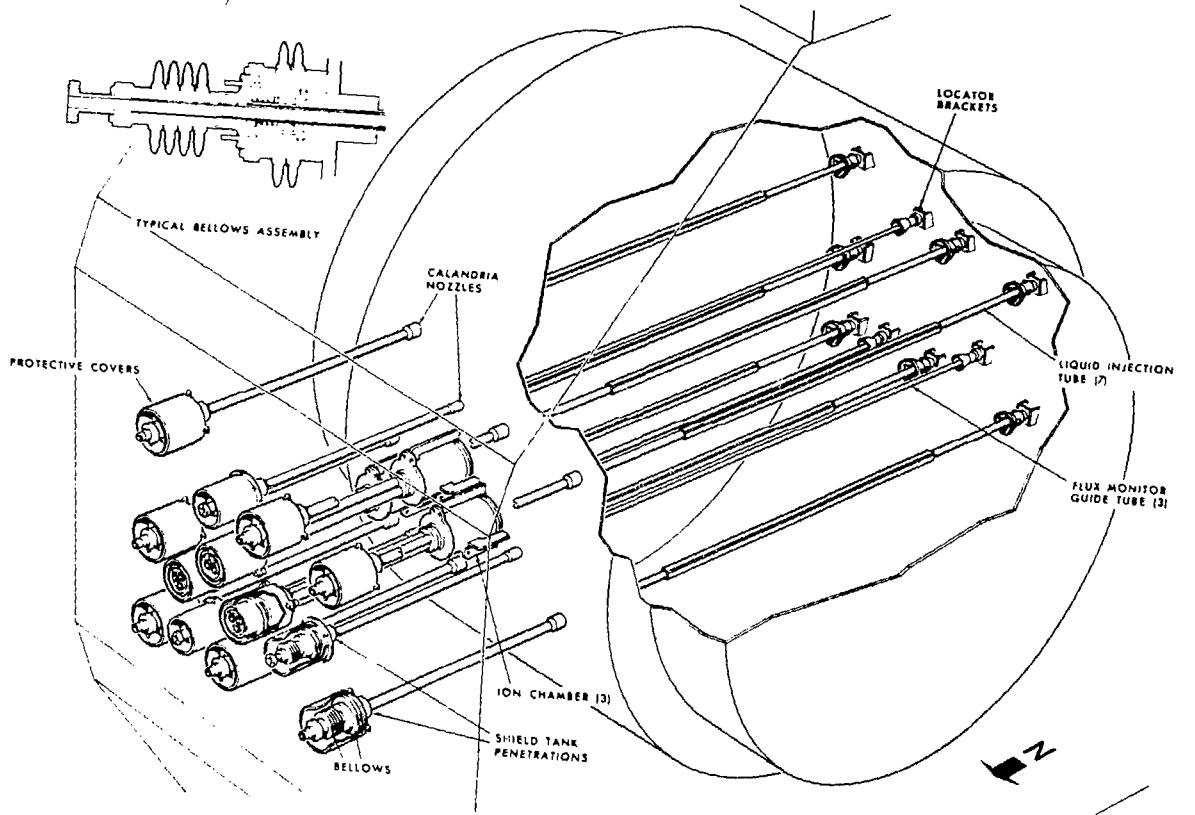
Figure I - 32



Schematic of Second Shutdown System Piping

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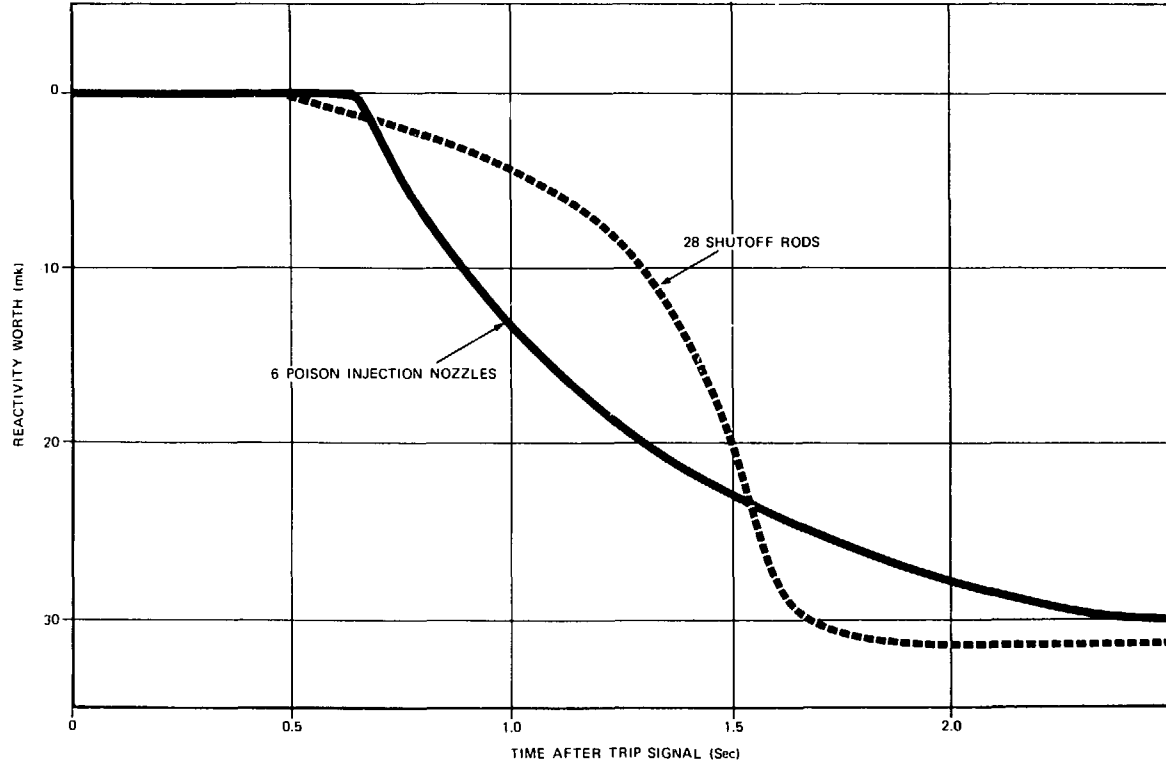
Figure I - 33



Layout of Piping for Second Shutdown System



Figure 1 - 34



Reactivity Versus Time for First and Second Shutdown Systems Following a Trip at 0.0 Seconds

a slightly higher level of excess shut-down capability during single-tank outage during testing and refill.

**(d) Emergency cooling**

91. Emergency cooling of the fuel, in the event of loss of normal cooling via the heat transport circuit, is necessary to remove heat produced by radioactive fission products after reactor shut-down. CANDU systems have two means of removing this heat: the ECIS (which refills the heat transport circuit so that normal cooling can be re-established), and the moderator surrounding the fuel channels (which has an independent heat removal circuit). The first of these systems is preferred because of its superior ability to limit the release of fission products from the fuel to the containment space.

92. The only role of the ECIS is to refill the primary heat transport system following a LOCA, which is defined as an accident involving loss of integrity of the heat transport system boundary. Note that a loss of integrity could be due to either a pipe break or failure of a valve. The ECIS must be driven by high-pressure pumps or other means of ensuring water flow through the channels and by a heat removal system, in order to carry out its emergency core cooling function. The ECIS has never been called into service in any CANDU reactor up to the present. Nevertheless, the system is designed and regularly tested in the field to the same reliability standard as the shut-down systems.

93. The smallest pipe break size for which the ECIS is required is the maximum size at which the heavy-water feed system, which supplies water to the heat transport system, can maintain pressure control. The largest failure size for which the ECIS is designed corresponds to severance of the largest pipe in the heat transport system. The ECIS operates in two or three stages. Following detection of a LOCA (identified by low pressure in the heat transport system combined with one of several other possible signals), high-pressure ordinary water is injected through piping connected to each of the heat transport system headers. This injection phase is sometimes carried out in two

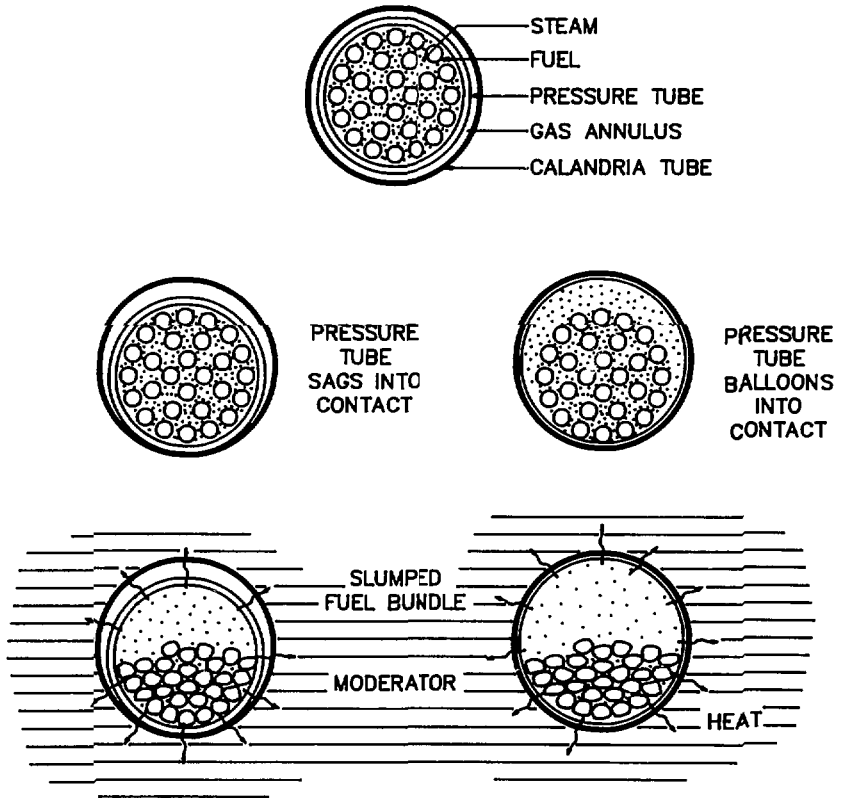
distinct stages. The long-term stage involves recovery of water from the reactor building floor and reinjection into the reactor.

94. The ECISs are designed either as stored-energy systems or as motive-power systems. The main advantage of a stored-energy system is that it requires no external power source to establish water flow immediately after a LOCA. Its main disadvantage is that the amount of stored energy is necessarily limited, so that it can act for only a limited period before a motive-power system must take over the task of keeping water in the heat transport system.

95. A motive-power ECIS has exactly reverse advantages and disadvantages. Once initiated, it can operate for as long as motive power is available; on the other hand, it relies on power being available to pump water into the heat transport system even immediately following the LOCA. Therefore, choice of one or the other of these systems depends to some extent on the reliability of short-term power supplies on the site.

96. Fuel cooling via the moderator system would take place only if the ECIS failed to respond on demand. (In extreme cases, a few channels may reject heat to the moderator for a short time even when emergency injection is working. Some fuel damage might occur in these cases, but no severe consequences would result.) Heat transfer would occur after heat-up of the pressure tube and either gravity-driven sagging or pressure-driven ballooning of the pressure tube into contact with the calandria tube. These two limiting cases are shown in Figure I-35. Many experiments done on out-reactor test channels heated with electricity have proved two important points. First, heat would be transferred efficiently from the fuel across the pressure tube/calandria tube boundary to the moderator; there would be no fuel melting. Second, the fuel channel structure would remain intact, thereby establishing a stable heat transfer configuration; decay heat would be removed via the moderator heat exchangers. Even though cooling would be ensured, under these conditions the relatively high fuel temperatures would result in considerable release of fission products to the containment space.

Figure I - 35



Heat Transfer Modes for Assumed Loss of Coolant Accident  
With Failure of Emergency Water Injection

Ontario Hydro

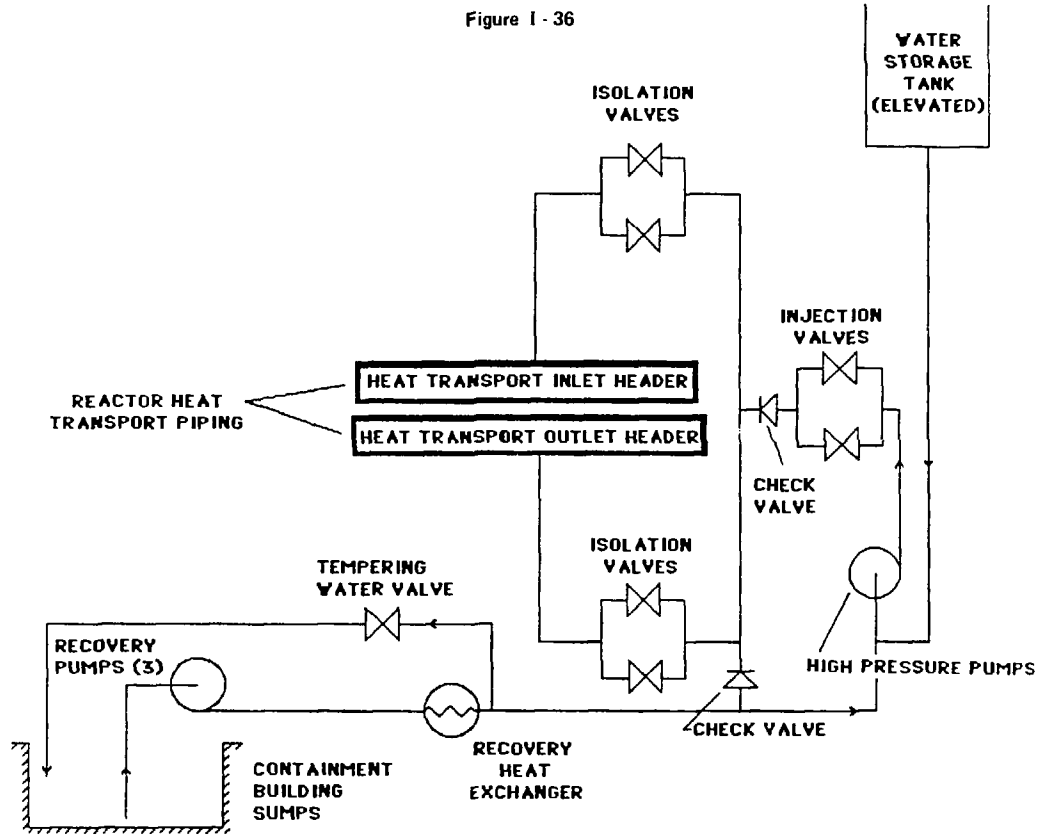
97. The Pickering B ECIS is a motive-power system driven either by site electrical power or by emergency generators. It has a lower initial flow capacity than the Bruce systems, but can continue to supply high-pressure water to the heat transport system indefinitely. An elevated storage tank supplies water to the inlet of the pumps for a limited period; recovery pumps then draw water from the floor of the containment building in the accident unit to supply the injection pumps. The general schematic of this system is shown in Figure I-36.

98. The system designed for the four Pickering B units is being extended to supply water to the Pickering A units as well. When this backfit is completed, there will be one high-pressure injection system for all eight Pickering units. The Pickering A system operates differently from Pickering B in the long term; in these units, the recovery pumps deliver medium-pressure water directly to the heat transport system of the accident unit, as shown in Figure I-37.

99. Bruce A and Bruce B have stored-energy injection systems, with sufficient storage to inject water without external power sources for between 3 min for a large pipe break and 40 min for a small break. Following this injection period, low-pressure pumps recycle water to the heat transport system either from a storage tank or, later, from the containment building floor. These pumps can be driven either by normal site electrical power or by emergency generators operated by gas turbine engines. There is one water delivery train for four reactor units. A general schematic of this system is shown in Figures I-38(a) and I-38(b). The Bruce A system was backfitted to the plant after initial operation in order to increase the fuel-cooling capability of the injection system.

100. Darlington utilises a motive-power injection system similar in concept to the Pickering system. The major difference lies in the higher flow capacity of the Darlington system; this is necessary because of the larger capacity and piping sizes in the Darlington heat transport system. The system schematic diagram is shown in Figure I-39.

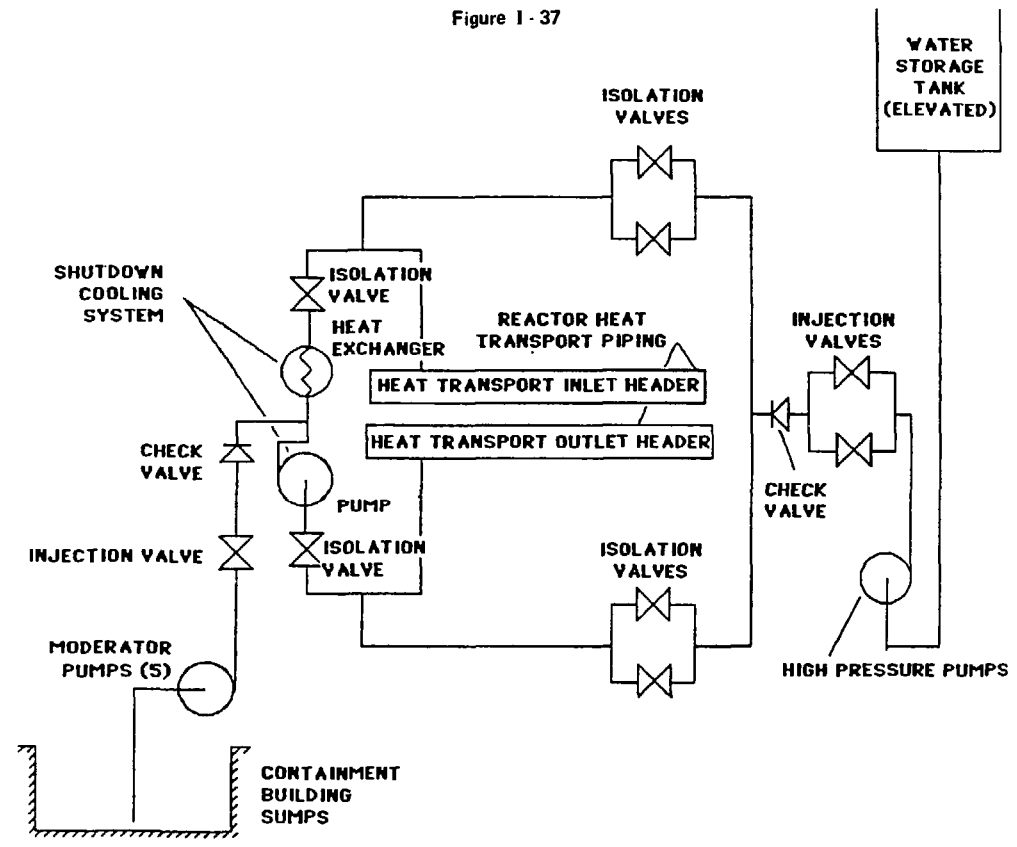
Figure I - 36



Pickering B Emergency Coolant Injection Schematic

D A. Meneley Assoc. Ltd.

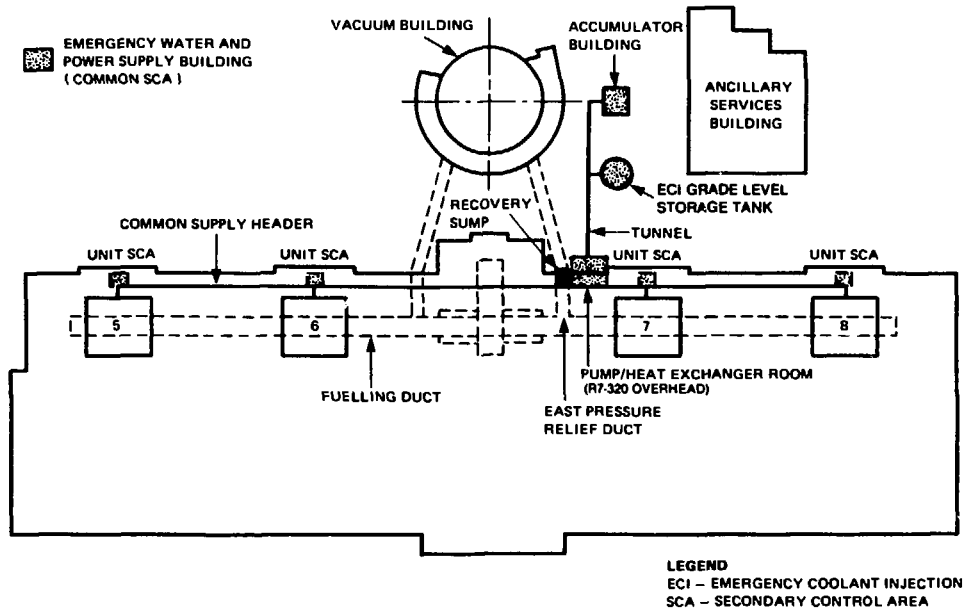
Figure I - 37



Pickering A Emergency Coolant Injection System Schematic

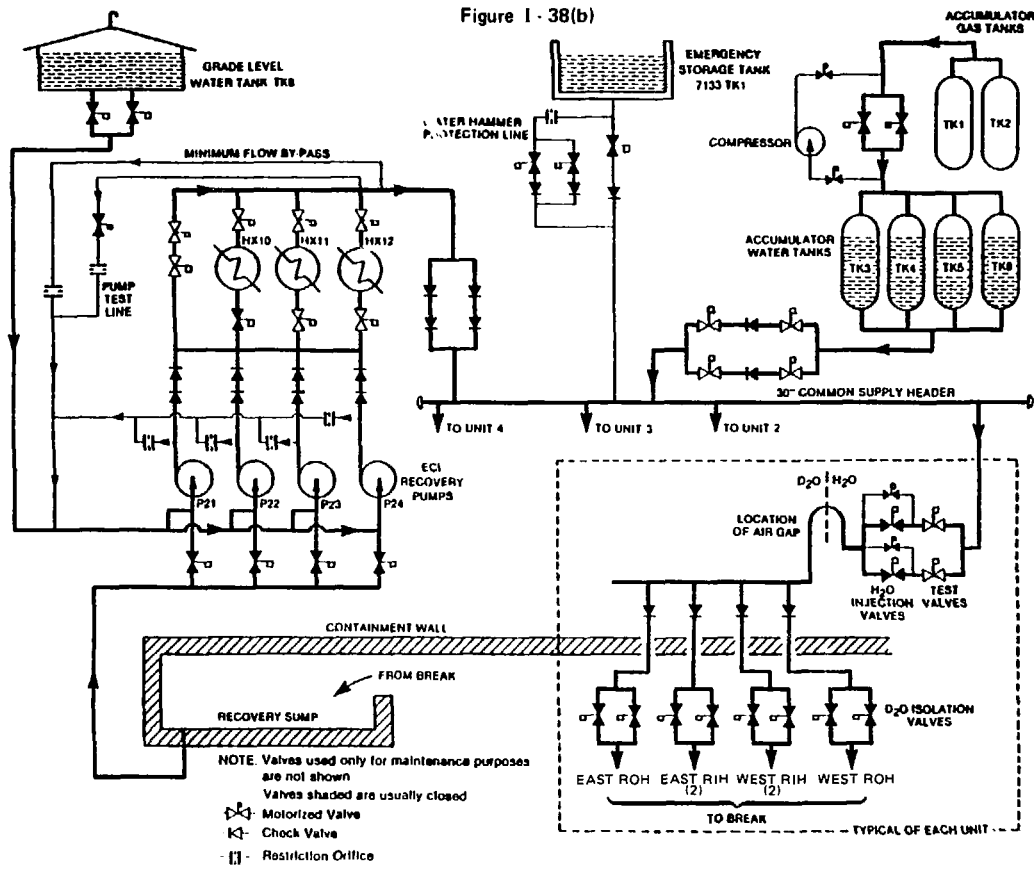
D.A. Meneley Assoc. Ltd.

Figure I - 38(a)



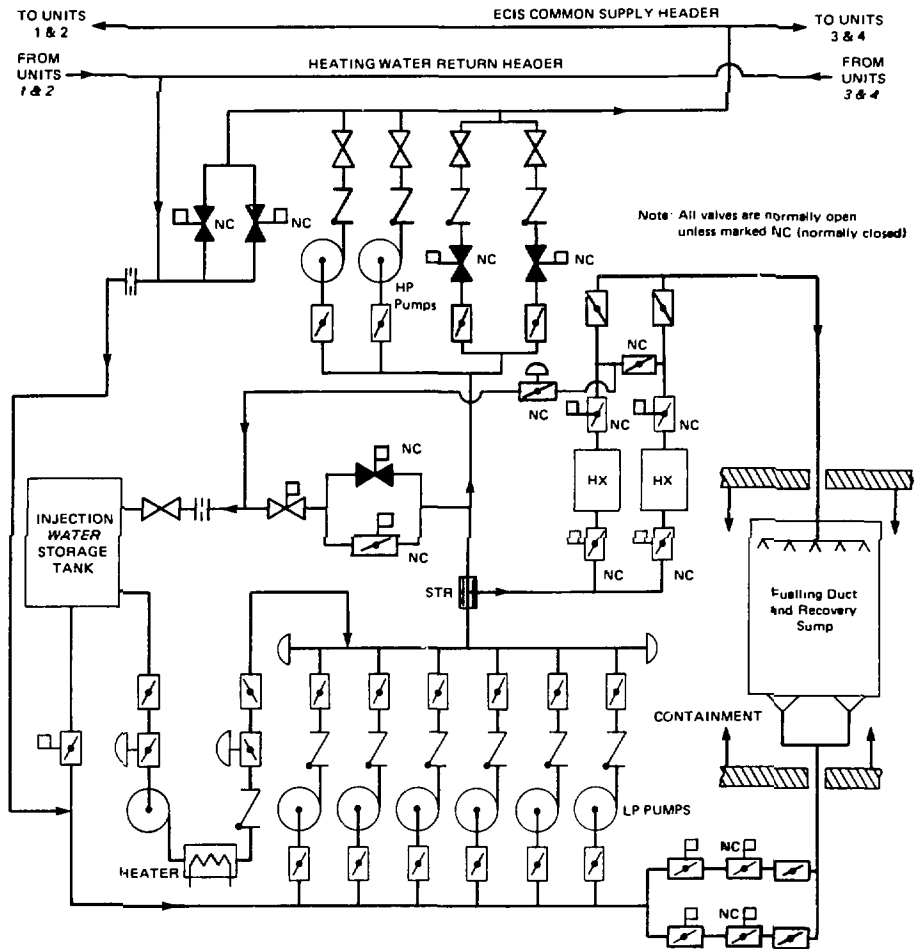
Bruce Emergency Injection - General Layout





Bruce Emergency Coolant Injection System – Pump and Piping Layout

Figure I - 39



Darlington Emergency Coolant Injection System  
Pump and Piping Layout

Ontario Hydro

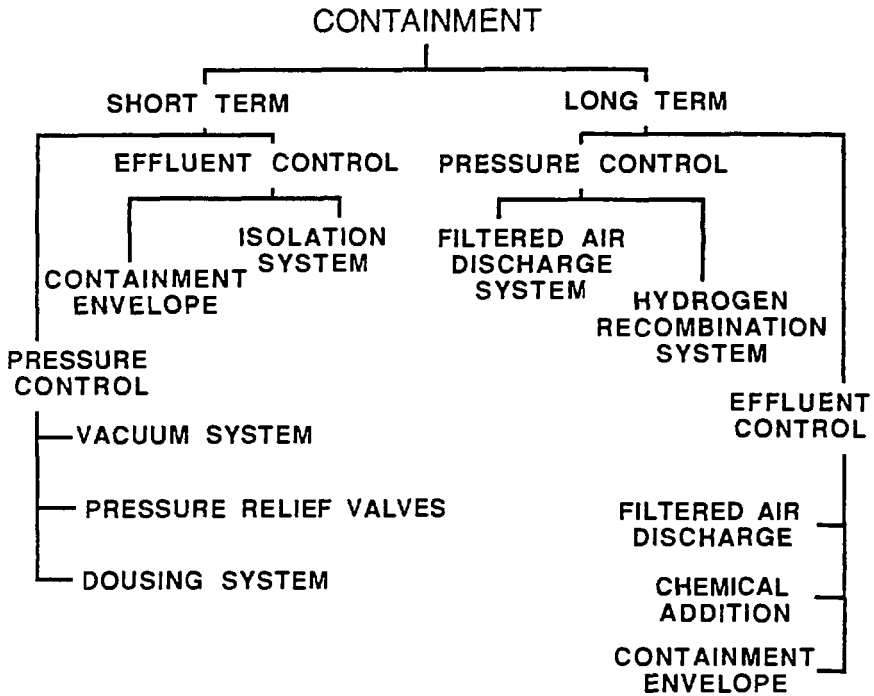
### (e) Containment

101. Containment is a special safety system; it has no important function during normal operation. The containment boundary encloses the piping of the primary heat transport system. Following an accident, its overall function is to prevent release of radioactive materials to the environment. Potentially dangerous quantities of these materials are sealed inside the fuel elements, which are located inside the primary heat transport system boundary. It follows that failure of these boundaries is a necessary pre-condition to release of radioactive materials into the containment space. Large numbers of fuel elements will not fail unless the reactor is not shut down promptly or, if it is shut down, unless the ECIS does not work.

102. Given a failure of the heat transport system boundary (LOCA), high-temperature water and steam will be released into the containment space. In this event, containment subsystems must perform a number of related functions, as shown in Figure I-40. The first of these is to ensure that the resulting pressure inside the containment remains below the design pressure of the structure. The building either is made large enough so that the peak pressure simply stays below the design pressure (in this case, between about 200 and 600 kPa) or is designed to control the pressure in some way--usually by relief or steam suppression. All CANDU reactors utilise the pressure suppression concept, with design pressures in the range between 140 and 200 kPa. Pressure suppression containment systems act to rapidly condense the steam inside the building and hence reduce the pressure. At the same time, the building ventilation system, which is essential during normal operation, is closed by automatic valves operated by high-pressure or high radiation level detectors.

103. Ontario Hydro multi-unit stations incorporate a unique containment system that combines pressure relief and pressure suppression. The main containment building is connected through large pressure-activated valves (as shown in Figure I-41) to a vacuum building, which is held at a very low pressure (about 15 kPa absolute) at all times. Following a LOCA the valves are forced open and the steam/air mixture flows from the reactor building into the vacuum

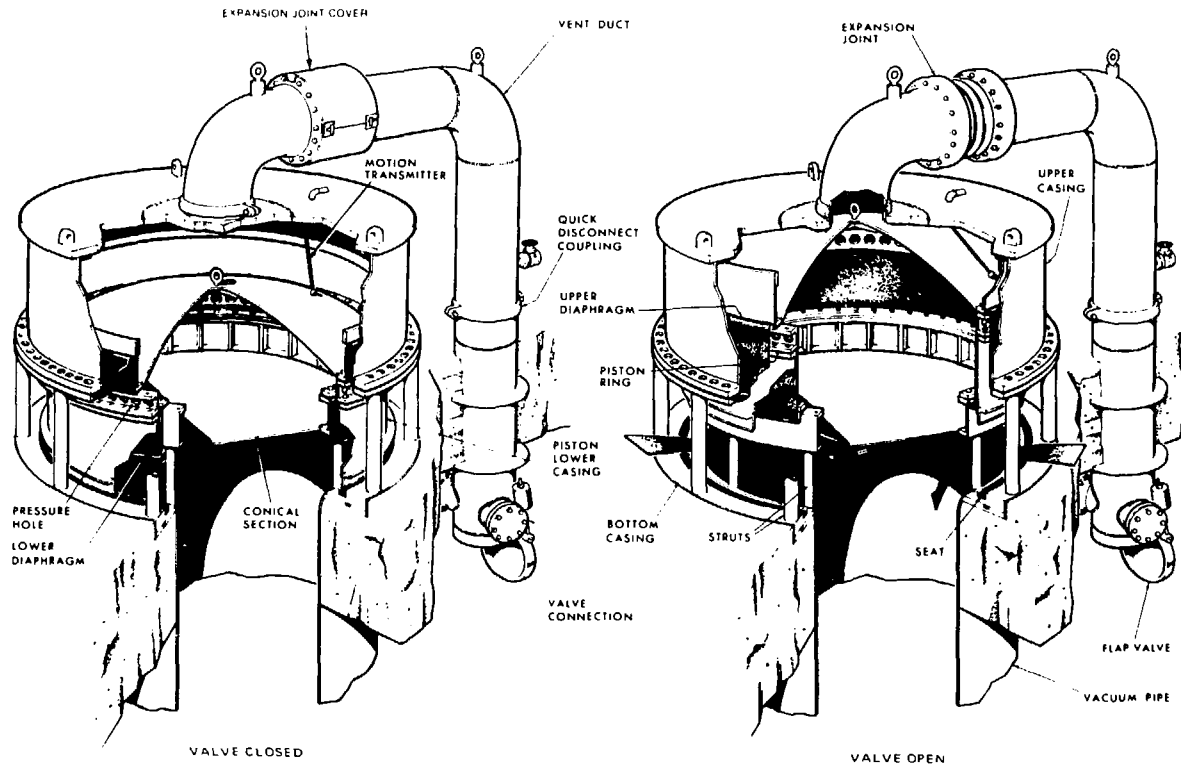
Figure 1 - 40



Containment Functional Requirements and Systems  
Used to Meet Those Requirements

D. A. Meneley Assoc. Ltd.

Figure I - 41



Pressure Relief Valve – Bruce NGS

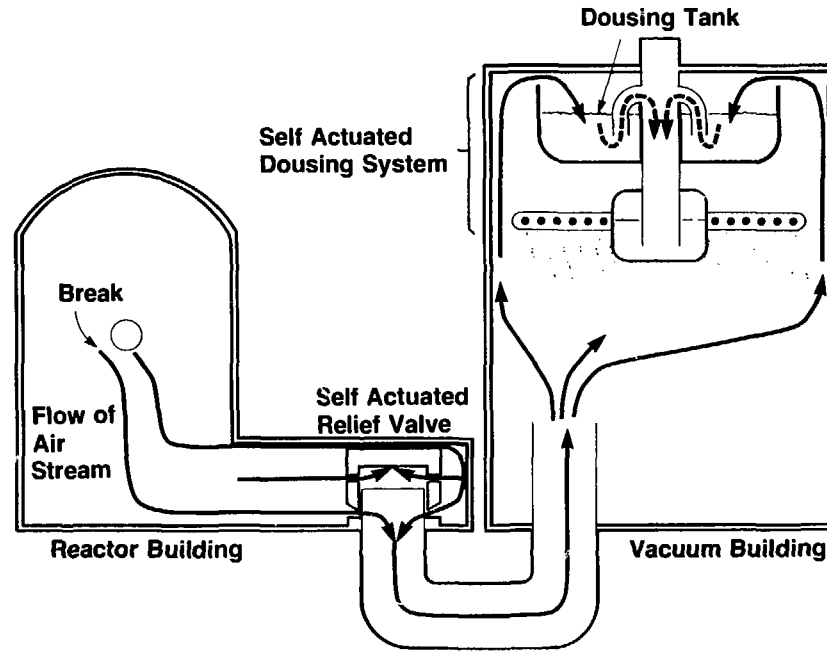
building. Within this vacuum building, the pressure rise following opening of the pressure relief valves initiates a dousing system fed from an elevated water tank. Steam is condensed. The reactor building returns to a pressure below atmospheric pressure in about 1 min. Naturally, airborne radioactive materials will not then leak out of any openings in the building into the outside air. The layout and concepts of operation of the vacuum system are shown in Figure I-42.

**104.** In the longer term, the objective of containment systems is to retain all the trapped radioactive materials. Because of the possibility of pressure rise inside the building in the long term, a filtered-air discharge system is installed. This system discharges air from the building through air driers, charcoal beds, and high-efficiency particulate filters, in case this becomes necessary. Essentially, only noble gases pass through these filters; because these gases do not form chemical compounds, they disperse in the atmosphere and are relatively harmless to biological systems. Residual steam inside the building is condensed by the vault air coolers in the long term; hydrogen-oxygen recombiners remove combustible hydrogen from the atmosphere.

**105.** All Ontario Hydro multi-unit stations incorporate a single containment building with a single vacuum building. Each reactor building is connected to the vacuum building via a pressure relief duct, which runs from one end of the station to the other.

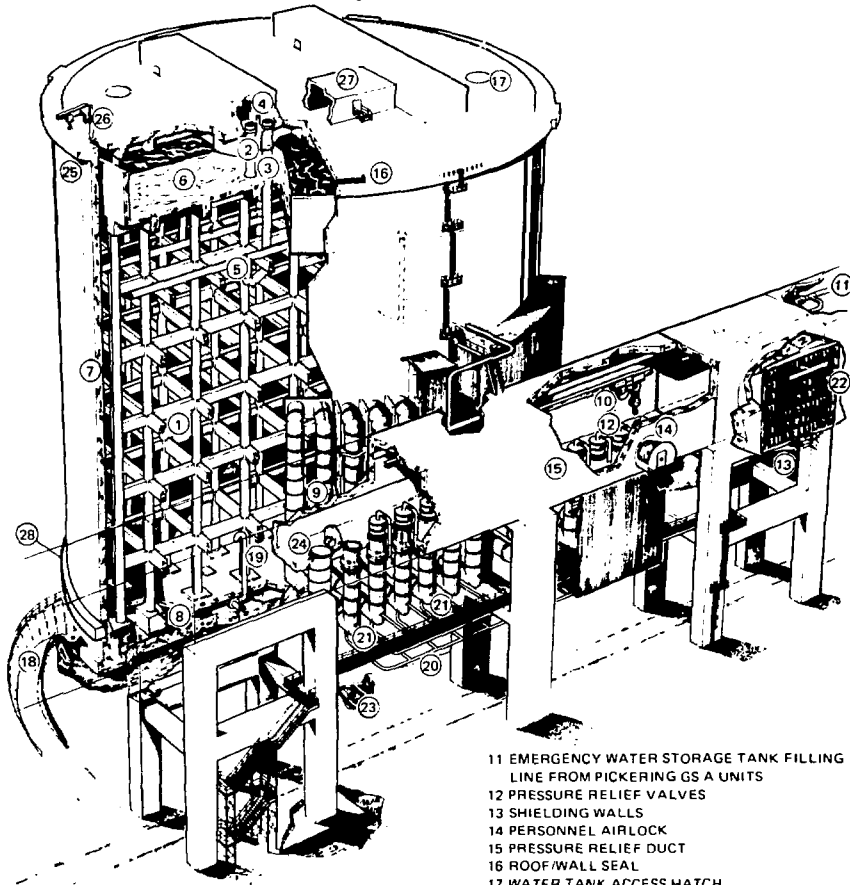
**106.** The major difference between the Pickering and Bruce containment concepts lies in the connection between the individual reactor building and the pressure relief duct. In the Pickering concept, this interface is closed by non-return valves, which would prevent flow of the steam/air mixture from the duct to a non-accident reactor unit following a LOCA. In the Bruce concept, there is no such non-return valve; the reactor buildings are all interconnected during normal operation. The Darlington design is an adaptation of the Bruce concept. The layouts of Pickering and Bruce containment systems are shown in Figures I-43(a), I-43(b), and I-43(c). Figures I-44(a) and I-44(b) illustrate the Darlington concept.

Figure 1 - 42



The Vacuum Containment Concept

Figure I - 43(a)



- 1 VACUUM BUILDING INTERNALS
- 2 PRESSURE ACTUATED WATER DISPLACEMENT SYSTEM INLET HEADER
- 3 PRESSURE ACTIVATES WATER DISPLACEMENT SYSTEM OUTLET HEADER
- 4 VACUUM CHAMBER
- 5 DISTRIBUTION AND SPRAY HEADER
- 6 EMERGENCY WATER STORAGE TANK
- 7 PERIMETER WALL
- 8 BASEMENT
- 9 VACUUM DUCTS
- 10 MONORAIL AND HOIST

- 11 EMERGENCY WATER STORAGE TANK FILLING LINE FROM PICKERING GS A UNITS
- 12 PRESSURE RELIEF VALVES
- 13 SHIELDING WALLS
- 14 PERSONNEL AIRLOCK
- 15 PRESSURE RELIEF DUCT
- 16 ROOF/WALL SEAL
- 17 WATER TANK ACCESS HATCH
- 18 BASEMENT ACCESS RAMP
- 19 VACUUM PUMP SUCTION HEADER
- 20 VACUUM DUCT DRAIN PIPE
- 21 VACUUM DUCT FILL PIPE
- 22 REACTOR BUILDING PRESSURE RELIEF PANELS
- 23 SERVICES TUNNEL
- 24 EQUIPMENT AIRLOCK
- 25 PERIMETER WALL MONORAIL
- 26 JIB CRANE
- 27 VACUUM BUILDING INSTRUMENT ENCLOSURE
- 28 REINFORCED CONCRETE RING

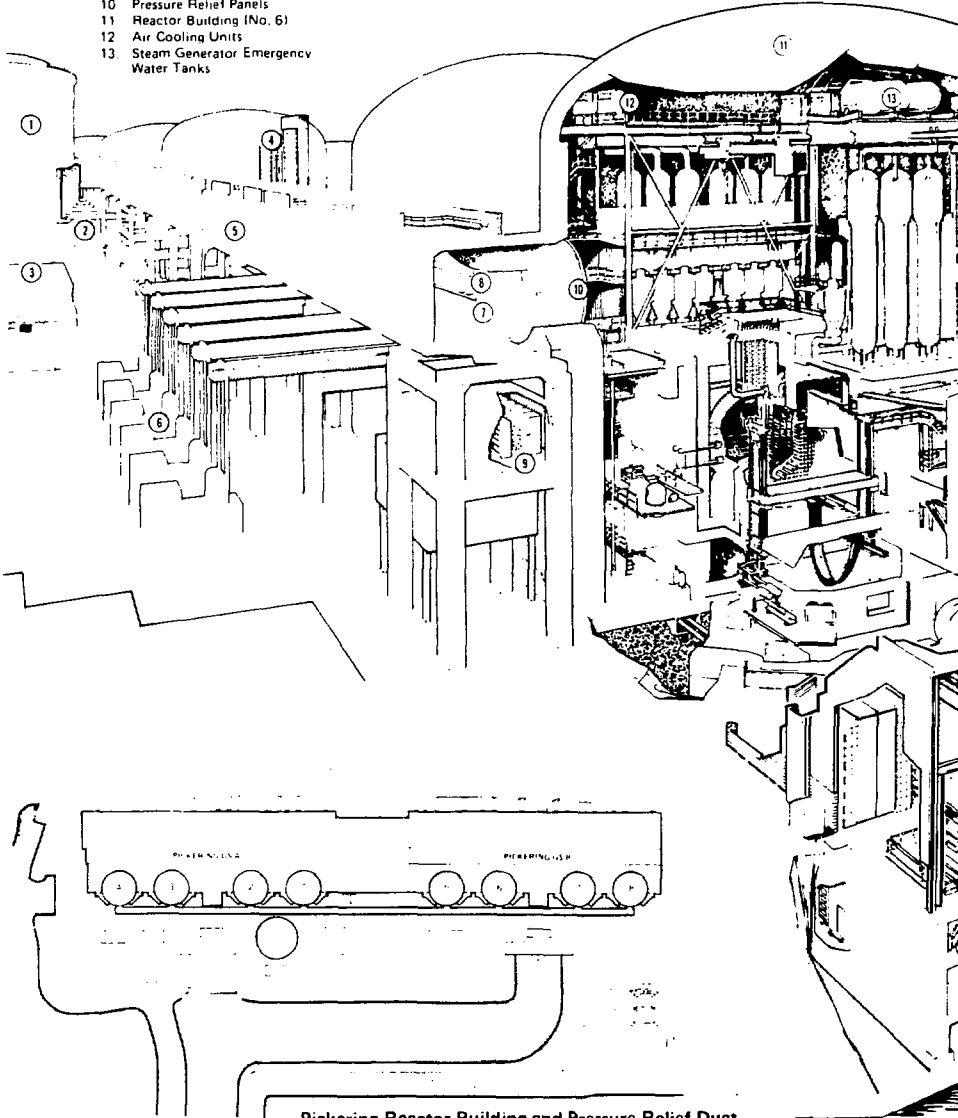
Pickering Vacuum Building

Ontario Hydro



Figure I - 43(b)

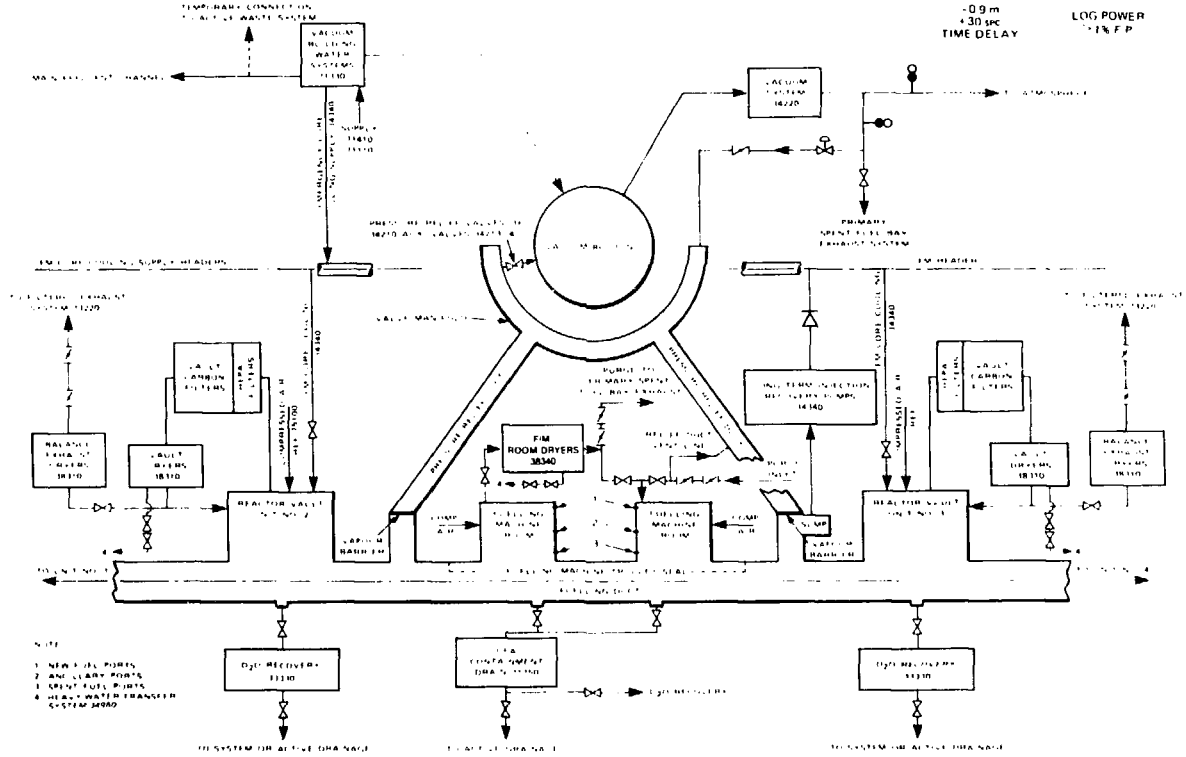
- 1 Vacuum Building
- 2 Vacuum Ducts
- 3 Water Treatment Building
- 4 Heavy Water Upgrading Towers
- 5 Pressure Relief Duct
- 6 Standby Power Generators
- 7 Emergency Coolant Injection Lines
- 8 Emergency Storage Water Supply Line
- 9 Unit Emergency Control Centre
- 10 Pressure Relief Panels
- 11 Reactor Building (No. 6)
- 12 Air Cooling Units
- 13 Steam Generator Emergency Water Tanks



Pickering Reactor Building and Pressure Relief Duct

Ontario Hydro

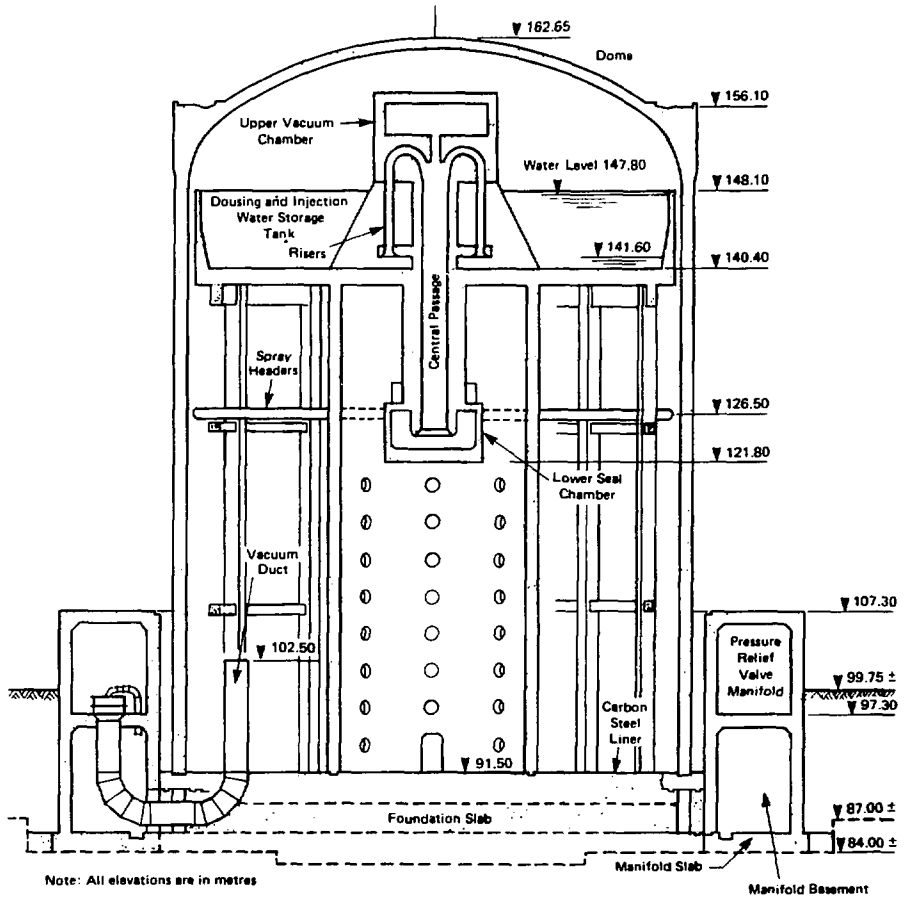
Figure I-43(c)



-0.9 m  
+30 sec  
TIME DELAY  
LOG POWER  
T/F P

Bruce Containment Envelope

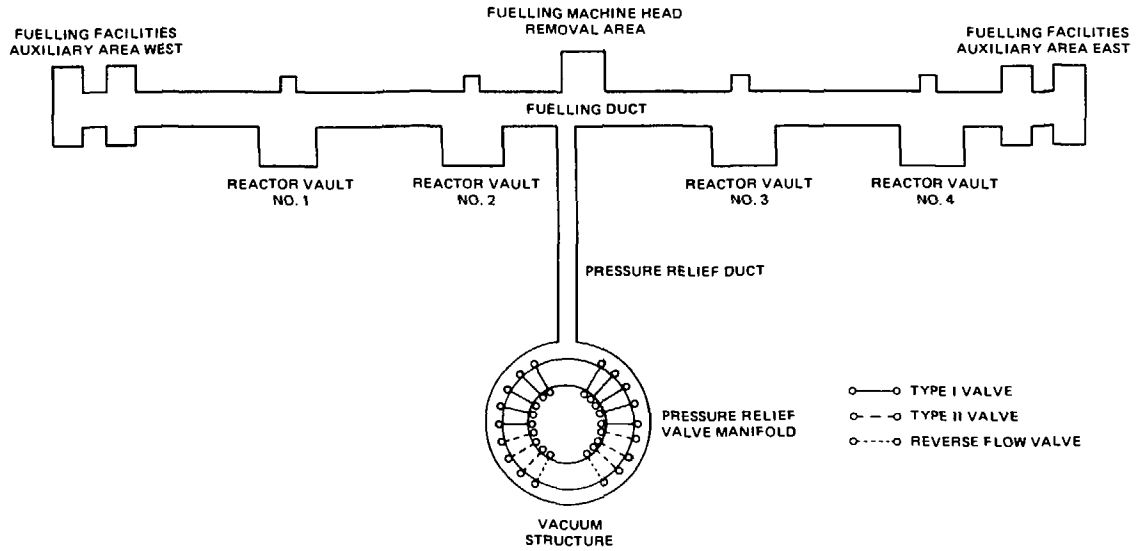
Figure 1 - 44(a)



Darlington Vacuum Building

Ontario Hydro

Figure I - 44(b)



Darlington Containment Envelope

### **(f) Separation of systems**

107. To guard against accident events such as fire and earthquake, which might involve a large area of the station, safety system control equipment and cabling are separated physically and functionally. The purpose is to ensure that, after any accident, at least one means is available to shut down the reactor, remove decay heat from the fuel, and monitor post-accident conditions. The Group 1 systems are SDS1, normal power and cooling water supplies, and main control room. The Group 2 systems are SDS2, emergency power and water systems, and a secondary control area. Group 2 systems can be operated from the secondary control area in the event that the main control room is unusable.

108. The degree of separation and protection of safety systems has increased with time as the safety analyses and licensing requirements have become more sophisticated. Pickering B, Bruce B, and Darlington all have a high degree of system separation. The older stations, Pickering A and Bruce A, are less fully protected in this sense. Even in these stations, there was a considerable amount of attention given to physical separation and protection of essential functions; recent design reviews have resulted in relatively minor changes.

### **(g) Probabilistic safety evaluation**

109. During the past 10 yr, accident analysis has been extended to a large number of potential accident sequences, and consideration of the effect of support systems on the progression of accidents has been added. Probabilistic evaluation has been developed to organise the large number of possible accident sequences and to detect potential weaknesses in the detailed design of safety systems. Depending on the expected frequency of occurrence and the potential consequences, new systems or procedures may be developed to mitigate the effects of these accident sequences. It is a characteristic of this analysis that the detailed design must be essentially complete before probabilistic evaluation is done in order for the method to be very effective.

110. Pickering B and Bruce B were analysed using an early form of this technique, the so-called safety design matrices. Fifteen major station systems were analysed separately using detailed reliability models of the system components. Several design changes were adopted as a result of this work.

111. Darlington is the first CANDU station to which a fully detailed probabilistic evaluation has been applied. Ontario Hydro is currently carrying out the same type of evaluation on all its operating stations.

## **E. Safety Support Systems**

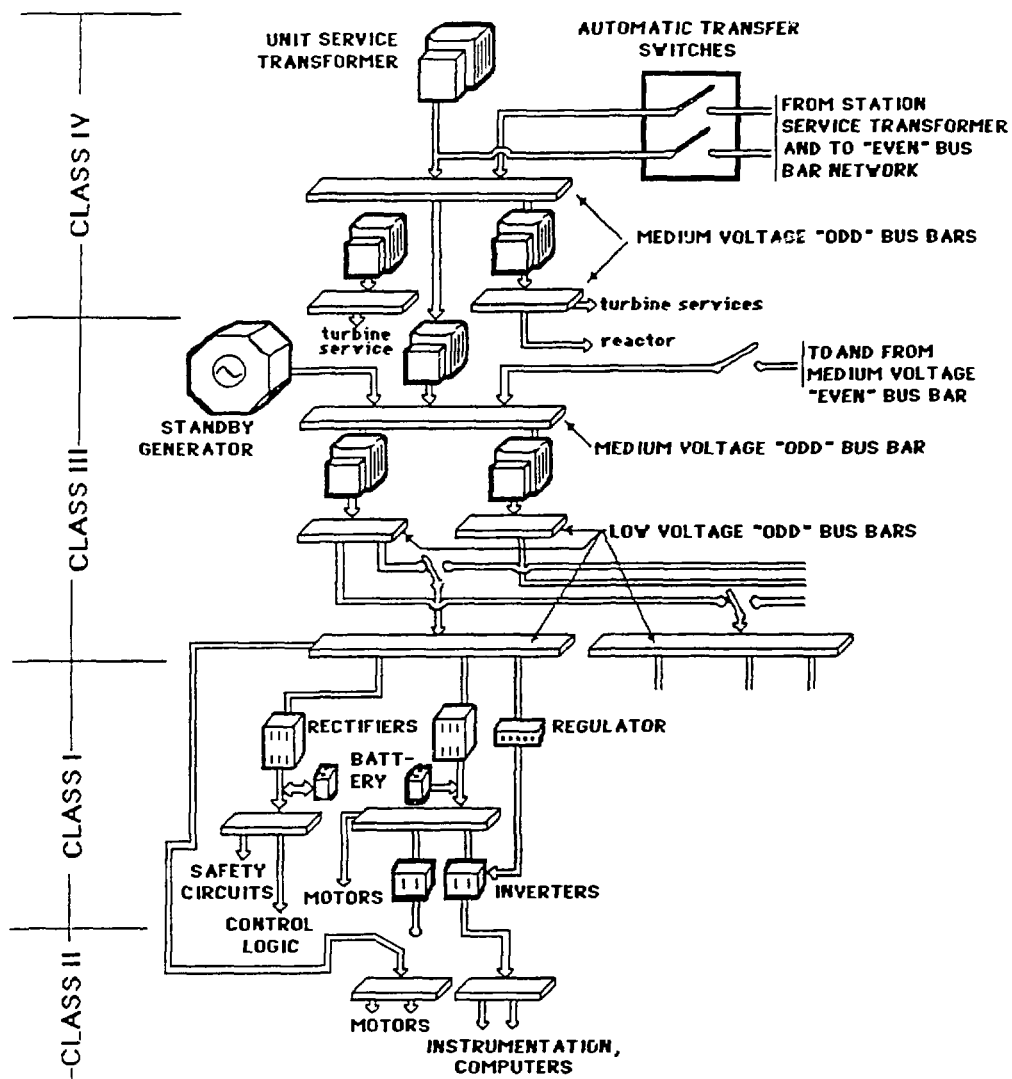
112. These systems include all those that are necessary to some degree to the proper functioning of safety systems after an accident and to the monitoring of the state of the accident unit. The type of system installed and the degree of redundancy built into the system depend on its relative importance to fuel cooling and containment. Reactor shut-down requires essentially no safety support because of its short action time and stable final state.

### **(a) Electrical power--station service systems**

113. Electrical power is essential to the functioning of all pumps in the station, as well as many control and monitoring systems. CANDU station service systems are divided into four classes. These classes are defined by the reliability requirements of various components in the station. A simple schematic of a typical station service system is shown in Figure I-45.

114. Class IV station service power consists of the normal power supplies, driven by the unit generator or from the station switchyard. The arrangement of power supply to the Class IV system is shown in Figure I-46. Ontario Hydro's multi-unit stations have several units feeding power to the switchyard, so that the units that have suffered no accident are capable of delivering Class IV power to the accident unit.

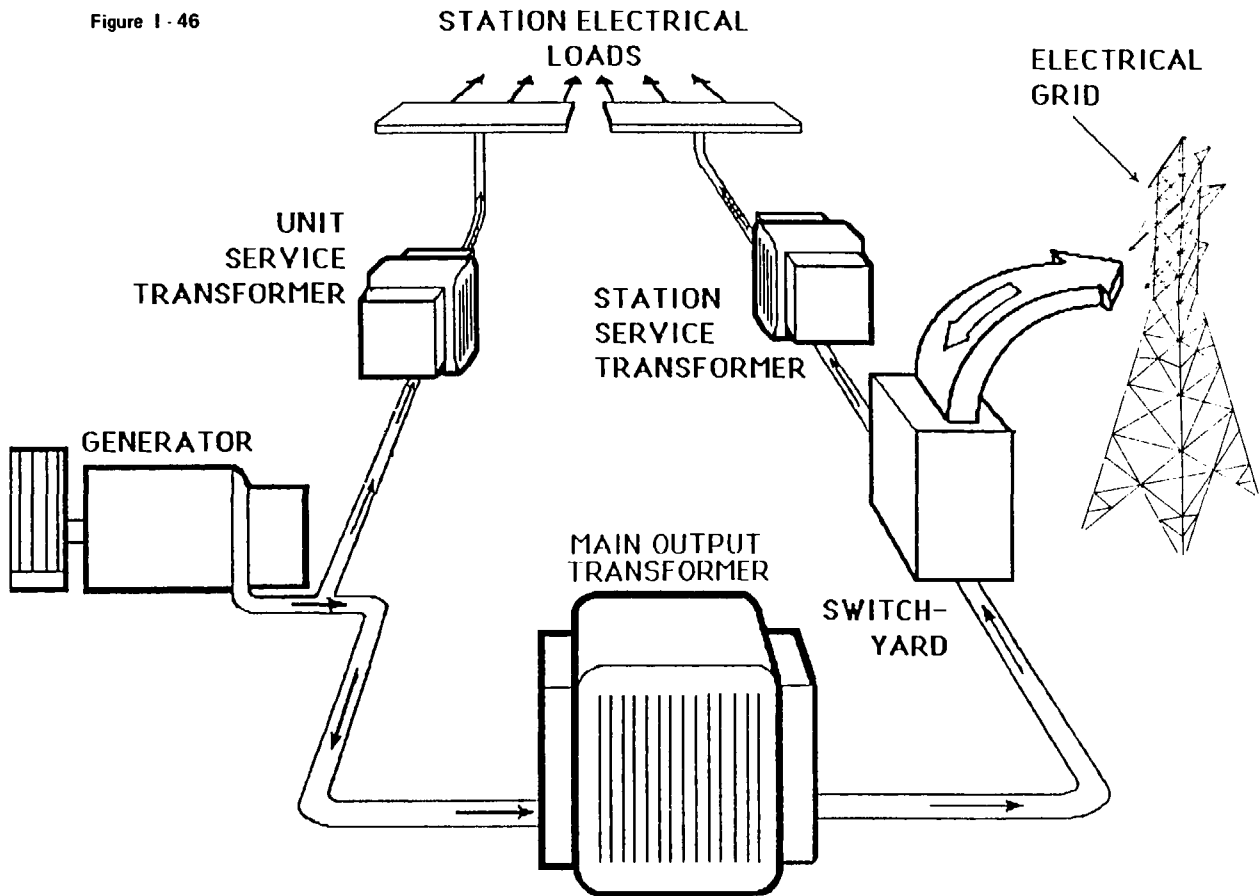
Figure 1-45



Typical Station Electrical Service Layout  
 -Class I is Highest Reliability

D. A. Meneley Assoc. Ltd.

Figure 1 - 46



Class IV Power System Layout



115. The highest reliability requirement is placed on Class I (direct current) and Class II (alternating current) supplies. In the absence of other power, these are driven by storage batteries capable of supplying essential loads for several hours. Because these essential loads are quite small in total, the battery supplies could be augmented quite easily, if required, by conventional generators stored at the station.

116. One of the potential emergency situations is that which might follow an initial accident coupled with complete loss of normal and emergency power supplies. CANDU stations are designed so that the probability of this event is extremely small. This is especially true on multi-unit sites where there is a great diversity of power supplies available. The so-called station black-out situation is below the minimum probability requirement for station licensing. Nevertheless, were the situation to occur, fuel damage would not begin for several hours because of the large amount of stored cooling water available. This would allow time for remedial action. Releases of radioactive material from the station probably would not occur at any time. The reason is that decay heat decreases steadily following reactor shut-down, so that the amount of heat that must be removed from the fuel to prevent overheating becomes very small after a few hours.

117. The Class III power system is fed power either from the Class IV system or from emergency generators. These generators start automatically following some accident signals; they are capable of delivering electricity to the station bus bars within 1-3 min.

118. Class IV power is delivered to station services either through the unit service transformer or through the station service transformer, which is connected directly to the switchyard. Automatic transfer switches are provided in case of loss of power to the unit service bus bar.

119. Independent of the normal power systems, a seismically qualified emergency power supply system is provided to drive essential systems after a

hypothetical earthquake that disables all other supplies. Separation and redundancy provisions are made where necessary in all power supply systems.

120. Although there are many differences in detail, the level of power supply reliability in Pickering and Bruce is equivalent and consistent with the design of other safety systems. Pickering, for example, has an extensive site electrical system for rapid connection of supplies to an accident unit from adjacent operating units. This is consistent with the type of ECIS installed in that station, which would require electrical power immediately after a LOCA.

### **(b) Instrument air**

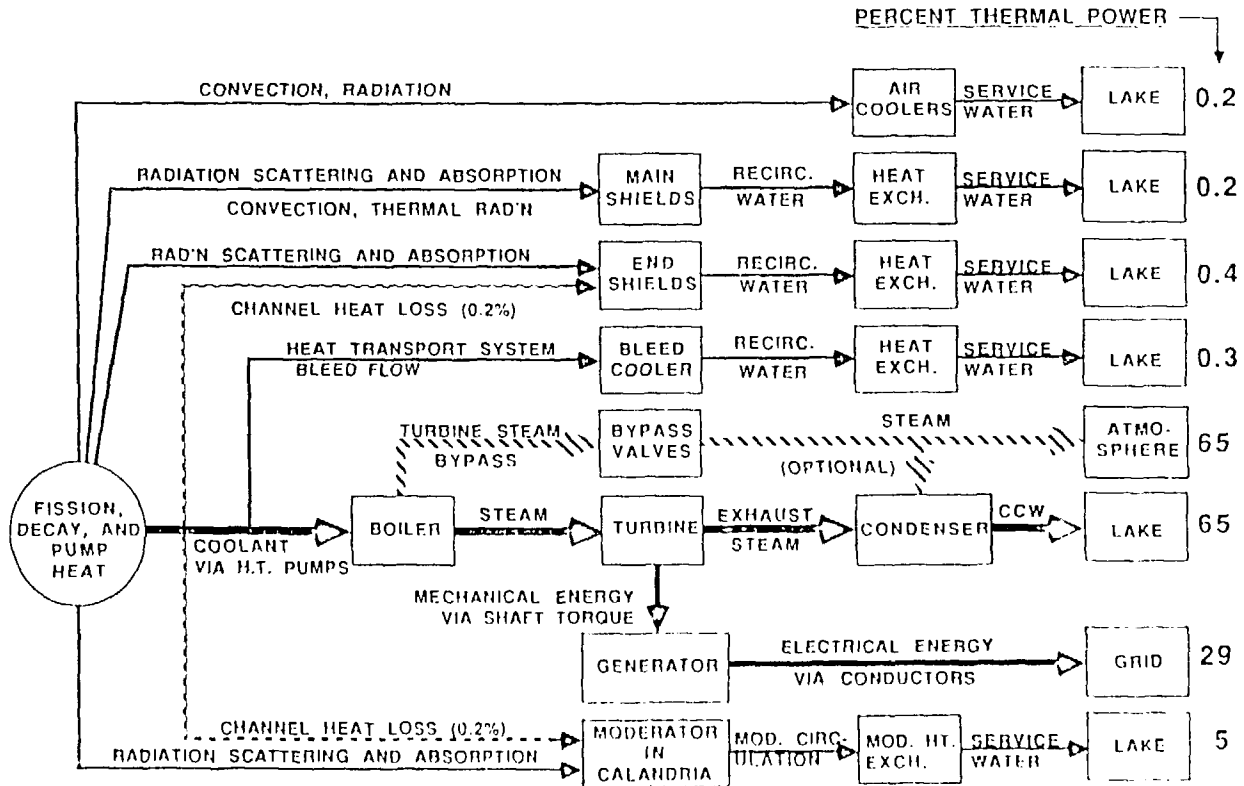
121. Compressors supply instrument air to the actuators of many valves in the station. This air is required for proper operation of normal process systems as well as safety-related systems after an accident. Compressors are driven by highly reliable electrical power supplies. Later plants have local air receivers that move valves to a pre-defined safe state position following loss of power. The systems installed in various plants are very similar, with a general trend toward more redundancy and diversity in later designs in response to more stringent regulatory requirements and operational safety concerns.

### **(c) Cooling water**

122. The ultimate heat sink for removal of almost all the post-accident residual heat is a local water body. A small amount of residual heat can be rejected in the form of steam from the secondary side of the steam generators. These heat flow pathways require electrical power to circulate cooling water through heat exchangers.

123. The heat flow pathways during normal operating conditions and under various post-accident conditions are shown in Figure I-47. The main heat flow pathway under many of these conditions is the steam generators. Relief and safety valves are connected into the secondary side piping so that steam can be released either to the main condenser or to the atmosphere, as shown in

Figure 1-47



Heat Flows During Normal Operation

Figure I-48. Water lost from the steam generators can be replaced from a variety of redundant sources. Figure I-49 illustrates a typical arrangement of steam generator water supplies.

124. One important feature of all CANDU reactors is a shut-down cooling system including heat exchangers and pumps that can remove decay heat from the primary heat transport system at full pressure. Heat can be removed from the containment space either through vault air coolers or via the emergency injection steam heat exchangers.

125. Many minor differences exist in the heat removal systems of Pickering B and Bruce B stations, but the overall reliability of heat removal is approximately the same in both cases. Pickering A has a slightly lower level of redundancy in water supply to the secondary side of the steam generators, because it does not have an emergency water system. The Bruce A station includes interconnections between the steam generator feedwater systems of the units, so that water can be transferred to an accident unit from a non-accident unit.

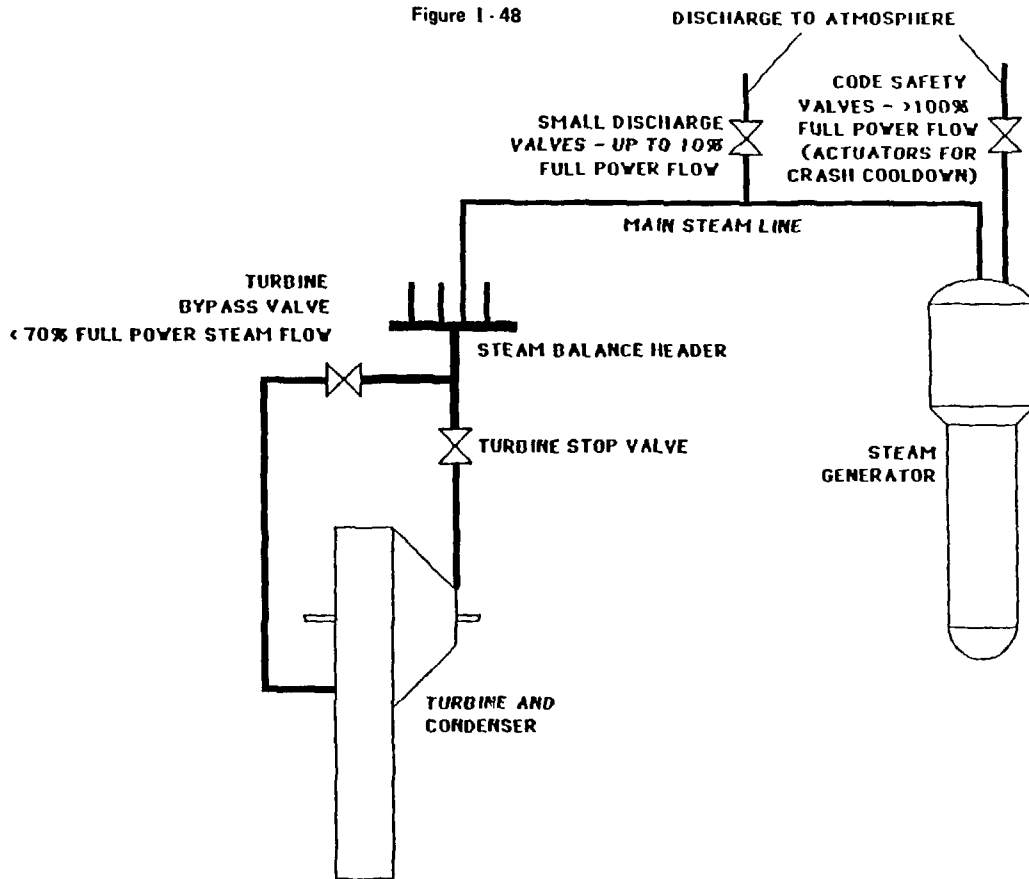
#### **(d) Monitoring**

126. Post-accident monitoring is required only for those parameters that need control in the long term. The major quantities to be monitored are the reactor shut-down state, containment temperature and pressure conditions, and various coolant flow rates.

#### **F. Comparison with Light-water Reactors (LWRs)**

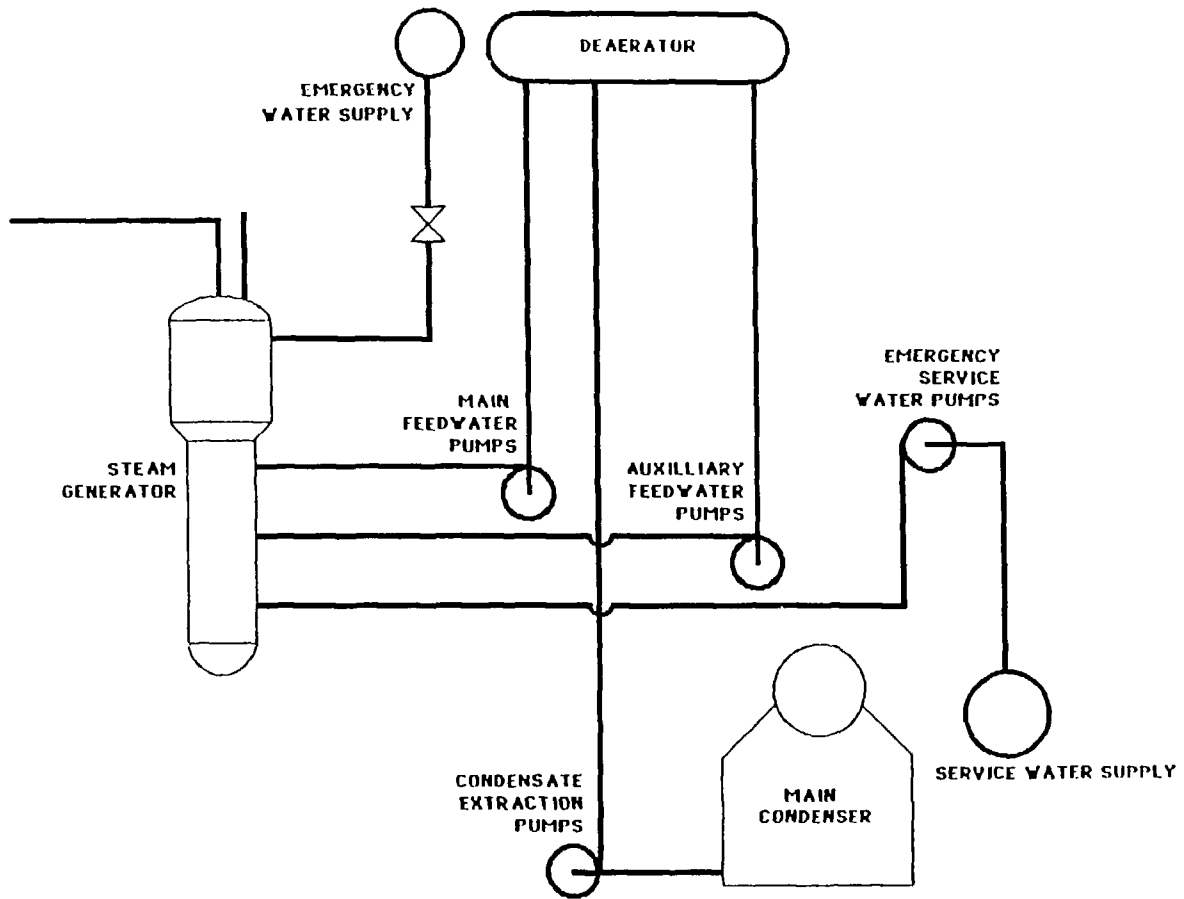
127. A majority of the approximately 400 commercial nuclear stations now operating in the world use a different reactor concept from the CANDU system. The difference arises mainly from the choice of ordinary water to slow down or moderate the fission neutrons, in place of the heavy water used for this

Figure I - 48



Steam Relief Pathways

Figure 1 - 49



Steam Generator Water Supply Alternatives

purpose in CANDU. Both advantages and disadvantages follow from this choice.

128. The number of neutrons absorbed in an ordinary-water moderator during operation is much larger than that in a heavy-water moderator. In order to make a chain reaction possible, the uranium fuel must be enriched, meaning that the number of uranium-235 atoms per kilogram must be increased above the level found in natural uranium. Enrichment of uranium is a difficult process, because the chemical properties of the two isotopes found in natural uranium, uranium-235 and uranium-238, are identical. A large and complex separation plant is necessary to produce enriched fuel. This was the main reason for selection of a heavy-water moderator in Canadian reactors. Separation of heavy water, which is present in ordinary water at a concentration of roughly one part in 7000, is also difficult. Heavy-water separation technology is, however, simpler than the technology of uranium enrichment.

129. The optimum economic design of a LWR leads to a smaller reactor core than is feasible in a heavy-water reactor (HWR), for the same heat output. This makes it feasible to build a large pressure vessel completely surrounding the reactor. A few HWRs have been built with pressure vessels, but their maximum electrical output is limited by the maximum practical size of the pressure vessel. As discussed, this limitation of reactor size, as well as the inability to construct large pressure vessels in Canada, led the CANDU planners to adopt the pressure tube design.

130. The simpler layout of a LWR and the high cost of heavy water lead to a potential for lower capital cost in the LWR than is possible for CANDU. This capital-cost advantage of the LWR is overcome, however, by the lower cost of CANDU fuel supply in most cost-accounting systems. Only when capital cost is weighted very heavily relative to operating cost does the LWR become more economical. With the Canadian financing arrangements, the lifetime cost of CANDU is significantly lower than that of a LWR, according to both Ontario Hydro calculations and data supplied by the North American Utility Cost Group,

which lists capital costs of all nuclear plants completed or under construction in North America.

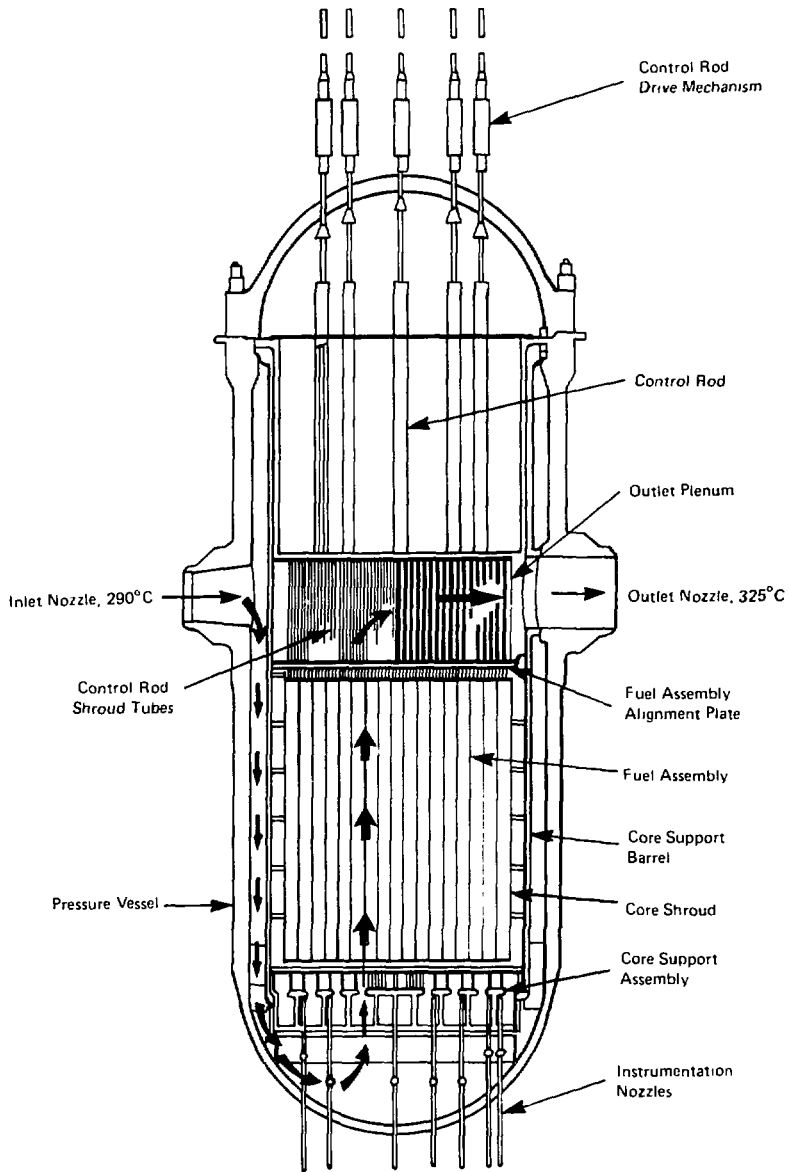
131. A typical layout diagram for a pressurised water reactor (PWR) is shown in Figure I-50(a), and the piping arrangement in a typical PWR station is shown in Figure I-50(b). The primary disadvantage of a pressure vessel is the difficulty of access to the reactor core; all control devices and detectors must penetrate the pressure boundary. This complicates the design of emergency shut-down systems, for example. In addition, refuelling of a pressure vessel reactor can be done only following reactor shut-down, cool-down, and opening of the top of the pressure vessel. CANDU reactors are refuelled at full power. This limitation brings with it another disadvantage. Because refuelling must be done with the reactor shut down, the intervals between refuelling must be made as long as possible, and a large amount of fresh fuel must be added at each step. The disadvantage is that a LWR that has just been refuelled carries a large excess of fissionable material, so that large amounts of neutron-absorbing materials must be added via control rods as well as those dissolved in the cooling water. Large control capability brings with it the possibility of accidental removal of large amounts of neutron absorber. The balancing advantage of LWRs is that they have a strong negative power coefficient: i.e., any increase in power immediately decreases the number of excess neutrons in each chain reaction cycle. CANDU has a much smaller control requirement, but also a smaller negative power coefficient.

132. The advantages of on-power fuelling in CANDU bring with them a disadvantage: the fuelling machines must open and close the high-pressure heat transport system boundary each time fuel is changed (two or three times per day). The potential for small loss of coolant failures is thereby increased. One such event occurred in the NPD reactor in 1963. Since then, the operating systems for fuelling machines have been greatly improved, and there have been no further accidents of this type.

133. There are two basic differences in design and operating philosophy between LWRs and CANDU reactors. First, CANDU reactors use fully automatic

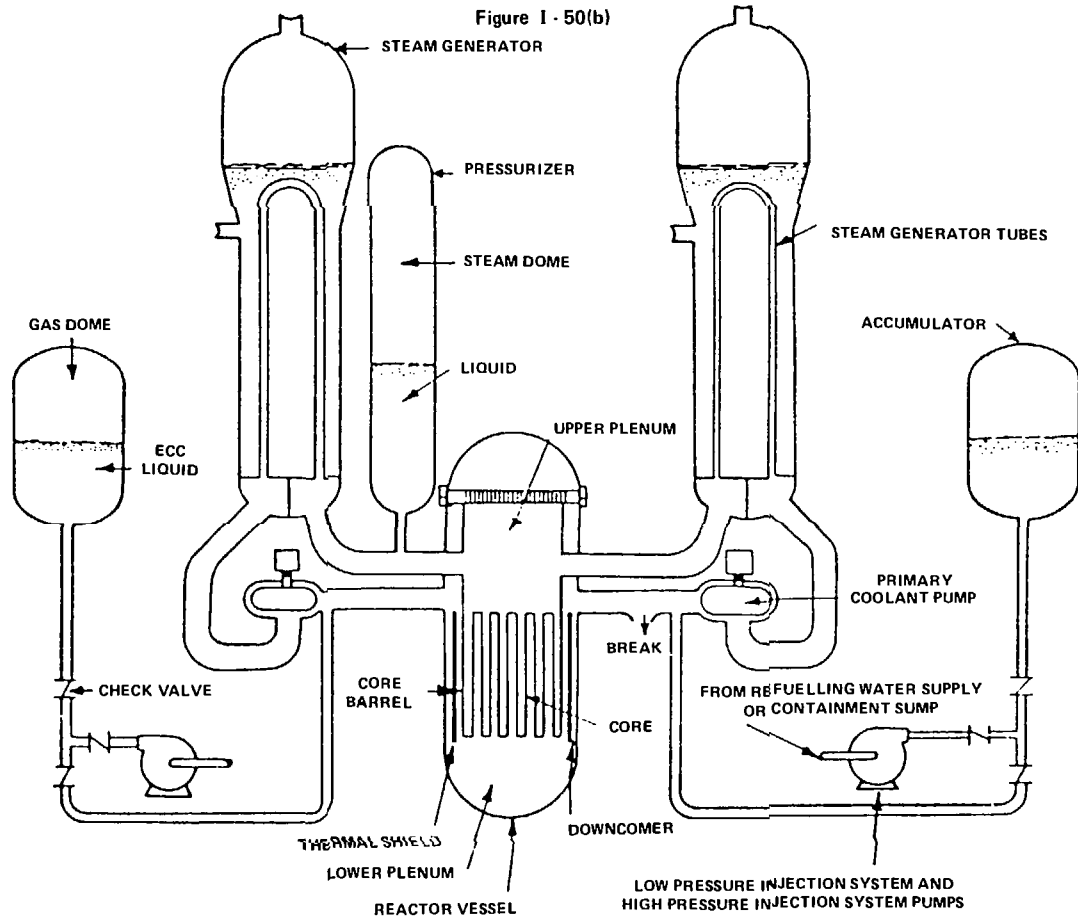


Figure I - 50(a)



Pressurised Water Reactor

Ontario Hydro



**Figure I - 50(b)**  
**MULTIPLE LOOP PWR SYSTEM DURING REFLOOD**  
**Piping Arrangement in Typical PWR Station**

Tong, L.S. and Weisman, J., "Thermal Analysis of Pressurized Water Reactors," Second Edition, American Nuclear Society, 1979

control during normal operation, so that the operators can concentrate on system testing and maintenance supervision. Designers of LWRs have not adopted this approach, although recent trends are moving in this direction. Second, the current Canadian regulatory approach demands complete physical and functional separation between process systems and special safety systems. LWRs are not designed to achieve complete separation.

134. The LWR has an inherent safety advantage relative to CANDU under LOCA conditions. (Both reactor types are similar in susceptibility to this type of accident, in that their coolants are under high pressure. If the coolant boundary fails, the coolant is lost very quickly.) In this type of accident, release of coolant from the heat transport system induces boiling in the reactor. In the CANDU system, this boiling leads to an increase in the neutron multiplication rate, whereas in LWRs, the multiplication rate decreases. The reactor power therefore would go down after a LOCA in a LWR: this is a change in the safe direction. The reason the power goes down in a LWR is obvious--the coolant is also the moderator, and removal of much of the moderator from a thermal reactor always results in a reactivity decrease.

135. In a CANDU reactor, failure of the primary coolant piping leads to loss of none of the moderator water. Removal of the heavy-water coolant immediately surrounding the fuel elements results in an increase in reactivity as the net result of several effects of opposite sign. The increasing power level in a CANDU reactor during a LOCA therefore requires immediate action of automatic shut-down systems to reduce the power level. Such action is, however, a simple process, because the shut-down systems need enter only the low-pressure moderator tank instead of a high-pressure vessel. One system is actuated by gravity and the other by compressed gas. The reactor geometry permits installation of independent, redundant, and diverse shut-down mechanisms.

136. Any pressure vessel design has a weakness that must be very carefully addressed by designers. If the large pressure vessel were to fail, the rapidly released energy of the high-pressure cooling water could burst any practical containment structure at the same time as it would greatly reduce cooling of

the reactor fuel. Because CANDU reactors contain only relatively small diameter, high-pressure piping, this severe potential accident scenario does not exist. Cooling-water piping can fail during operation in either type of reactor. Emergency cooling water would be added to the circuit automatically in either case. The basic difference is that, should this emergency coolant system not work for any reason, decay heat would still be removed from a CANDU reactor via the moderator system without any melting of uranium dioxide.

137. If this type of accident occurred in a LWR, large amounts of fuel would melt rapidly. Regaining control of fuel cooling would be very difficult, and, in extreme circumstances, such a molten mass might penetrate the floor of the containment building. (This produces the condition commonly known as the China Syndrome.) More probably, the molten core would solidify inside containment, and no radioactive materials would leak out. The most difficult aspect of this type of accident in the LWR is the possibility of fairly rapid steam formation when molten fuel mixes with water and the formation of non-condensable gases inside the containment if molten fuel interacts with dry concrete.

138. Given successful post-accident shut-down, there is almost no chance of melting large amounts of fuel in a CANDU reactor. This makes accident management much simpler and reduces the risk of containment failure in either the short term or the long term. Fission product retention is much better in fuel that has not melted than it is in fuel that has reached the molten state at some time during the accident.

**(a) Pressurised water reactors (PWRs)**

139. The PWR is similar to the CANDU reactor in many respects. Both use steam generators that isolate the reactor coolant from the turbine, and the reactor coolant is pressurised so that it does not boil significantly during normal operation. Other than the generic safety-related differences between pressure vessel and pressure tube reactors mentioned above, there are few important differences that are specific to the PWR. One of these is related to

the possible failure of a steam line on the secondary side of the steam generators. In the PWR design, such a failure would lead to rapid cooling of the heat transport water on the primary side, which would in turn increase the multiplication rate and necessitate rapid shut-down action. To prevent this occurrence, fast-acting steam isolation valves are installed in the steam lines. By comparison, a steam line failure in CANDU has no important effect on the reactor multiplication rate. However, in both systems the reactor must be shut down to establish a stable cooling regime before the steam generators are depleted, because the cooling rate will then decrease.

### **(b) Boiling water reactors (BWRs)**

140. In the BWR design, the steam formed in the reactor passes directly through the reactor outlet piping to the turbine. This layout gives an advantage in that the heat transport pressure can be lower for the same steam pressure at the inlet to the turbine. On the other hand, reactor water that may carry small amounts of radioactive materials from defective fuel is always present in the condenser. Experience shows that BWRs release higher levels of radioactive materials to the environment during normal operation than do either PWRs or CANDU reactors. In addition, any failure of steam lines outside the containment structure in a BWR results in a direct pathway between the reactor core and the environment. To reduce the probability of such an occurrence, BWRs must have very fast acting steam line isolation valves activated by low-pressure sensors. Further, if these valves close inadvertently, bypass of steam to the condenser must be initiated rapidly to prevent reduction of the steam fraction inside the reactor. Such a reduction increases the multiplication rate because it increases the amount of moderator water inside the reactor core.

### **(c) Comparison of overall safety characteristics**

141. It is not possible to make an overall judgement as to which design is "safer" in the broadest sense of the word. It is already clear that all of these designs can be operated with very low risk to public health. Each has inherent safety advantages and disadvantages relative to the others. In reality, choice

of basic design parameters for any of these concepts allows very little latitude in minimisation of disadvantages or maximisation of advantages without incurring severe economic penalties. In making these basic choices, one gets a package of both advantages and disadvantages. Engineered safety features must then be designed to counteract the disadvantages so as to arrive at a safe design. Finally, the most important evidence of safety is to be found in the operability of the plant and in the competence of the operating organisation.

## Appendix II

### A Review of the Design-related Aspects of the Safety of Ontario Hydro's Nuclear Generating Stations

by  
Peter M. Fraser

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## A. Introduction

1. In reviewing the safety of Ontario Hydro's nuclear generating stations, the Ontario Nuclear Safety Review (ONSR) examined three distinct aspects:

- those directly and indirectly related to the design;
- those related to the operation of the station; and
- those concerned with emergency measures that might be required in the event of a serious accident.

This appendix is concerned with the first of these.

2. In determining whether the design-related aspects of these stations pose an undue risk to the health and safety of the public, the ONSR took three steps. First, the most important areas were identified, because not all could be examined. Second, investigations into these areas were conducted with the aid of independent consultants. Third, submissions from numerous groups and interested members of the public were invited and considered.

3. The three most important areas of study identified were:

- hardware, specifically the pressure tubes, containment, ageing, performance of process and safety systems, and backfitting;
- analysis of accidents; and
- the probability of occurrence and the consequences of severe accidents.

4. For each general area of study, sets of criteria were devised to evaluate the safety-related aspects. These were listed as a set of questions, intended to aid the ONSR and its consultants:

(i) Hardware

- Are the design and construction satisfactory?
- Is the monitoring of the operating system adequate?

- Is adequate research support available to diagnose and solve problems?
- What improvements are advisable?
- (ii) Accident analysis
  - Is the approach sound?
  - Have all the sources of risk been identified?
  - Are the methods employed adequate?
  - Is the work well documented, well reviewed, and well validated by appropriate experiments?
  - Is there adequate ongoing research to support areas of poorer understanding?
  - What improvements are advisable?
- (iii) Probabilistic Risk Assessment
  - What are the chances of a serious accident?
  - Are the present methods used to evaluate this complete?
  - Is operating experience incorporated in the estimates?
  - Is the work well reviewed?
- (iv) Analysis and consequences of severe accidents
  - Is there sufficient work done in this area?
  - What are the risks and consequences of serious accidents?
  - Have all the relevant consequences been enumerated and considered?

5. In the following sections, each of the three important areas will be discussed in turn, summarising issues raised by the various consultants and intervenors.

## **B. Types of Reactor Accidents**

6. Although many malfunctions can occur, the events that can lead to the release of fission products from a nuclear generating station are actually few in number.

7. A basic safety principle is that the heat being generated within the fuel must be removed by the coolant. Because of the large amount of heat being generated within the fuel, the centre of each fuel element is very hot. If the ability to remove the heat is lost, the fuel can quite rapidly become molten; the fuel sheath, although much cooler than the fuel, can weaken or fracture; and the gaseous fission products trapped within the fuel can be released within the primary heat transport system. It is only if this system and the containment both fail that radioactive material can escape to the environment.

8. These failures could occur if (Figure II-1):

- the power being generated within the fuel increases out of control (loss of regulation leading to power excursion);
- the coolant to remove the heat from the fuel bundles is lost (loss of coolant accident [LOCA]);
- the heat removal pathway from the coolant to the environment is interrupted (loss of cooling); or
- the flow of cooling water over the fuel slows down or ceases (flow impairment).

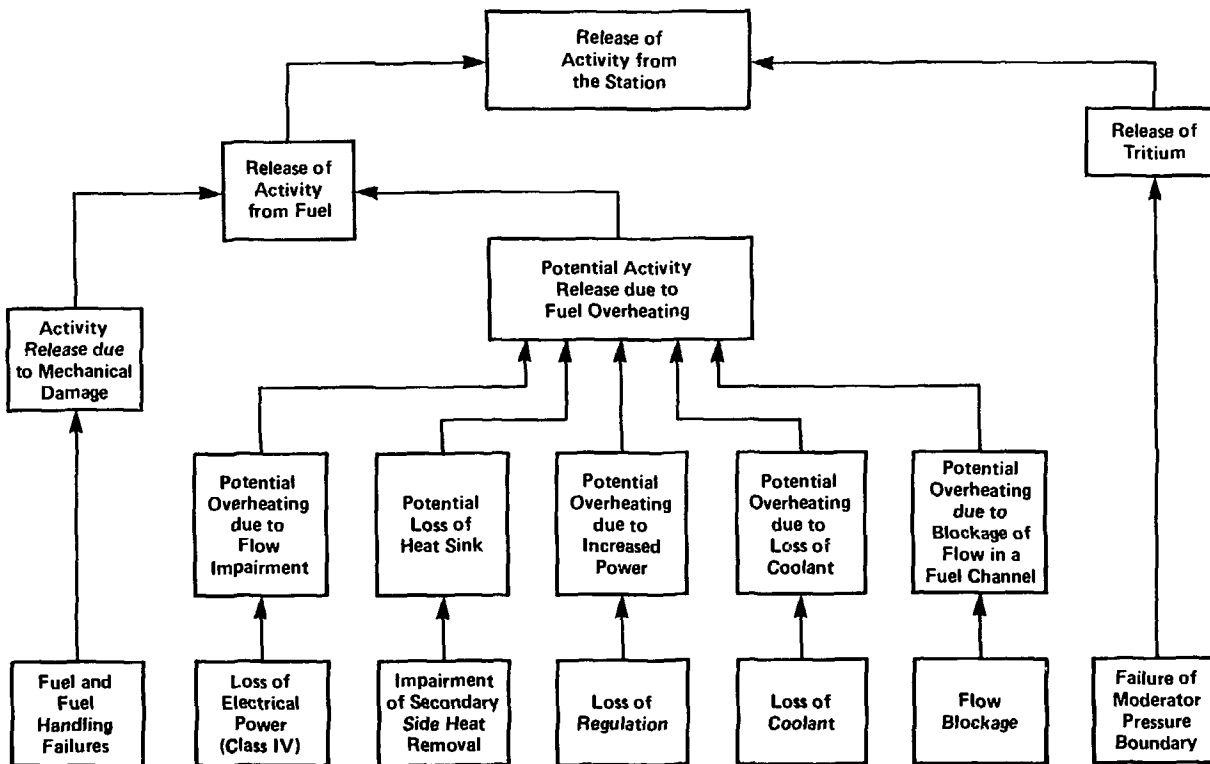
9. Another possible cause of a serious accident is the rupture of the heat transport system as a result of its pressure being raised beyond what it can withstand. Means of relieving pressure exist, but must be assessed for their adequacy under any conceivable conditions.

10. In addition, one needs to consider accidents with the fuel handling system, because this system also handles spent reactor fuel, which requires cooling.

11. In CANDU, because the moderator system contains a substantial inventory of radioactive tritium, leaks from the moderator system must also be considered.

12. In the subsections that follow, each of the above mechanisms is discussed as a potential cause of a severe accident.

Figure II - 1



Potential Mechanisms for Release of Activity

**(a) Power excursions**

13. In order to achieve constant power in a nuclear reactor, it is necessary that two-thirds of the neutrons generated in nuclear fission do not cause fission themselves. Many of the neutrons are removed by absorption by uranium-238 in the fuel and by structural materials: the final balance is achieved by adding or removing neutron-absorbing material from the reactor core.

14. A severe power excursion has been the key feature of several of the world's nuclear reactor accidents. In these accidents, for example at Chernobyl, at Chalk River (the NRX accident), and at Idaho Falls (the SL-1 accident), a large increase in reactor power to many times its normal value led to fuel melting or break-up.

15. The behaviour of various components and systems during such a rapid and highly energetic event is difficult to predict. For this reason alone, reactor safety design has focussed on the prevention of such events. The best way to prevent this particular event is to design the reactor so that such an excursion becomes physically impossible. This strong requirement cannot be met by the CANDU reactor, nor by most of the other commercial reactors operating in the world today. Thus, these reactors have possible failure mechanisms that can lead to severe power excursions. Active power regulation is required to keep power constant. Outright failures of the power control system could, if uncorrected by other means, lead to an uncontrolled increase in power. Furthermore, changes in other conditions (such as the coolant pressure) can lead to power excursions.

16. As a consequence, all such reactors must rely on the intervention of special shut-down systems, to detect and terminate the power excursion. This approach cannot eliminate the possibility that such events could occur, should these special systems fail or (as at Chernobyl) be circumvented. The goal is rather to make the events sufficiently improbable by building a reliable shut-down system that is capable of terminating any power excursion effectively.



17. The design of shut-down systems is determined by analysis to show what power excursion is possible given the particular design of reactor. In general, the designers have to consider two causes of power excursion:

- failures in the power control system itself, leading to an increase in reactivity and power, e.g., withdrawal of absorbing material; and
- failures in other systems that induce power excursions indirectly, e.g., a loss of coolant.

### 1. Regulating (power control) system failures

18. The power control (also called the regulating) system consists essentially of neutron-absorbing material that is moved into and out of the reactor core to control reactor power (see Appendix I). The rate at which absorbing material can be withdrawn from the reactor core is limited by the design of the hardware and interlocks, which function independently of the regulating system. The reactivity devices of CANDU reactors are located in the low-pressure moderator system, thereby reducing the chance of forced ejection of these rods from this system. The regulating system in CANDU reactors is computer-controlled, allowing more rapid and precise regulation of power. Seven failures of the regulating system have been reported at Pickering A, all attributed to errors in the computer software. Six of these occurred in the first 4 yr of operation, the seventh in 1982. In all cases, the loss of power control was terminated by the shut-down system. No fuel damage occurred as a consequence of these events.

### 2. Induced power excursions

19. In addition to the power excursions caused by failures of the power control system itself, power excursions may be caused by other upsets to the reactor system.

20. A characteristic of CANDU (and some other reactors) is the positive void reactivity coefficient. This means that boiling of water in the reactor core can

lead to a power increase, and thus to more boiling. The boiling of water in a large number of channels following a large pipe break would add sufficient reactivity to make the reactor prompt-critical within 2.2 s (Pickering A). This event leads to the most rapid rise in power. More detailed analysis of this postulated accident has been submitted to the ONSR and will be discussed in Section F. However, other scenarios can also lead to an increase in boiling in the core, and consequently a rise in power, if both regulatory and shut-down systems do not function. It is therefore essential that the shut-down system be able to detect any increase in power beyond the control of the regulatory system and promptly shut the reactor down.

21. There are other intrinsic design features of the CANDU reactor that affect its behaviour in power excursions. The use of natural uranium fuel means that the fuel temperature coefficient is less strongly negative than with enriched fuel. The power reactivity coefficient at nominal power is near zero or slightly positive, meaning that a reactivity increase is not naturally self-limiting.

22. It is primarily to counteract the positive void reactivity effect that the CANDU reactor has faster shut-down systems and more shut-down system capability than most other designs. The low-pressure moderator makes it relatively easy to employ two separate systems to shut down the reactor.

23. Since construction of the Pickering Nuclear Generating Station (NGS) A reactor, all CANDU reactors have been equipped with two shut-down systems, each of which is independently capable of shutting down the reactor under any conditions.

#### **(b) Loss of coolant**

24. Loss of water from the heat transport circuit can occur through a break in one of the coolant pipes; from one of the systems connected to the heat transport system; or through a valve on the system that could stick open. A small loss can be accommodated by normal make-up in the process system. A

large loss of cooling water could mean a degradation of heat removal, and the fuel could begin to heat up. The rest of this section is concerned exclusively with a large loss of coolant accident (LOCA).

25. The importance of the LOCA is that it represents the breaking of one of the barriers to the release of fission products (the heat transport system), and the possible lack of cooling of the fuel means that the first two barriers (the fuel and the fuel sheath) might also become ineffective. Hence, designers have focussed on providing another source of cooling water through an emergency coolant injection system (ECIS).

26. The LOCA in a CANDU reactor has several implications:

- (i) The positive void reactivity coefficient in CANDU reactors means that a power excursion will be initiated by a large pipe break. Shut-down by the special shut-down system(s) is necessary to avoid serious damage to the reactor.
- (ii) The two-loop design (Appendix I) at Pickering and Darlington means that half the fuel channels can be isolated from the effects of the accident.
- (iii) The relatively large length of piping increases the chance of a LOCA.
- (iv) The cool unpressurised moderator acts as a heat sink should emergency coolant injection not occur. In such an accident, pressure tubes in poorly cooled channels either balloon or sag to come into contact with the outer calandria tubes. The heat from the fuel channels is then transferred to the cool moderator.

27. To act as an effective heat sink, the initial temperature of the moderator must be low enough that heat removal from the tubes is effective, and moderator cooling must remain largely uninterrupted. To ensure that the first condition is met, the Atomic Energy Control Board (AECB) limits the moderator temperature. In order to meet this limit at Pickering B NGS, it is necessary to reduce power occasionally during the late summer months.

28. The key finding of the accident analysis of this event has been that loss of coolant does not cause failure of pressure tubes and surrounding calandria tubes, so a coolable fuel geometry is maintained. However, some recent analysis of pressure tube behaviour in such an accident suggests that it is possible that some of the tubes might rupture during the event. An active research programme is in place at Whiteshell to investigate the phenomenon. This concern is discussed further in Section D.

29. If, in addition to the other failures, the moderator cooling is unavailable or the fuel channels begin to fail, fuel melting is inevitable. The consequences of such an event are discussed further in Section F.

### (c) Loss of cooling

30. Another possible cause of fuel overheating is a failure to remove the heat from the fuel to somewhere in the outside environment where it can be dissipated naturally, e.g., a lake or river. This could be caused by a failure to cool the coolant. This type of accident is referred to as a loss of heat sink. For Ontario Hydro's CANDU stations, the normal path of heat removal is from the fuel to the coolant; from the coolant to the steam generator; and from the steam generator to cooling water drawn from a lake. If the circulation of the cooling water in the steam generator is cut off (loss of steam generator feedwater), if a pipe breaks in the steam--or feedwater--circuits, or if the cooling water from the lake is not circulated (loss of service water), then the heat sink is said to have been lost.

31. The CANDU reactor must be shut down in the event of this accident to avoid extensive reactor damage, because of the positive void reactivity coefficient. If this is accomplished, the following CANDU features are important in considering the sequence of events:

- an auxiliary pump to circulate steam generator feedwater has sufficient capacity to remove shut-down heat due to fission product decay;

- a sufficient amount of water is present in the steam generator to remove 30 min worth of shut-down heat, giving an operator the chance to make alternative heat sinks available;
- a shut-down cooling system exists for the primary heat transport system, capable of functioning at full pressure, thereby supplying an alternative cooling mode; and
- there are emergency water and power supplies, which are seismically qualified (Pickering B, Bruce B, and Darlington).

32. The most serious of the loss of heat sink scenarios are those that affect the greatest number of systems. The two most important are a loss of service water and a loss of electrical power, which leads to a loss of the pumping power required to circulate the service water. If back-up supplies are unavailable, the steam generator cooling water will boil away, and the primary coolant itself will begin to boil away. However, there are two aspects of CANDU design that significantly delay these failures. First, as with a LOCA, the moderator can act as a heat sink to remove heat. As no heat is being removed from the moderator, it will begin to boil away, eventually leading to the failure of fuel channels. Such a process would require many hours. Second, should the point be reached where the fuel channels fail, the reactor shield provides a back-up heat sink for the fuel inside the calandria. This would slow down the downward movement of any melted fuel for many hours more.

33. The consequences of such events are expected to be similar to those from the LOCA without a heat sink, alluded to above. However, events are expected to occur more slowly, because both the primary and secondary coolants must boil away before much fuel melting can occur. At that time, the decay heat is already very low and decreasing, so the temperature increase is quite slow. As a result, tens of hours are required to melt the fuel. Further discussion of this event is given in Section F.

34. A fourth class of accident that could lead to fuel melting is a loss of flow of the coolant over the fuel. This could occur as a result of the loss of

power to one of the primary coolant pumps. In CANDU, the loss of flow must be accompanied by a reactor shut-down to prevent overpressurisation of the primary system. Natural circulation of the coolant is effective in the short term for removing the heat from the fuel following shut-down.

#### **(d) Flow blockage**

35. A blockage of flow in a fuel channel could occur if a chunk of material inside the heat transport system plugs up the feeder piping. However caused, a blockage of channel flow would mean that cooling of the fuel in that channel would be diminished, and the fuel could begin to heat up. If the blockage were not nearly complete, the flow of coolant would still be sufficient to prevent fuel melting, although the fuel could be damaged. If, however, the blockage of flow were complete or nearly complete, the fuel in the channel would begin to melt. The molten fuel could fall onto the pressure tube, causing it to burst. The steam and hydrogen mixture escaping from the broken pressure tube would break the surrounding calandria tube. As a result, the contents of the channel, including the fuel, could be injected into the moderator system.

36. The possible ejection of the fuel and coolant into the moderator raises two important issues. The first is the ability of either shut-down system to detect the upset conditions and shut down the reactor, and keep it shut down, when one considers the damage that could be done by the broken channel and the ejected (and possibly molten) fuel. Ontario Hydro has shown that no failures of adjacent channels would occur as a result of this event. The second issue is the damage that could be caused by molten fuel on guide tubes of the shut-off rods and on other fuel channels.

#### **(e) Fuel handling accidents and end-fitting failure**

37. Those reactors that use on-power refuelling, including CANDU, have a class of accidents associated with fuel handling. For example, the end of the channel can break off (an end-fitting failure), which may lead to the ejection of the fuel onto the floor of the reactor vault. The fuel would be damaged by

its impact and would start to release its fission products. Although the total amount of the radiation thereby released is only a small fraction of the total amount in the reactor core, there is a possibility that radioactive fission products may be released from containment prior to containment isolation.

38. Similarly, an accident with the fuel handling machine while transporting spent fuel can lead to fission product releases from the used fuel contained in the fuelling machine.

39. Irradiated fuel is stored in the irradiated fuel bay. A loss of fuel by coolant or cooling could lead to the heat-up of irradiated fuel bundles.

#### (f) Moderator system piping failures

40. A leak from the moderator system could lead to the release of the moderator water containing relatively high concentrations of tritium. There have been several incidents of leakage of tritium from moderator heat exchangers into cooling water from the lake. Although the amounts of tritium so released have been well below derived release limits, they do pose a problem.

41. In addition, there has been a concern lately that a relatively rapid loss of moderator could release deuterium and gases produced by radiolysis of the moderator to the gas space in the calandria, where they might ignite. The source of ignition could be the cobalt (or steel) control rods (called adjuster rods), which would begin to heat up by radioactive decay once the moderator that cools them was removed. Such a detonation may have occurred during the National Research Experimental (NRX) reactor accident in 1952 (Hurst 1953). The concern is that such an event could damage the calandria itself.

#### (g) Summary

42. In this section, some of the more serious things that can go wrong with a reactor have been summarised. What follows is how these accidents are assessed by Ontario Hydro for the AECB. The next section documents the

accident analysis that is carried out, compares the results of accident analysis historically, and lists defects that have been pointed out by consultants and intervenors. The work done on probabilistic safety evaluations in reducing the risk from Ontario Hydro's nuclear generating stations is discussed in Section E. Finally, Section F reviews the consequences of severe accidents. Recent assessments by Ontario Hydro, the AECB, and Atomic Energy of Canada Limited (AECL) are included.

## **C. Hardware**

### **(a) Introduction**

43. This section reviews various investigations undertaken by our consultants and the submissions of intervenors on the reactor hardware. Recommendations arising from these reviews are presented at the end of each subsection.

### **(b) Pressure tubes**

44. The integrity of the heat transport system is an important barrier to the release of fission products from the fuel to the environment. Pressure tubes have proven to be the most fragile part of the heat transport system. Twenty-three pressure tubes have leaked at Ontario Hydro reactors. Two pressure tube ruptures have occurred: one at Pickering Unit 2 (channel G 16) in 1983, the other at Bruce Unit 2 (channel N 06) in 1986.

45. In the Pickering incident, the action of the operators ensured that neither automatic shut-down nor emergency coolant injection were required. The leak, although substantial, was small enough that the reactor could be shut down slowly using the regulating system, and the coolant could be made without using the ECIS.

46. In the Bruce Unit 2 accident, an operator was attempting to locate a channel with a small leak. The operator increased the coolant pressure to its



normal value while the reactor was shut down and cooled down. This action, contrary to approved procedure, caused the leaking tube and its surrounding calandria tube to rupture. Most of the fuel bundles remained in the channel; however, some of the fuel elements broke off the bundle, and pieces were swept into the moderator system. Most of these pieces ended up at the bottom of the calandria vessel. One, however, got stuck at a valve in the moderator system: its removal was made difficult by the associated serious radiation hazard. None of the adjacent fuel channels was damaged.

47. Although neither pressure tube failure was hazardous to the public, the clean-up of radioactive water and fuel did involve a significant radiological hazard to the staff. Furthermore, there was a concern that a single pressure tube failure could cause the failure of other channels as well.

48. The pressure tube failure at Pickering led to the discovery that numerous tubes had degraded in the two reactors with Zircaloy-2 tubes: Pickering units 1 and 2. As a result, the two reactors had their pressure tubes replaced. This process took over 4 yr and cost \$414 million. Similar degradation is occurring at the reactors with pressure tubes composed of zirconium - 2.5% niobium, although it appears to be occurring at a slower rate. Ontario Hydro expects that the life of these tubes will be 25 yr and has announced its intentions to retube all operating reactors once during their life span.

49. The cause of these pressure tube ruptures and the need for retubing are related to the metallurgy of CANDU pressure tubes. For this reason, the ONSR employed Dr. Derek Northwood to investigate the degree of understanding possessed by Ontario Hydro on the metallurgy of the pressure tubes.

50. The importance of metallurgical processes relevant to pressure tube integrity became apparent soon after Dr. Northwood was contacted to act as consultant. The ONSR learned that a pressure tube pulled from Pickering Unit 3 had a concentration of deuterium four times higher than had been found in previous tubes. The measured concentration was close to the level at which the formation of hydride blisters became possible. The ONSR requested Dr.

Northwood to investigate the implication of this result for the timing of the planned retubing of all the reactor units.

51. In addition, many of the safety-related concerns with the CANDU design are related to the consequences of pressure tube failures. The ONSR asked Dr. David Burns to review Ontario Hydro's programme to investigate the mechanisms and consequences of pressure tube ruptures in CANDU reactors.

52. The findings of the two consultants are summarised below. In addition, the recommendations in the submission of Mr. N. Teekman (1987) on the time of retubing and the sources of hydrogen ions in pressure tubes are considered.

### 1. Dr. Northwood's report

53. Dr. Northwood's (1987) task was to review the status of the Canadian investigations into the metallurgy of the pressure tubes in Ontario Hydro's nuclear reactors. The following seven aspects were examined:

- the metallurgy of zirconium-niobium;
- ageing-related processes;
- status of research and technical support programmes;
- relevant international experience with this alloy;
- non-destructive examination techniques and in-service inspection methods;
- the time at which these reactors should be retubed; and
- the higher deuterium levels in a pressure tube in Pickering Unit 3 (P3L09).

54. A summary of his findings is presented here. Recommendations are given at the end of this subsection.

### Metallurgy

55. The understanding of the metallurgical processes of the CANDU pressure tubes is adequate. The one possible exception is the effect of iron impurities on the rate of deuterium ingress.

### Ageing-related processes

56. Both thermal and irradiation-induced processes appear to be well understood. At present, the major concern is the rate and cause of deuterium pick-up on pressure tubes. Although understanding of the processes involved in deuterium pick-up has improved, the results from P3L09 create serious doubts about the present models. It remains to be seen whether this tube is anomalous. Dr. Northwood noted that coolant boiling might be expected to have an effect on deuterium pick-up rate.

### Research and technical support

57. Research and technical support programmes are adequate. In the past, *programmes appear to have been more reactive than proactive*, although this situation has improved.

### International experience

58. Several other countries have had experience with zirconium-niobium alloys in reactors. The experience of these countries has been quite limited, with the notable exception of the Soviet Union. The Canadian expertise in zirconium-niobium alloys lies principally within AECL, Ontario Hydro, and various authorities at Canadian universities. Information exchange with the Soviet Union could prove beneficial, once the necessary agreements are in place.

## Non-destructive examination and in-service inspection methods

59. Non-destructive examination techniques and in-service inspection methods have greatly improved in the past 5 yr. The removal of pressure tubes for examination has become a routine process. Ontario Hydro's in-service inspection programme for 1987-88 will be inspecting channels to investigate the effect of contact between pressure tubes and calandria tubes and to detect manufacturing and commissioning-induced flaws. An ultrasonic technique for detecting cracks in individual tubes has been developed. A new method for measuring deuterium levels in pressure tubes--the pressure tube scraping technique--has also been developed very recently. This method appears to be both a safe and reliable means of assessing deuterium levels in pressure tubes.

## Retubing

60. Retubing is currently scheduled for all operating reactors after 25 yr of service. The P3L09 tube, if it proves to be typical rather than exceptional, could upset these plans. Criteria are being developed by the AECB and by Ontario Hydro to determine when a reactor ought to be retubed. Dr. Northwood points out such criteria will be difficult to formulate in light of the complexity of the real reactor situation. He is also concerned that present models of pressure tube behaviour are based on an assumption of average material structure. He argues that the differences between individual pressure tubes may be important, and it may well prove that the exceptional pressure tubes are the ones that can exhibit unexpected behaviour.

## High deuterium levels in P3L09

61. Although pressure tube P3L09 at Pickering Unit 3 remains an outstanding exception, there is a substantial possibility that this tube is typical: a rapid increase in the deuterium concentration in zirconium-niobium pressure tubes is occurring, beginning at about 15 yr of age. A similar acceleration of deuterium concentration was observed in the Zircaloy-2 pressure tubes at the Nuclear Power Demonstration (NPD) reactor. The results from this tube imply

that more frequent sampling of deuterium levels in pressure tubes is required. Ontario Hydro has increased its sampling rate using the new pressure tube scraping technique.

## 2. Dr. Burns' report

62. Dr. Burns (1987) was asked to review the status of research and understanding of the causes and consequences of the failures of pressure tubes and calandria tubes in Ontario Hydro's CANDU reactors. The review consisted of four phases:

- causes of pressure tube failures;
- calandria tube integrity following a pressure tube failure;
- consequences of a fuel channel rupture; and
- fuel channel integrity during a LOCA.

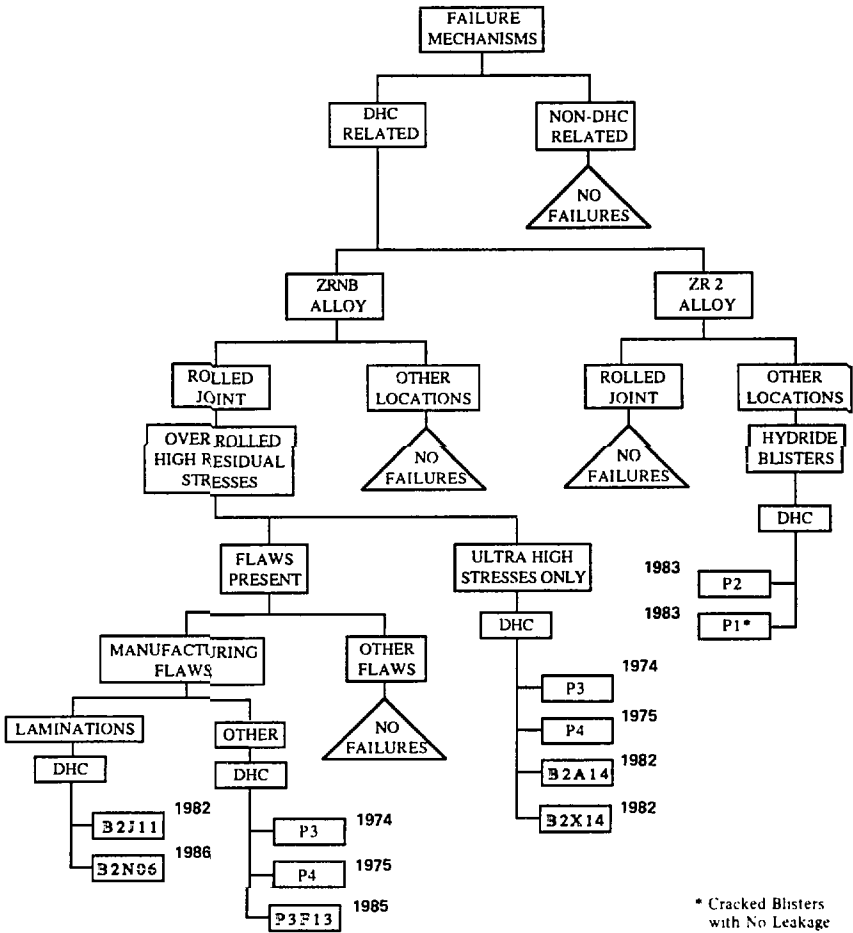
63. The results of Dr. Burns' review are summarised here.

### Causes of pressure tube failures

64. As shown in Figure II-2, the pressure tube failures at Ontario Hydro's nuclear generating stations have all originated from delayed hydride cracking. The delayed hydride cracking occurred as a result of one of two mechanisms: stresses at rolled joints, or hydride blistering. Although some of these stresses have been caused by manufacturing flaws, a number of tubes were stressed as a consequence of incorrect installation (overrolling the rolled joint). In all cases, the tubes leaked initially, and these leaks were detected.

65. In one case, at Bruce Unit 2 in 1986, the pressure tube subsequently ruptured. This rupture was caused by an operator's actions, contrary to approved procedure, to increase the coolant pressure to full pressure with the coolant cold.

Figure II - 2



Pressure Tube Failures in Ontario Hydro Power Reactors

Ontario Hydro

66. The pressure tubes in Pickering units 3 and 4 and Bruce Unit 2 have rolled joints that were overrolled. Some rolled joint cracks may still develop in these units. There have been significant improvements in the leak detection systems since the rolled joint cracks leaked in Bruce Unit 2 in 1982. Research programmes are in place to develop a better understanding of crack shape development, leakage rates, delayed hydride cracking (DHC) crack velocities, and critical crack length. There is some evidence that irradiation increases DHC crack velocities and reduces critical crack length. This reduces the time available for detecting a leak from a rolled joint crack.

67. The pressure tubes in Pickering units 3-4 and Bruce units 1-4 have garter springs that are known to be out of their design positions. Indeed, in at least one channel, there do not appear to be any garter springs at all (Knutsen 1987). Irradiation-induced deformation of the pressure and calandria tubes and misplaced garter springs together may lead to early contact between the tubes. This contact and the gradual pick-up of deuterium will propagate into the pressure tube. This crack may propagate by DHC along the tube and lead to a rupture of the pressure tube. This is what occurred at Pickering Unit 2 in 1983.

68. Inspection techniques are now well established for locating loose garter springs in reactors. Because garter spring movement from the design position may lead to contact between pressure and calandria tubes, which can cause blistering, there is a need to check garter spring positions periodically for all reactors, including those in which springs have been repositioned and those with tight garter springs.

#### Calandria tube integrity following a pressure tube failure

69. Perhaps the key difference between the ruptures at Pickering and Bruce was the fact that at Pickering the calandria tube remained intact, but at Bruce it ruptured along its seam. The reason for this appears to be that at Pickering the coolant was hot (the reactor was almost at full reactor power), whereas at Bruce it was cold, as the reactor had been shut down and cooled down to find

the leaking channel. When the Pickering pressure tube broke, it released water at more than 260°C through its crack. The water, in passing through the crack, changed into steam, so that the force of the jet was cushioned. By contrast, the coolant at Bruce was cold and hit the calandria tube as a water jet, with a much greater force than that of a steam jet. Also, the Bruce pressure and calandria tubes, being cold, were much less ductile than the hot Pickering ones.

70. This experience suggests that at ordinary full-power operation, the failure of a pressure tube may not cause its outer calandria tube to fail. There has been an attempt to confirm this in experiments at Canadian Westinghouse Incorporated, funded by the CANDU owners. Preliminary results appear to confirm this (Hadaller and Muzumdar 1987). However, Dr. Burns notes that there is a very small database for the mechanical properties of irradiated calandria tubes.

#### Consequences of a fuel channel rupture

71. Should a fuel channel rupture, the calandria vessel, adjacent fuel channels, or shut-down system components could be damaged by any of the following mechanisms:

- the pressure rise in the calandria from the injection of hot pressurised coolant;
- the force of the coolant jet;
- mechanical damage to adjacent channels caused by the moving broken fuel channel; or
- the ejection of pieces of fuel or fuel bundles into the calandria, which could damage components within the calandria.

The potential damage of each of these mechanisms to each component will now be considered.

72. Analysis and experiments indicate that pressure amplitudes and impact loads of a fuel channel failure are too small to damage the calandria vessel.



73. Analyses indicate that rupture of a fuel channel will not lead to failure of adjacent fuel channels. The steam bubble created will cause some pressure tubes to collapse onto their calandria tubes. The damage caused by pipe whip on irradiated pressure tubes has been shown to be small. A concern that has not been addressed is the effect of pipe whip damage on an adjacent fuel channel with an irradiated and blistered pressure tube. Although unlikely, it may be a relevant factor when setting limits for permitted hydrogen levels in pressure tubes. Based upon experimental evidence, the effect of the impact of fuel on adjacent fuel channels is not likely to be significant; the calandria tube acts as an effective shield, even if it has collapsed.

74. Damage to the mechanisms of the shut-down systems will not be severe enough to prevent effective shut-down. Some shut-off rod guide tubes may be damaged, but sufficient numbers of rods remain to provide adequate shut-down margin. Damage to poison injection nozzles would not prevent the poison injection system from operating.

#### Fuel channel integrity during a LOCA

75. Should a large pipe break occur (a LOCA), coolant in some of the fuel channels will be lost, and the pressure tube will begin to strain. However, the strain around the tube will be uneven owing to some temperature variations. If large enough, this may lead to the rupture of the straining pressure tube before it makes contact with the outer calandria tube. A model has been developed by Ontario Hydro, which predicts that pressure tube rupture would not occur during a large-break LOCA (Locke et al. 1987). Ontario Hydro and AECL are carrying out an experimental programme at Whiteshell Nuclear Research Establishment (WNRE) in an attempt to validate the model (So et al. 1987).

### 3. N. Teekman's submission

76. In his submission to the ONSR, Mr. N. Teekman (1987) argues that the nuclear sources of the hydrogen ions could be a significant contributor to the

hydriding of pressure tubes. Dr. Northwood, in his report, commented that he believed that the nuclear sources of hydrogen ions were not significant.

77. Mr. Teekman also suggests that serious consideration be given to retubing Pickering Unit 3 after 15 yr of service because of the smaller number of garter spring spacers and because "fuel channel tubes will have to be replaced after 25 years of service regardless, so early retubing will not affect total unit energy cost of the station" (Teekman 1987:37).

#### **4. Retubing of Pickering units 3 and 4**

78. Ontario Hydro has recently informed the ONSR that it is considering retubing Pickering units 3 and 4 within the next 2 or 3 yr. Two reasons can be given for considering the earlier date. First, as has been mentioned previously, the trend toward higher deuterium pick-up rates that has been observed and the contact between pressure tubes and calandria tubes suggest that the probability of a pressure tube failure has increased significantly.

79. Second, at Pickering units 3 and 4, the axial growth of the pressure tube will require adjustments to be made to the fuel channels in late 1989 in Unit 3 and in late 1990 in Unit 4. As a result of these adjustments, the gap at one end of the fuel channels will be larger than before. Should a pressure tube failure occur, the flow might well be large enough to require the automatic safety systems to be activated, unlike the break at Pickering Unit 2.

80. The increased probability of a pressure tube failure and the increased likelihood that such a failure would require the special safety systems to operate would reduce the defence-in-depth built into the station. As a result, Ontario Hydro is considering replacing the pressure tubes in these two reactor units before it becomes necessary to make the fuel channel adjustments.

## 5. Conclusions and recommendations

81. Pressure tube failures in the past have not proved to be a hazard to public safety. The knowledge of pressure tube metallurgy has improved substantially over the past few years. Furthermore, the methods of leak detection at all reactors have been or soon will be upgraded.

82. However, the high deuterium reading from P3L09 shows that all the phenomena relevant to ageing of pressure tubes are not yet well understood. Therefore, the following measures to prevent pressure tube failures are recommended:

- (i) There should be an increased frequency of in-service inspection to measure the deuterium concentration and the rate of increase of deuterium concentration in pressure tubes in order to have a statistically representative sample. This is especially desirable for Pickering units 3 and 4, the reactors with the oldest pressure tubes.
- (ii) More measurements of these quantities should be made in fuel channels in which coolant boiling occurs (e.g., in Bruce).
- (iii) Garter spring positions should be checked periodically in all reactors to satisfy the claim that they will not move.
- (iv) Retirement indices or "fitness for purpose" codes for pressure tubes should be devised quickly enough to be useful for determining when retubing at Pickering units 3 and 4 is required.
- (v) Research effort should be increased to determine the cause of the increase in deuterium pick-up rate observed in P3L09.
- (vi) Ontario Hydro appears to have quite a good grasp of the consequences of pressure tube failure. The initial results of experimental studies appear to confirm that pressure tube failures will not cause a calandria tube failure while the reactor is under normal operating conditions. To confirm this conclusion, more data on irradiated calandria tubes should be sought.
- (vii) A very important area that is currently being investigated is the possibility of fuel channel failure during a LOCA. Experiments are

under way to confirm predictions that fuel channel failures are not expected during such an event. However, it is not clear whether the pressure tube behaviour is similar if the tube is irradiated and has a high deuterium content. More research effort should be undertaken to determine whether a high deuterium content in pressure tubes would adversely affect pressure tube integrity during a LOCA.

- (viii) In light of the importance of the coolable fuel geometry to the confidence in the predictions made by accident analysis, the consequences of fuel channel failure on adjacent channels require further investigation. It is our understanding that the AECSB has requested that such tests be done. Such studies should include the effects of significant levels of deuterium in the pressure tubes.
- (ix) The ONSR has been informed that Ontario Hydro is considering replacing the pressure tubes at Pickering units 3 and 4 at an earlier date than had previously been considered. By the fall of 1989 in Unit 3, and by the fall of 1990 in Unit 4, adjustments will need to be made to fuel channels in these reactors. These adjustments, combined with the increase in deuterium absorption and the contact between the pressure tube and calandria tube, diminish the standard of protection for public health and safety. Retubing before these adjustments become necessary is the most sensible and safest solution.

### (c) Containment

83. Containment is the ultimate barrier to prevent or reduce the release of fission products from the fuel elements to the environment. The presence of a robust containment structure at Three Mile Island (TMI) limited releases to very small fractions of the total radioactive material released from the fuel. The lack of a robust containment at Chernobyl is the reason, according to some

authors, that releases to the environment from the Chernobyl accident were so high (Donahue et al. 1987\*).

84. The ONSR decided to consider two aspects of the containment systems at Ontario Hydro reactors. First, considering that Ontario Hydro multi-unit containment is an unusual containment design, it was important to know whether these containments were designed, built, and operated to the highest international standards. The ONSR hired the consultants J.D. Stevenson and Associates of Cleveland, Ohio, to undertake this review. Second, another remarkable feature of the containment used at Ontario Hydro nuclear generating stations is the plan, in the event of a serious accident, to gradually exhaust the containment air through filters. This venting would begin within a few hours (if there were an impairment of containment) to a few days after the accident began. The ONSR hired Dr. R. Jervis and Dr. G. Evans of the Centre for Nuclear Engineering at the University of Toronto to assess whether more might be done to further reduce the releases of radioactive material from the deliberate venting. The mandates and reports of these two groups of consultants are summarised below.

**1. The design, construction, and operating performance of the containments at Ontario Hydro's nuclear generating stations**

85. The ONSR requested J.D. Stevenson and Associates to provide an overall safety assessment of the containment structures, systems, and components used in Ontario Hydro's nuclear power plants. The consultants were asked to compare Ontario Hydro design and operating procedures with accepted international practices. To accomplish this, the consultants undertook a review of the criteria, procedures, and calculations used to design the containments, both the civil (concrete) structures and the metallic components. They reviewed containment construction experience, including the non-conformance reports and design changes filed during construction, to see whether the containment, as

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\*In a preface to this article in Nuclear Safety, the journal editor noted that "No general consensus yet exists . . . on the issue of whether a hypothetical Western-style containment could have contained the releases from the Chernobyl accident."

built, meets its design specifications. Containment tests were reviewed to determine the containment performance. Significant Event Reports (SERs) relating to containment operation were reviewed in an attempt to uncover operating problems that might reduce containment effectiveness. The consultants also carried on discussions with the AECB concerning its role in reviewing the containment design.

86. A summary of the findings of Stevenson and Associates (1987) is given here.

### Design features

87. The consultants noted two features particularly unique to the containment designs of Ontario Hydro reactors:

- the unique and positive design feature of the use of a vacuum building as part of the negative pressure containment; and
- the containment building design at Bruce and Darlington, which does not enclose the secondary side of the steam generators or the pump motors. However, the opinion of the consultants was that this was not necessarily a drawback of this containment design: similar limitations existed in all pressurised water reactor (PWR) containments.

### Design loads

88. The consultants considered both normal loads and loads from external or internal events. A summary of their findings for the different stations is presented in Table II-1. The following points were most noteworthy:

- (i) Normal design loads: Ontario Hydro containments have much lower design pressures than other containment designs owing to the vacuum building design. The action of the vacuum building in an accident means that Ontario Hydro containments experience less

Table II-1

## Containment Design Considerations for Ontario Hydro Nuclear Generating Stations

<u>Design load</u>	<u>Pickering A</u> <u>1964-71</u>	<u>Bruce A</u> <u>1969-76</u>	<u>Pickering B</u> <u>1974-82</u>	<u>Bruce B</u> <u>1975-85</u>	<u>Darlington</u> <u>1977-88</u>
Earthquake	1	1	2	2	3
Wind	1	1	1	1	1
External flood	1	1	1	1	1
Pipe break					
Jet impingement	0	0	x	x	x
Pipe whip			x	x	x
Compartment pressurisation					
<i>flooding</i>	0	0	5	5	5
Hydrogen deflagration	0	0	5	5	5
Turbine missile	0	0	0	x	x
Internal missile	7	7	7	7	7
Tornado	0	0	0	0	x
External explosion	4	4	4	4	x
Aircraft crash					
Small < 5000 kg	0	0	0	0	0
Large > 5000 kg	6	6	6	6	6
Fire	1	1	1	1	1

x Normally considered in design.

0 Not considered in design.

1 Design requirements of the National Building Code of Canada.

2 Acceleration levels as determined by Canadian seismologists and analytical methods as developed by US NRC.

3 Acceleration levels as determined by Canadian design standards CSA N289 and CSA N287.

4 Considered on a site-by-site basis as a function of potential sources of explosion.

5 Not originally considered as a design basis, but subsequent evaluation has shown structure capable of withstanding postulated hydrogen deflagration.

6 Determined to be of very low probability <  $10^{-7}$ /yr.

7 Not considered in CANDU design; no obvious source of such missiles.

Source: Stevenson et al. 1987.

stress than other designs. As a result, concerns about containment integrity, which arose subsequent to construction (e.g., hydrogen burn), are accommodated by the present design.

- (ii) **Earthquake loads:** Ontario Hydro containments are designed to accommodate a much smaller earthquake than is assumed in US designs across Lake Ontario. Furthermore, explicit seismic design requirements for mechanical systems were not done at Pickering A NGS or at Bruce A NGS. The consultants maintain that the Ontario sites are of low seismicity, and explicit design for earthquake loads on piping and structures is unnecessary.
- (iii) **Tornado loads:** Unlike US practice, neither Pickering nor Bruce has considered tornadoes as part of its design. The consultants consider the national building code loads to be sufficient protection.
- (iv) **External floods:** Although the design is again less stringent than US practice, the consultants conclude that the use of historical flood data to determine the possible flooding of the station from one of the Great Lakes is appropriate.
- (v) **Pipe breaks:** Ontario Hydro plants appear to have fewer pipe whip restraints than current US design. However, the current provisions appear to be adequate.
- (vi) **Turbine missiles:** Concern is expressed over the lack of a turbine missile shield to protect the steam drum at Bruce A, even though such a provision was made at Bruce B. Ontario Hydro has informed ONSR that the frequency of such events at Bruce B was calculated to be very low ( $<10^{-10}/\text{yr}$ ). However, as supporting basis, they reference a calculation of a turbine missile striking a primary pump motor rather than a steam drum.
- (vii) **Missiles generated within containment:** Analysis of the consequences of a missile generated within containment by a valve bonnet or other device is not considered in Ontario Hydro design. The consultants recommend that this be instituted.
- (viii) **External explosion:** This appears to have been considered only for Darlington. The consultants query whether one was needed for Bruce, because of the location of the heavy-water plant nearby.



- (ix) Aircraft crash: The calculated probability of an aircraft crash,  $10^{-6}$ /yr at all stations, is acceptably low.
- (x) Fire: The consultants conclude that Ontario Hydro requirements match National Building Code requirements. We have subsequently learned that Ontario Hydro follows the more rigorous requirements set out in Canadian Standards Association standard no. CAN/CSA-N293-M87.

### Design calculations

89. In addition to comparing the criteria used by Ontario Hydro with other standards, the consultants reviewed four sample calculations used in the design. They conclude that:

- design calculation methods were adequate;
- design documentation of record did not include the calculations made to support the design;
- the checking of the Pickering and Bruce calculations seems to have been done within the same group within a few days; the Darlington process uses an organisation independent of the original design team to check the calculations; and
- the process of design verification now appears to take place months or years after the original calculations were made; the consultants recommend that design adequacy be determined prior to construction or installation.

### Containment construction experience

90. The consultants reviewed the containment construction experience as reported in the following types of documents:

- Non-conformance Reports
- Test Reports
- Construction Notices of Deviation

- External Work Requests
- Turnover Record Continuation

Formal documentation of problems or non-conformances dates back only to 1980. A review of those reports did not reveal any problems of safety significance.

### Containment operating experience

91. The consultants reviewed and documented significant events that affect containment performance. No recommendations are made.

#### **2. Radionuclide venting and behaviour in CANDU containment under upset conditions**

92. The ONSR commissioned R.E. Jervis and G.J. Evans, of the Centre for Nuclear Engineering at the University of Toronto, to review the strategy and the means for the venting of airborne radioactivity from the containment in the event of a serious LOCA using the filtered air discharge system.

93. In their report (Jervis and Evans 1987), the authors argue that not all is being done that could be done to prevent radionuclides from being released or filter radionuclides released from the station following a serious reactor accident. They advocate that new technologies should be added to the station as they become available in order that releases can be further reduced, if they are considered an appropriate use of resources. They feel the objective should be, in principle, to eliminate post-accident releases: "The objective of such future technology, to be able to eliminate virtually any finite post-accident discharges from containment, would be consistent with the defence in depth safety approach of CANDU that cites containment as an essential safety system" (Jervis and Evans 1987:18). The technologies that they suggest could be implemented to further reduce the radionuclides emitted include:

- increasing the time of repressurisation of the containment to reduce the in-leakage of air;

- the addition of chemicals to containment water to prevent radioiodines from being released; and
- noble gas abatement methods.

94. Ontario Hydro (1987a), in its reply to this submission, notes that it has made and continues to make improvements to reduce the doses to the public following such an event. It notes the installation of filtered air discharge systems at all stations, a commitment to install pre-discharge monitoring at Pickering NGS, a commitment to reduce air in-leakage at Pickering A NGS in order to increase the repressurisation time, and the existence of chemistry control facilities at Pickering and Darlington. Ontario Hydro also notes that it is committed to carrying out research in noble gas abatement, as noble gas discharge abatement mechanisms are currently extremely expensive (hundreds of millions of dollars) and introduce a new risk, that of the failure of the abatement system releasing all the filtered noble gases at once to the atmosphere.

### 3. Another proposal

95. Nevertheless, the possibility for further changes should not be discounted. Passive filtered venting systems have been implemented in Sweden (by law) (Tiren 1987), and new proposals are being generated for such systems in the United States, which hold promise to further reduce any emissions that might occur.

96. One such proposal was presented by J.L. Dooley et al. (1986) for a radically different containment system, which, it is claimed, could make existing reactors "ultrasafe."

97. The proposed system is called the chill-vent filter system. Rather than "contain" the airborne releases within the vacuum building during an accident, this design would continuously filter the gases escaping from the reactor building and collect effectively all the airborne fission products on the filter beds. The filters consist of a large pile of rock, on top of which is a layer of

activated charcoal. This material, chilled to  $-60^{\circ}\text{C}$  (by a refrigeration plant, which would not be required to operate during an accident), would filter out the noble gases. The activated charcoal would remove iodine and similar fission products.

98. Dooley et al. (1986) estimate the cost of this addition to a single reactor unit to be \$4-5 million. This cost presumably does not include the cost of operating the refrigeration plant. One system could serve a multi-unit station, as the vacuum building does at Ontario Hydro stations.

#### 4. Conclusions and recommendations

99. Stevenson and Associates conclude that the Ontario Hydro containments provide an acceptable level of protection of public health and safety. In their conclusions, they state that two items warrant further investigation:

- shields against turbine missiles for the steam drum; and
- review of fire protection guidelines.

100. Ontario Hydro (1987a) appears to have responded adequately to these recommendations. However, it has not responded to other recommendations buried within the text. These recommendations are that:

- potential sources of internal missiles should be surveyed;
- design documentation of containment should include design calculations; and
- design verification should be undertaken prior to construction.

These recommendations are adopted by the ONSR.

101. The report by Jervis and Evans (1987) and the reply by Ontario Hydro (1987a) make it clear that, through its implementation of filtered air discharge systems at all its nuclear generating stations, Ontario Hydro has been willing to implement improvements to decrease releases from a serious accident. At

present, however, it is not clear what precisely its backfitting policy is on further improvements. It is therefore recommended that Ontario Hydro and the AECB make clear their backfitting policy on the implementation of further improvements to their containment systems. In particular, methods to remove noble gases from the controlled releases of radioactive material, which would be needed following a serious accident, should be further considered.

#### **(d) Ageing**

**102.** The degradation of equipment as it ages is merely one of many possible forms of degradation of performance of process or safety system components. Ageing-related degradation is of particular safety significance for two reasons:

- ageing-related degradation of common components can reduce the reliability of plant systems, thus increasing the frequency of process system failures; and
- ageing-related degradation can reduce redundancy through the simultaneous failures of safety components, even though they may be physically separated.

**103.** The best-known example of ageing-related degradation of components is the corrosion and hydriding of pressure tubes at Pickering units 1 and 2. The failure of a single pressure tube at Pickering Unit 2 led to a discovery that many pressure tubes in the two reactors had hydride blisters. The chances of another pressure tube failure were felt to be sufficiently high that the two reactors were taken out of service to have their pressure tubes replaced.

**104.** The pressure tube problem emerged in large part because there was a failure to recognise that such a phenomenon was possible, and because there were no regular inspections of the conditions of the tubes.

**105.** Other instances of ageing-related failures are cited in the submission of Slee and Rubin (1987) and in a paper by Pachner (1987) of the AECB, presented

at an international conference on reactor ageing. Some examples of ageing-related degradation include:

- lubricants on ECIS valves at Bruce A degraded rapidly, causing valve failures;
- clogging of level indicators for emergency sump water at Pickering A became clogged, impairing the long-term emergency cooling system; and
- high-power detectors for the second shut-down systems at one of the Bruce A units became corroded and failed.

106. Commenting on these examples, Slee and Rubin (1987) state:

These events showed that degradation of components and materials has happened in unexpected ways, which have been detected, where they have been, only by considerable testing. The importance of testing in order to detect such unexpected degradation behaviour is crucial, and underlines the earlier observation that aging phenomena in general are still ill-understood by the nuclear industry. Research will not prevent these kinds of problems from happening. Rather, it is only by learning from actual failures that many age-related problems have been recognized and dealt with.

107. As they point out, testing and inspection must have primary roles to play in the discovery of ageing-related degradation. Indeed, in the examples they cite, it was the testing of the safety systems that revealed their degraded performance. Discovery of the defects on one reactor led to the inspection of other units to see if a similar problem had developed.

108. The effectiveness of testing, inspection, and reporting systems in place at Ontario Hydro is thus the key to detecting ageing-related failures as they occur. An evaluation of this system is presented in Appendix III.

109. Regular inspection of all safety-related components is the most important lesson learned from the pressure tube failure at Pickering. Certain of these components, especially those inside the calandria, are particularly difficult to

get access to and to inspect. Remote methods are being developed to aid this inspection process.

110. Designers can also be of service in anticipating safety-related degradation. US NRC researchers argue convincingly that probabilistic risk assessments can be used to identify components such as valves whose common ageing-related degradation may significantly increase the chance of an accident (Vesely and Vora 1987). It is recommended that the probabilistic safety evaluations to be performed at all stations take account of the performance of aged equipment.

111. Finally, Slee and Rubin (1987) point out that a retirement or delicensing policy is needed by the AECB so that it can state a priori its basis for delicensing a reactor. They argue that it is essential to take into account the effects of ageing-related degradation on the reliability of process and safety systems, and to acknowledge that some of the degradation may not have been detected by normal inspection procedures. It is recommended that the AECB formulate a retirement policy for power reactors, taking into proper account the effect of ageing on the safety of the reactor.

#### **(e) Process system performance**

112. Accidents are avoided if the process systems do not fail. High reliability of the process systems is an indication of sound operation and of sound maintenance practices. In this subsection, two aspects of process system reliability are reviewed:

- frequency of process system failures; and
- reliability of process systems as a function of age.

This review is carried out with the aid of the SERs generated by Ontario Hydro. Nuclear Awareness Project (1987), in its submission to the ONSR, argues that the SERs show that the process systems at Pickering A are degenerating with age. This assessment is also reviewed in this subsection.

## 1. Frequency of process system failures

113. As noted in Section B, reactor accidents can be classified into seven different types:

- loss of coolant from the heat transport system;
- loss of cooling;
- loss of regulation;
- failure of the electrical power systems;
- blockage of flow in a fuel channel;
- failures in the fuel handling systems; and
- failure of the moderator pressure boundary.

Several accidents of each of these types have occurred at an Ontario Hydro nuclear generating station. None has led to a serious accident, i.e., a significant increase in the rate of release of radioactive material from the station.

114. Table II-2 is a list of the frequency of process failures of various types that have occurred at Ontario Hydro's nuclear generating stations. Some examples of each type of accident are summarised below.

### Loss of coolant

115. The two most serious LOCAs (leakage of heavy water from the heat transport system) that have occurred were the two pressure tube failures discussed above. In both cases, significant quantities of heat transport fluid were discharged from the heat transport system. Very small leakage from pressure tubes has occurred on 23 other occasions. Small leaks of heavy water (a few kilograms per hour or less) have occurred at all stations as a consequence of leaky valves or similar problems. On a few occasions (e.g., Ontario Hydro 1985a), the fuelling machine has leaked while refuelling a channel, resulting in a leak of heat transport system water. As with the



Table II-2

Ontario Hydro Experience with Initiating Events of  
Each Accident Type

<u>Accident type</u>	<u>Number and type of event</u>
Loss of coolant	2 Pressure tube failures
	23 Pressure tube leaks
	8 Leaks related to fuelling machine
Loss of regulation	8 Unplanned increases in bulk power
	31 Unplanned increases in regional power
	23 Unplanned decrease in pressure
	7 Unplanned increase in pressure
Loss of cooling	48 Dual computer failures
	15 Loss of cooling leading to a reactor trip
Loss of electrical power	5 Loss of electrical power requiring reactor trip
Flow blockage	2 Flow reduced less than 70% in a fuel channel
Failure of fuel handling systems	3 Fuel bundle handling accidents
Failure of moderator system	3 Moderator heat exchanger tube leaks

Source: Ontario Hydro, adapted from King et al. 1987.

pressure tube failure, the water was recovered from the floor of the reactor building.

### Loss of regulation

116. Three types of control system failures are considered: a loss of power control; a loss of pressure control; and a failure of both control computers. There have been eight unplanned increases in reactor power, seven of which necessitated safety system action. These seven occurred at Pickering A, six of these in the first 4 yr of operation, the seventh in 1982. According to the Pickering A Safety Report, the intended reliability of this system was to be one failure per 100 yr<sup>\*</sup>, meaning that only one failure was expected to occur in the period considered. In addition to these global power increases, which demanded safety system action, 31 unplanned changes occurred in power in some part of the reactor. These have occurred, on some occasions, through unexplained actions of the power control system. For example, there have been at least two cases in which the regulating system has driven out an adjuster rod for no apparent reason. The rod had to be driven back in again by the operator (Ontario Hydro 1979). Loss of pressure control leading to a fall in pressure has occurred on 23 different occasions. The failure of both control computers has occurred 48 times.

### Loss of cooling

117. Impairment of heat removal has most often occurred as a consequence of a reduction in circulation of the steam generator feedwater. On 15 occasions, a reduction of steam generator feedwater flow has occurred leading to a reactor trip; sufficient flow remained to be capable of removing decay heat.

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\*The Pickering A Safety Report specifies the reliability of the control system to be 0.01 per reactor per year.

### Electrical power system failures

118. There has been one occasion when the regular electrical supply, called Class IV power, has been lost to a reactor unit during normal operation. This occurred at a Pickering A unit in 1974. In other cases, Class IV power has been lost to part of a reactor unit. This also requires a reactor shut-down.

### Fuel handling system failures

119. There were a number of serious problems with fuel handling in the early years of Pickering A NGS. In 1973, two irradiated fuel bundles fell off an underwater elevator and became stuck. The bundles were eventually removed, although radioactive debris remained. Also in 1973, a failure of the fuel transfer machine resulted in the heat-up of two irradiated fuel bundles it was carrying. Fuel sheaths on the two bundles ruptured, releasing radiation inside the containment. In 1975, two fuel bundles collided during fuel transfer, breaking one of the bundles. Also in 1975, an irradiated cobalt rod was manoeuvred too close to the fuel storage bay surface. As a consequence, the radiation field from the cobalt forced evacuation of the storage bay area. More importantly, the storage bay was located directly underneath the control room. Operators for units 2 and 3 stood on the opposite side of the control room from the control panels in order to reduce their exposure.\* Performance of the fuel handling system has been better in recent years. The safety-related incidents that have occurred have involved loss of cooling water from the fuelling machine. In these incidents, fuel failures have not occurred.

### Loss of moderator coolant

120. Although less important for the safety of the plant, leakage of moderator has a safety significance because of the tritium contained in moderator water. At Ontario Hydro stations, the moderator heat exchanger is cooled by lake water; if a tube in one of the moderator heat exchangers breaks, tritiated

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\*These incidents, taken from Ontario Hydro Significant Event Reports, were cited in McKay 1983.

heavy water from the moderator system can escape to the lake. Problems with leaky moderator heat exchangers have occurred at Pickering A, Pickering B, and Bruce A. No leak has led to a tritium concentration in drinking water in excess of permissible limits. Because of a faulty design, Ontario Hydro has replaced the moderator heat exchangers at Pickering B.

121. In its submission to the ONSR, Environment Canada (1987) argues that leakages in this heat exchanger are avoidable and that the system should be redesigned to eliminate the possibility that a leakage of moderator water would lead directly to the leakage of tritiated heavy water into the lake. Indeed, AECL has designed its 600-MWe generating station in this way.

## 2. Process system failures and reactor ageing

122. How have these failures been distributed in time? In general, more of the most serious failures have occurred during the early years of operation of the station. For example, six of the eight bulk power increases occurred during the first 4 yr of operation of Pickering A NGS: the most recent occurred in October 1982. The newer stations, Pickering B and Bruce B, have not had any serious process system failures to date.

123. Process system faults are reported in the SERs as consequence code 7. Three types are distinguished:

- 7/3 "Serious" process system faults--those faults requiring safety system action to prevent fuel damage;
- 7/2 "Near-miss" process system faults--those faults that could have been serious had conditions (e.g., reactor power) been different; and
- 7/1 "Other" process system faults--other malfunctions of process systems that may or may not have any direct safety impact on reactor operation.

124. Table II-3 lists the number of process system faults of each type at Pickering A and B and Bruce A and B between 1980 and 1986.

Table II-3

Process System Faults at Ontario Hydro Nuclear  
Generating Stations, 1980-86

A. Category 7/1 Other (Less Serious) Process System Faults

<u>NGS</u>	<u>1980</u>	<u>1981</u>	<u>1982</u>	<u>1983</u>	<u>1984</u>	<u>1985</u>	<u>1986</u>
Pickering A	12	25	7	18	11	24	43
Bruce A	22	11	36	50	23	35	18
Pickering B	-	-	2	17	15	47	28
Bruce B	-	-	-	2	5	13	15

B. Category 7/2 Type B (Near-miss) Process System Faults

<u>NGS</u>	<u>1980</u>	<u>1981</u>	<u>1982</u>	<u>1983</u>	<u>1984</u>	<u>1985</u>	<u>1986</u>
Pickering A	2	0	1	1	1	1	0
Bruce A	2	3	7	1	4	1	1
Pickering B	-	-	0	0	1	1	0
Bruce B	-	-	-	0	0	1	1

C. Category 7/3 Type A (Serious) Process System Faults

Only one fault of this type has occurred since 1980. This occurred at Pickering A NGS in 1982.

Source: Ontario Hydro Significant Event Reports.

125. In its submission, Ontario Hydro (1987b) argues that the frequency of more serious process system faults has been decreasing with time. At Pickering A, nine of the 10 most serious process system faults (those requiring safety system action) occurred prior to 1980. There has been only one occurrence since 1980, and that occurred in 1982. It is a noteworthy comment on this classification scheme that the pressure tube failures in 1983 at Pickering and in 1986 at Bruce did not qualify as serious process system faults because special safety system action was avoided through operator action.

126. In the Nuclear Awareness Project (1987) submission, it is argued that the less serious process system faults (7/1) are occurring with increasing frequency as Pickering A NGS ages. An examination of the Nuclear Awareness Project data for this type of event shows that this conclusion is based primarily on the fact that the number of 7/1 events in 1986 was much higher than in any previous year (Table II-3).

127. One check of the validity of the Nuclear Awareness Project argument is to compare the results for 1987 with the value predicted by their correlation. As of 5 October 1987, there were 18 incidents reported for 1987. The Nuclear Awareness Project formula predicts that 27 incidents would have been reported. If no systematic increase were occurring, only 14 incidents would have been expected.

128. This does not rule out the possibility that the reported indication is an ageing-related problem. A review of the incidents in 1986 was done by ONSR to determine the source of the high number in that year. Of the 44 incidents, 14 were reported that affected common services, whereas in a typical year, only one incident of this type would occur. Most of the problems were attributable to the sewage system, the heavy-water upgrading plant, and stack monitoring systems. Unit 4, with 14 incidents, has had more reports than Unit 3 (with seven). It appears that Unit 4 has had many more process system faults (58 since 1980) than Unit 3 (35) and appears to be having such events with increasing frequency.

129. There does not appear to be a similar pattern for the Bruce A reactors.

### 3. Conclusions

130. On 10 occasions, serious process system faults have occurred that required the action of safety systems to avert fuel damage. On none of these occasions was there a significant release of radioactive material that harmed the public.

131. Other less serious process system faults have occurred that have made it necessary for reactors to be shut down, although fuel damage was not imminent.

132. Most of the serious process system faults have occurred at the older reactors during the early years of their operation. The two pressure tube failures, although not formally defined as "serious," are important exceptions to this pattern.

133. Less serious near-miss process system faults continue to occur, but less frequently than before.

134. The newer reactors are experiencing fewer serious or near-miss process system faults than are Pickering A and Bruce A. The less serious failures appear to be as common in these reactors as before.

135. Less serious process system faults have not, to date, shown a systematic increase with age in most reactor units. Pickering Unit 4 is a possible exception.

#### (f) Safety system performance and reliability

##### 1. Introduction

136. In any reactor, special safety systems must work when required. These systems, and their historical development, are described in Appendix I; here,

experience with the systems is reviewed to see what lessons can be learned from past performance to help predict their reliability in the future.

137. For this purpose, performance of the systems is measured as availability, or its reverse, unavailability, because this term relates to licensing requirements. A system is said to be unavailable if it is not fully available, even if it is only marginally incapable of performing its intended function under certain circumstances.

138. The experience summarised in this subsection is, of necessity, selective: Ontario Hydro reports annually complete statistics for the unavailability of the special safety systems at its stations in its Quarterly Reports.

## 2. Shut-down system performance

139. The shut-down systems at Ontario Hydro's CANDU stations have been activated many times, but seldom because they were required to prevent potential fuel damage. In 128 reactor-years of experience, the shut-down systems (SDS1 and SDS2) at Ontario Hydro's stations have been activated 450 times. Of these, 52% were in response to transients that would not have led to fuel damage even without shut-down system error (Table II-4). Equipment malfunction or operator error caused 46% of the trips. On 10 occasions (approximately 2%), the system was actually required to prevent fuel damage. On seven of these occasions, between 1972 and 1982, a rise in power at one of the Pickering A reactors had to be stopped by the single fast shut-down system that is installed in these reactors. These rises in power occurred as a consequence of control computer failures that have since been corrected. The three other occasions were:

- a loss of Class IV power at Pickering A because of a loss of the external power line during severe weather in 1974;
- a heavy-water leak at NPD Rolphton in 1962; and
- a partial loss of an uninterruptable power supply at NPD in 1964.



Table II-4  
Causes of Reactor Trips

<u>Trip causes</u>	<u>% of trips while reactor critical</u>		
	<u>SDS1 (total 420)</u>	<u>SDS2 (total 30)</u>	<u>Both systems (total 450)</u>
Transient--possible fuel damage in absence of trip	2	0	2
Transient--no fuel damage even in absence of trip	53	46	52
Other (operator error, equipment malfunction)	45	54	46

Source: Ontario Hydro.

The pressure tube failures at Pickering A and Bruce A did not require activation of the shut-down systems: in the former case, the operators slowly decreased power using the power regulating system; in the latter, the reactor was already shut down when the pipe break occurred.

140. The reliability of the shut-down system can depend on human actions as well as on the state of the components. For example, operators can change the setpoints of particular reactor trips or block the activation of a shut-down system. Conversely, there has been a case at Bruce Unit 4 where the trip setpoints should have been changed but were not, resulting in the high-power trip on both shut-down systems being unavailable.

141. The information provided by tests and by reviews of the effect of humans on the system is summarised in overall performance indicators: the reported unavailability and inoperability of the systems. Unavailability refers to any diminution of the system capability for an accident, because of an erroneously set setpoint, a meter that was reading incorrectly, or some other hardware failure. Even when "unavailable," the system may still be operable and able to respond to some (probably most) upsets. It is intended to measure the reliability of the equipment rather than its effectiveness when it is called upon. The shut-down system effectiveness is presumed to have been demonstrated by designers through other safety assessments.

142. To meet AECB requirements, unavailability of the shut-down systems at Pickering A NGS must be less than 0.0003, which translates into an upper limit of 24 h/yr. The AECB has required that more stringent limits be applied to the shut-down systems of subsequent reactor units (0.001, or 8.75 h/yr). Tables II-5 to II-8 list the shut-down system unavailabilities for all reactors to the end of 1986. In general, performance at all stations has been very good and within targets.

143. For those reactors equipped with two independent shut-down systems, both systems must be unavailable simultaneously for the reactor to have real shut-down unavailability rather than a loss of redundancy. The unavailability of

Table II-5

Unavailability of Shut-down System,  
Pickering A Nuclear Generating Station

(value x  $10^{-3}$  yr/yr)  
(regulatory target:  $1 \times 10^{-3}$  yr/yr)

<u>Year</u>	Unit			
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>
1972	0.1	0.09	0	-
1973	0.13	1.27	0	0.54
1974	0	0	1.5	0
1975	0	0	13.7	0
1976	0	0	0	0
1977	0	0	0	0
1978	0	0	0	0
1979	0	0	0	0
1980	0	0	0	0
1981	0	0	0	0
1982	0	0	0	0
1982	0	0	0	0
1983	0	0	0	0
1984	0*	0*	0	0
1985	0*	0*	0	0
1986	0*	0*	0	0

\* Out of service for retubing.

Source: Ontario Hydro 1972-1986, summarised as Table F1 in Nuclear Awareness Project 1987.

Table II-6

Unavailability of Shut-down Systems,  
Bruce A Nuclear Generating Station

(value x  $10^{-3}$  yr/yr)  
(regulatory target  $3 \times 10^{-3}$  yr/yr)

Year	SDS1 at Unit:				SDS2 at Unit:			
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>
1976	-	0	-	-	-	0	-	-
1977	0	0	0	-	0	5.04	21.9	-
1978	2.2	0.10	4.5	-	4.7	0.02	2.2	-
1979	0	0	0.36	0.10	2.2	0.03	0.8	0.1
1980	0.03	0.09	0	<0.01	0.02	0.1	0	5.8
1981	0.01	0	0	21	0.86	0	0.08	21
1982	0.15	0.83	0	0	0.64	0.83	0	0
1983	0	0	0	0	0	0	0	0.38
1984	0	0	0	0	0	0	0	0.38
1985	0	0	0	0	0	0	0	1.48
1986	0	1.07	0	0	0	3.27	0	0

Source: Ontario Hydro 1976-1986, taken from Table F3 in Nuclear Awareness Project 1987.

Table II-7

Unavailability of Shut-down Systems,  
Pickering B Nuclear Generating Station

(value x 10<sup>-3</sup> yr/yr)  
(regulatory target 1 x 10<sup>-3</sup> yr/yr)

<u>Year</u>	<u>SDS1 at Unit:</u>				<u>SDS2 at Unit:</u>			
	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>
1983	0	-	-	-	0	-	-	-
1984	0	0	-	-	0	0	-	-
1985	0	0	0	-	0.1	0	0	-
1986	0	0	0	0	0	0	0	0

Source: Ontario Hydro.

Table II-8

Unavailability of Shut-down Systems,  
Bruce B Nuclear Generating Station

(value x  $10^{-3}$  yr/yr)  
(regulatory target  $1 \times 10^{-3}$  yr/yr)

<u>Year</u>	SDS1 at Unit:				SDS2 at Unit:			
	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>
1984	-	0	-	-	-	0	-	-
1985	0	0	-	-	0	0.02	-	-
1986	0	0	0	-	0	0	0	-

Source: Ontario Hydro.

both systems is required by the AECB to be less than one in a million. Assuming perfect independence of the two systems, this requirement suggests that each system should not be unavailable for more than  $1 \times 10^{-3}$  yr/yr, i.e., about 32 s/yr, or more than about 16 min for a 30-yr station life.

**144.** Table II-9 lists these cases of dual system unavailability at Bruce A NGS. Some of these have occurred when a change in the operating state of the reactor has required a change downward of the trip setpoint for high reactor power. Such changes can be the result of reactor transients, when the system unavailability can be measured in seconds. In some cases, however, such as those involving a higher than normal coolant temperature or a lower than normal coolant pressure, the incidents have lasted minutes or hours.

**145.** There was one case at Bruce A in which the two shut-down systems did not provide adequate protection against a slow increase in power, had one occurred during a 7-d period. When the power distribution in the reactor core is unusual, the trip setpoints for measuring high power have to be adjusted downward. At present, at Bruce A, this operation requires the operating staff to adjust the setpoints (by turning a switch) to the lower values. If these adjustments are not made, the high-power trip is ineffective. For most possible system upsets, ineffectiveness of the high-power trip would only reduce the redundancy of the shut-down system: other trip parameters would be detected, leading to a trip of the shut-down system. If, however, a power excursion were to occur slowly owing to failure of the regulatory system, the high-power trip would provide the sole protection on each shut-down system. In the incident at Bruce A, the high-power trip for both systems had been made inoperable by the failure of the operators to change the trip setpoints. Thus, there was no automatic protection against a failure of the regulating system leading to a slow power increase.

**146.** Clearly, the event described in the previous paragraph (Ontario Hydro 1981c) accounts for nearly all the unavailability. From this analysis, it is possible to estimate the dual shut-down system unavailability for Bruce A as 0.0004, much larger than the one-in-a-million design goal. The other CANDU

Table II-9

Bruce A Nuclear Generating Station Shut-down Systems,  
Coincident Unavailability

<u>SER</u>	<u>Dual system actual past unavailability frequency (<math>\times 10^{-3}</math>) (time)</u>	<u>Cause</u>
<u>Unit 1</u>		
80-017	0.03 (14 min)	Pump trip
81-023	<0.01 (5 min)	Pump trip
82-031	<0.01 (5 min)	High coolant temperature
82-081	0.10 (55 min)	Incorrect NOP* handswitch position
<u>Unit 2</u>		
80-108	0.09 (49 min)	Coolant pressure <8.8 MPa
80-118	<0.01 (53 s)	Pump trip
81-78	0.11 (60 min)	Coolant pressure <8.8 MPa
82-9, -10	0.05 (24 min)	Coolant pressure <8.8 MPa
82-32	<0.01 (1 min)	Pump trip
82-69	0.83 (7.3 h)	High coolant temperature
<u>Unit 3</u>		
82-14	<0.01 (41 s)	Coolant pressure <8.8 MPa
82-16	<0.01 (4.2 min)	Coolant pressure <8.8 MPa
<u>Unit 4</u>		
80-76	<0.001 (28 s)	Pump trip
80-141	0.03 (18 min)	Coolant pressure <8.8 MPa
81-92	19.18 (7 d)	Incorrect NOP setpoint

\* Neutron overpower.

Source: Ontario Hydro.



stations with two shut-down systems (Pickering B and Bruce B) have had no reported cases of dual shut-down system unavailability. Adding their experience, the estimate for dual shut-down system unavailability drops to 0.0003.

147. The effect of this relatively high unavailability on the frequency of accidents leading to fuel element failure can be demonstrated by calculating the estimated frequency of a slow rise in power combined with a failure of the shut-down systems to operate. According to the Darlington Probabilistic Safety Evaluation (DPSE) (King et al. 1987), the frequency of an unplanned increase in bulk power is 0.08 per reactor-year. Hence, the frequency of this triple failure would be approximately  $0.08 \times 0.0003 = 0.00002$  per reactor-year. The frequency of this scenario is well above the cut-off frequency for analysis by the DPSE, but it appears that analysis of this event was not carried out.

148. The other instances of shut-down system unavailabilities affected only one system and not the other. Hence, these cases did not imply that shut-down capability was unavailable.

149. In addition, there has been one instance in which both shut-down systems were made inoperable. During the pressure tube failure at Bruce Unit 2, with the reactor already shut down, the operator, believing that the level of water in the calandria was low, was concerned that if either shut-down system were activated, the shut-off rods might be damaged, or a poison injection might damage the calandria. To avoid these possibilities, the operator lowered the shut-off rods one by one into the calandria, added neutron poison to the moderator system, and turned off the poison injection system. As the reactor was already shut down and the shut-off rods were inserted, the reactor was effectively shut down at the time.

### 3. Emergency coolant injection system performance

150. The evolution of the ECIS for CANDU reactors is described in Appendix I. The need to avoid inadvertent operation of the system means that it has

never been used in practice, and on only one occasion has a full test been performed to demonstrate that it works as designed. Demonstration of the system's effectiveness must therefore rely on a combination of analysis and experiments. For the latter purpose, experimental rigs are constructed to reproduce the conditions under which injection of emergency coolant may be needed. The latest of these experimental rigs, RD-14 at Whiteshell, is the first attempt to simulate at full scale all the key aspects of the CANDU heat transport system for testing under loss of coolant conditions. The results of these experiments are helpful in increasing the understanding of the phenomena affecting the emergency coolant injection effectiveness. These experiments also provide data with which the accuracy of analytical tools used to predict the results of a LOCA can be assessed.

151. Experience with ECIS operation in actual reactors is very limited. In fact, the system has only once been triggered in any of Ontario Hydro's operating reactors. This occurred at Bruce B NGS in 1985 and was due to an error on the part of the operator during a test while the reactor was shut down. The ECIS operated as designed during this accidental test. A commissioning test at Pickering B Unit 7 was also performed to determine how effectively the system operated. The Pickering B system was shown to perform satisfactorily.

152. As with the shut-down systems, the reliability of the ECIS is assessed by regular testing of its components. Table II-10 lists some of these tests and their frequency for Bruce A. Operating maintenance also may uncover faults or flaws in the ECISs. Unlike the shut-down systems, there are no full-system performance tests to evaluate the response of the entire system. This is an important check in shut-down system reliability, but is not possible for the ECIS. Thus, the quality of information on ECIS reliability is not as good as for the shut-down systems.

153. The unavailability data for the ECISs at the various stations are given in Tables II-11 to II-14. Both Pickering A and Bruce A had dreadful unavailability records in their early years of operation. Pickering A continues to have

Table II-10

Emergency Coolant Injection Testing Frequency,  
Bruce A Nuclear Generating Station

<u>Test</u>	<u>Frequency</u>
Heavy-water injection valves	1 per 4 weeks
Pressuriser level alarm	1 per 4 weeks
Emergency coolant recovery pumps	2 per 4 weeks
Recovery system check valves	4 per 4 weeks
Heat exchanger cooling water	1 per 4 weeks

Table II-11

Emergency Coolant Injection System Unavailability,  
Pickering A Nuclear Generating Station

(value  $\times 10^{-3}$  yr/yr)  
(regulatory target:  $1 \times 10^{-3}$  yr/yr)

Year	Unit			
	1	2	3	4
1972	2.6	12.7	2.5	NA
1973	2.2	1.7	11.1	1.1
1974	119	1.2	621	514
1975	840	500	500	0.4
1976	11	340.4	0.3	340.4
1977	0	0.01	0	52.1
1978	0.13	9.59	0	79.43
1979	9.9	19.5	0.4	10.1
1980	13.1	1.7	29.9	26.4
1981	1.0	0.7	0.3	0.3
1982	0.3	0.3	0.3	0.3
1983	9.9	9.9	10.0	19.5
1984	0.3	0.3	1.07	0.3
1985	0	0	0.3	0.3
1986	0.3	0.3	19.8	1.35

Source: Ontario Hydro 1972-1986, taken from Nuclear Awareness Project 1987.

Table II-12

Emergency Coolant Injection System Unavailability,  
Bruce A Nuclear Generating Station

(value x 10<sup>-3</sup> yr/yr)  
(regulatory target: 1 x 10<sup>-3</sup> yr/yr)

<u>Year</u>	Unit			
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>
1976	-	0	-	-
1977	22.3	15.9	121.9	-
1978	22.7	21.3	12.5	-
1979	204	206	215	194
1980	11	6	7	3
1981	0.28	0.28	0.28	6.88
1982	0.04	0.04	0.09	0.04
1983	0	0	0.04	0
1984	<0.01	<0.01	<0.01	<0.01
1985	0	0	0	3.31
1986	0	0	0	0

Source: Ontario Hydro 1976-1986, taken from Nuclear Awareness Project 1987.

Table II-13

Emergency Coolant Injection System Unavailability,  
Pickering B Nuclear Generating Station

(value x  $10^{-3}$  yr/yr)  
(regulatory target  $1 \times 10^{-3}$  yr/yr)

<u>Year</u>	<u>Unit</u>			
	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>
1983	3.2	-	-	-
1984	0	0	-	-
1985	0	0	0	0
1986	0	0	0	0

Source: Ontario Hydro.

Table II-14

**Emergency Coolant Injection System Unavailability,  
Bruce B Nuclear Generating Station**

(value x 10<sup>-3</sup> yr/yr)  
(regulatory target 1 x 10<sup>-3</sup> yr/yr)

<u>Year</u>	<u>Unit</u>			
	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>
1984	-	0	-	-
1985	1.19	1.19	-	-
1986	0.3	0.3	0.3	-

Source: Ontario Hydro.

problems meeting unavailability targets, although the record is somewhat improved. The Bruce A system availability has improved substantially, with all units meeting their targets since 1980 with a single exception. This unavailability does not take into account the discovery that the ECIS would not be fully effective in some cases when it was needed. The statistics of Tables II-11 to II-14 are concerned with only the unavailability and not the effectiveness. The latest example of this point was revealed when Ontario Hydro safety analysts discovered that for a pipe break in a certain location at Pickering A, neither the old nor the new ECIS would be fully effective. After informing the AECB of the problem, they suggested an alteration of the design that would solve the problem. However, they were adamant that this change not be counted as part of the unavailability statistics, arguing that this statistic should reflect equipment performance rather than design capability.

154. Many of the ECIS faults were caused by a failure of a valve to open or close when it was tested. In a few cases, electrical failures made certain key system components (e.g., pumps) unavailable. Two recent events are especially significant. First, the ECIS at Bruce B was impaired for 9.75 h in 1985 when pressurised nitrogen leaked out of the emergency water injection tank. The operator on shift when this occurred was not aware that the system was impaired. The impairment was uncovered by an operator on the next shift. Second, in 1987, at Bruce A, seven out of eight injection valves for one unit were set incorrectly in the closed position. This would have severely diminished the capability of the ECIS had a LOCA occurred.

#### **4. Containment performance**

155. Containment is the ultimate barrier to prevent the release of fission products to the environment in the event of an accident. None of Ontario Hydro's operating CANDU stations has ever required containment isolation and activation of the vacuum building. The dousing system was activated at NPD once in 1962 as a result of a leak of heavy water from a fuelling machine. Also, an accidental douse at the CANDU 600-MWe reactor at Gentilly was



induced by a faulty test procedure. There has been no other actual operation of the systems.

156. The performance of containment during an accident depends on several factors. Proper operation of the containment in response to a discharge of steam and water into the reactor vault should include the following:

- (i) The containment must be isolated on detection of high pressure or high activity in the vault. This includes turning off the ventilation system and closing the dampers.
- (ii) The self-actuating pressure relief valves must actuate to connect the vacuum building (which must initially be at very low pressure) to the reactor vault so as to suck the radioactive steam into the vacuum building. The rise in pressure inside the vacuum building should induce dousing water to rain downwards and condense the steam.
- (iii) No leaks must exist in the containment. Such holes could arise from faulty construction, equipment malfunction, or operator negligence.
- (iv) The filtered air discharge system must be operable to allow a controlled discharge in the long term.

157. Containment tests are routinely carried out on the components that activate containment isolation. Some components are tested only rarely, and a test of the vacuum building and dousing system cannot be performed unless all the reactors in the station are shut down. Containment tests at Bruce A are listed in Table II-15.

158. The vacuum building dousing system is tested approximately every 10 yr. The Pickering vacuum building had its last douse test in 1980, Bruce A in 1987. The Pickering test showed that the dousing operation functioned as expected, with one exception. Within a few hours, the water used for dousing leaked through the floor of the building into the basement, where important control equipment was located. It was doubtful whether access to this equipment would

Table II-15  
Containment Testing Frequency,  
Bruce A Nuclear Generating Station

<u>Test</u>	<u>Frequency</u>
Leak rate test	4 per year
Pressure relief valve inspection	8 per 4 weeks
Vault cooling fans	1 per week
Airlock leak rate test	1 per week
Containment pressure trend	1 per shift
Reactor vault pressure	Continuous

have been possible under accident conditions, especially as the water would have been radioactive. Ontario Hydro has repaired the floor with an epoxy sealant.

159. At the Bruce A douse test also, water leaked into the vacuum building basement. On this occasion, water flowed through the vacuum piping system into the basement and into a floor drain, which ultimately drained to the lake.

160. Unavailability of the containment systems at Pickering and at Bruce are documented in Tables II-16 to II-17 and II-18 to II-19, respectively. Pickering A has experienced many problems with airlock door seals, especially during the early years of operation. Many of these events occurred as a consequence of a single door seal being deflated when the other door was opened. There have also been some more serious problems. In 1974, a 130-cm<sup>2</sup> hole around a reactor building pipe penetration in Unit 2 was discovered. The size of the hole would have led to a significant release in the event that the containment became pressurised. In 1977, a 14-cm<sup>2</sup> hole was discovered. However, containment unavailability has been very low since 1980.

161. Pickering B containment unavailability has been within target in every year of operation.

162. Up to 1983, Bruce A containment unavailability was always within target. However, both Bruce A and Bruce B have had incidents that led to unavailability targets being exceeded in the past 2 yr. At Bruce, unavailability of containment at one unit usually means that the containment is unavailable for an accident in any of the four units. The most noteworthy incidents illustrating containment unavailability are as follows:

- (i) At Bruce A in 1976, 11 of the 16 pressure relief valves that are supposed to lift in case of a LOCA were found to be incorrectly installed. This would have diminished the rate at which the vacuum could have reduced containment pressure.

Table II-16

Containment System Unavailability,  
Pickering A Nuclear Generating Station

(value x  $10^{-3}$  yr/yr)  
(regulatory target:  $1 \times 10^{-3}$  yr/yr)

<u>Year</u>	<u>Unit</u>			
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>
1972	0	0.5	0	NA
1973	4.2	4.2	6.5	3.8
1974	1.1	502	1.1	2.0
1975	0	0.2	1.2	0.01
1976	8.2	8.2	8.2	8.2
1977	1.1	0.73	0.75	0.71
1978	0	0.03	0.02	0
1979	6.4	6.4	6.4	6.4
1980	1.6	1.6	1.6	1.6
1981	0	0	0	0
1982	0	0	0	0
1983	0	0	0	0
1984	0	0	0	0
1985	0	0	0	0
1986	0	0	0	0

Source: Ontario Hydro 1972-1986.

Table II-17

Containment System Unavailability,  
Pickering B Nuclear Generating Station

(value x  $10^{-3}$  yr/yr)  
(regulatory target  $1 \times 10^{-3}$  yr/yr)

<u>Year</u>	<u>Unit</u>			
	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>
1983	0	-	-	-
1984	0	0	-	-
1985	0.3	0	0	0
1986	0	0	0	0

Source: Ontario Hydro.

Table II-18

Containment System Unavailability,  
Bruce A Nuclear Generating Station

(value x  $10^{-3}$  yr/yr)  
(regulatory target:  $1 \times 10^{-3}$  yr/yr)

Year	Unit			
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>
1976	-	0	-	-
1977	0.01	0.01	0.01	-
1978	0.7	0.7	0.7	-
1979	0.01	0.01	0.01	0.01
1980	0	0	0	0
1981	0.08	0.08	0.08	0.08
1982	0.42	0.44	0.42	0.42
1983	0.30	0.31	52.1	0.30
1984	0	0.01	0	0
1985	4.47	0	0	0
1986	402.8	<0.01	<0.01	100.7

Source: Ontario Hydro 1976-1986, taken from Nuclear Awareness Project 1987.

Table II-19

Containment System Unavailability,  
Bruce B Nuclear Generating Station

(value x  $10^{-3}$  yr/yr)  
(regulatory target  $1 \times 10^{-3}$  yr/yr)

<u>Year</u>	<u>Unit</u>			
	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>
1984	-	0	-	-
1985	0.9	0.9	-	-
1986	0.06	0.06	0.06	-

Source: Ontario Hydro.

- (ii) In 1986, Unit 1 of Bruce A was unavailable for 180 d as a consequence of a bypass of the containment box-up monitors. The bypass meant that there was an unmonitored flow path out of containment. For most accidents in which radiation might be released, the containment would be isolated by other monitors. However, for one scenario (the melting of the enriched-uranium reactivity boosters), there would have been no automatic isolation.
- (iii) In 1985, a seal on the Bruce B vacuum building ruptured, and the pressure in the vacuum building reached atmospheric pressure. Because this would have made the vacuum building ineffective in the event of an accident, the operating reactors (Bruce units 5 and 6) had to be shut down.
- (iv) In 1987, the pressure in the upper chamber of the Bruce B vacuum building increased to one-half of the atmospheric pressure (the normal value is about one-tenth) as a result of malfunctioning of a vacuum pump.

System component unavailabilities are given in Tables II-20 and II-21.

### **(g) Backfitting**

**163.** The process of adding or modifying equipment at a nuclear generating station is called backfitting or retrofitting. There are two principal intentions of making such modifications: either the original design never worked as it was intended, or the system is being upgraded to meet more modern standards.

#### **1. Safety system backfits**

**164.** Tables II-22 to II-24 document the changes that have been made to the safety systems of Pickering A and other stations. Most of these changes were made to add design innovations (Figure II-3) that were included with newer reactors.



Table II-20

**Summary of Component Unavailabilities,  
Bruce A Containment**

<u>Component</u>	<u>Predicted future unavailabilities* x 10<sup>-3</sup> yr/yr</u>
A. Automatic containment isolation	0.118
B. Airtlocks, containment integrity	0.057
C. Pressure relief valves	0.111
D. Dousing water system	0.015
E. Negative differential pressure	<0.01
<b>Q<sub>CONT</sub> = Σ(A+B+C+D+E)</b>	<b>0.3110</b>

\* Values given are based on design data from Ontario Hydro (1985b).

Table II-21  
Summary of Component Unavailabilities,  
Bruce B Containment

<u>Component</u>	<u>Predicted future unavailabilities* x 10<sup>-3</sup> yr/yr</u>
A. Containment envelope	0
B. Containment isolation	0.16
C. Pressure relief valves	0
D. Auxiliary pressure relief	0.20
E. Instrumented pressure relief valves	0.12
F. Vacuum system	0.025
G. Emergency water storage and water spray system	0
H. Airlocks and transfer chambers	0.36
<b>Q<sub>CONT</sub> = Σ(A+B+C+D+E+F+G+H)</b>	<b>0.865</b>

\* Values given are based on design data from Ontario Hydro (1985b).

Table II-22

## Shut-down System Backfits

Original system

Pickering A original shut-down system used 11 shut-off rods combined with moderator dump to achieve fast shut-down. Subsequently, two independent shut-down systems have been used.

The number of shut-off rods used for the shut-down system at Pickering A (11) was much lower than at subsequent reactors.

Fewer detectors are used at Pickering A than at subsequent reactors.

Consequence

The consequence of having only a single shut-down system available is an increased chance of a failure of the reactor to shut down in case of a power excursion.

Fewer rods meant that higher fission product releases from the fuel could be expected should a LOCA occur.

The chance of shut-down system failure is increased.

Backfit

Ontario Hydro has argued that the addition of a second shut-down system is unnecessary, as the frequency of such an event is low and the consequences are tolerable.

Ten more shut-off rods have been added to units 1 and 2. Units 4 and 3 will get an additional 10 rods in 1988 and 1989, respectively.

Table II-23

## Containment System Backfits

<u>Original system</u>	<u>Consequence</u>	<u>Backfit</u>
No filtered air discharge provisions on original Pickering A and Bruce A.	Higher releases of radioiodines.	Filtered air discharge systems have been backfitted at Pickering A and Bruce A.
Repressurisation time higher at Pickering A than at subsequent reactors.	Earlier time for controlled releases to begin. Higher total releases.	In-leakage of pressurised air has been reduced.
Hydrogen mitigation systems not installed at any station.	Concern about hydrogen generation and ignition.	Backfits to all stations have been made (Bruce A not yet complete).
No instrumented pressure relief valves at Bruce A.	Limited relief capability at low-pressure conditions.	Backfit made.
No monitoring of vacuum building radioactivity prior to release.	Quantity of radioactive material being released could be assessed only once controlled discharge began.	Backfit made to allow recirculation of contaminated air through the filter system back to containment.

Table II-24

**Emergency Coolant Injection System Backfit****Original system**

Original Pickering A and Bruce A systems had low-pressure ECIS.

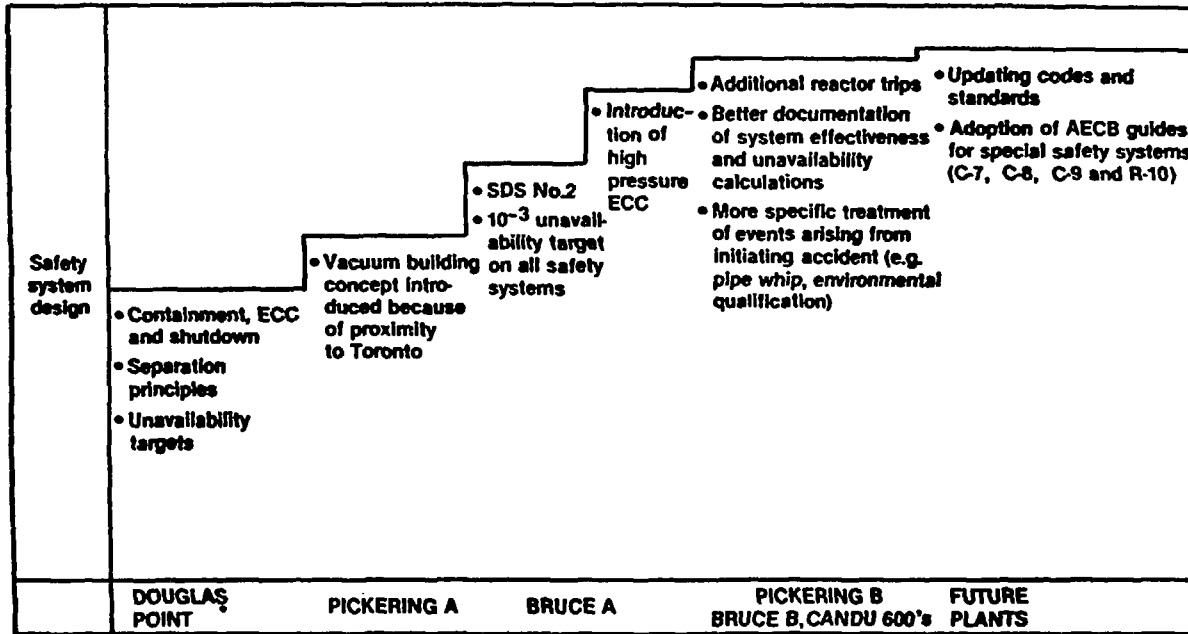
**Consequence**

This system was subsequently found to be ineffective for large pipe breaks. Releases would be much higher than anticipated.

**Backfit**

High-pressure systems have been added at Bruce A (completed in 1987) and at Pickering units 1 and 2. Pickering units 4 and 3 will be backfitted in 1988 and 1989, respectively.

Figure II - 3



Licensing Through the Years -- Safety System Design

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165. As Slee and Rubin (1987) point out in their paper, the addition of the high-pressure emergency coolant system at Pickering and Bruce was the outcome of research that showed that the original system did not perform as well as intended. A more disturbing aspect of these backfits was that it has taken a very long time between the recognition that the ECIS did not meet the original design intent and the installation of the better system.

166. Additionally, there are three important areas in which the older reactors are less safe than newer stations:

- Pickering A reactors have only one shut-down system;
- Pickering A and Bruce A reactors do not have emergency water supplies and emergency power supplies for protection against common-mode failures; and
- explicit environmental qualification of equipment was not done for Pickering A and Bruce A.

These areas are discussed below.

#### Second shut-down system

167. The need for a second shut-down system at Pickering A will be discussed extensively in Section F.

#### Common-mode failure

168. Additional provisions were made at stations subsequent to Bruce A for the prevention of common-mode failures (those externally caused events that could affect more than one reactor at the same time). These improvements were intended to ensure that all reactors could be shut down, the fuel could be cooled, and the condition of the reactor could be monitored. This was accomplished through the provision of an emergency power supply, an emergency water supply for heat removal, and a secondary control area to monitor the conditions inside the reactor. In addition, the ability of important safety-

related components to withstand an earthquake was studied and tested. Such improvements have been made at Pickering B, Bruce B, and Darlington. Pickering A and Bruce A do not have emergency water or power supplies.

### Environmental qualification

169. Environmental qualification is a term used to describe the formal process of verifying the capability of equipment to operate in a hot, moist, and potentially radioactive environment. Two types of environments are considered: the containment environment following a heat transport system pipe break, and the powerhouse and control room environment if one of the steam or feedwater pipes were to rupture.

170. If a pipe break occurs in the heat transport system, equipment inside the containment could be exposed to an environment of high temperature, humidity, and radiation. It is important that equipment that is needed as part of the accident response be able to withstand this environment. Formal environmental qualification places design requirements on equipment performance under these adverse conditions. This process was undertaken for Pickering B, Bruce B, and Darlington, although it is admitted that the existing standards are not up to world standards (Humphries, quoted in Slee and Rubin 1987). Pickering A and Bruce A have not had formal environmental qualification reviews.

171. A similar problem exists for equipment in the powerhouse and control room. The powerhouse environment would become hot and moist should a steam or feedwater line of one of the reactors rupture. This environment could be detrimental to the operation of equipment in the powerhouse or, more importantly, could affect people and equipment in the control room. Modifications were made to all stations to qualify equipment, to allow steam to be vented out of the powerhouse, and to protect the control room personnel and equipment.



## 2. Lack of backfitting policy

172. Slee and Rubin (1987) criticise the lack of a backfitting policy and the negotiation process that goes on in its place. They argue that the AECB should have a clear backfitting policy. Furthermore, they argue that the backfitting policy should be based upon the principle of the old reactors meeting the new reactor standards.

173. Ontario Hydro (1987a:E5-1), in its reply to the Slee and Rubin (1987) submission, argue that this recommendation:

is not required, is not achievable and is counter-productive. Ontario Hydro uses the original safety criteria as the minimum basis for continued operation. It is also policy to improve safety to the extent practical regardless of compliance with minimum standards and to upgrade existing plants where new insights show the original standards were unacceptable. The result of the recommendation would be that new safety developments would be harder to pursue.

174. The origin of the disagreement between the two positions is the question of what are the basic safety criteria for the licensing of a nuclear generating station, and what, if any, of the criteria are additional safety measures. Slee and Rubin (1987) argue that all AECB criteria or standards must be considered minimum standards. If this were the case, it logically follows that all plants must meet those minimum standards or not be permitted to operate.

175. Many of the improvements that have been made at Pickering A were made despite the fact that, according to the accident analysis at the time, the minimum standard (the dose limits) could be met with the current equipment. Interestingly, despite these improvements, the most recently estimated doses from a LOCA at a Pickering A reactor are predicted to be higher than they were prior to these improvements being implemented (see Section D). If these improvements had not been made at Pickering A, it is difficult to say whether the dose limits could have been met. It appears prudent, therefore (as Slee and

Rubin argue), to take account of the uncertainties in safety analysis in the evaluation of the necessity of backfitting.

176. It does seem prudent to assure that essential equipment is environmentally qualified to operate following a LOCA, as the functioning of this equipment was assumed in the original design of the plant.

### **3. Recommendations**

177. It is recommended the AECB promulgate a backfitting policy. This policy should take into account the effect of analysis uncertainties on the safety of operating reactors. Furthermore, the policy should make clear that schedules for needed retrofits will be adhered to.

178. If and when ageing-related degradation begins to reduce the reliability of process and safety systems in a nuclear reactor, the frequency of in-service inspection and maintenance of the affected equipment should be increased.

179. As recommended by Slee and Rubin (1987), the AECB should formulate a retirement policy for each plant. Such a policy should take into account the signs of an ageing nuclear plant, such as poorer reliability of its process and safety systems, and the fact that certain safety-related defects may go undetected. Such a policy must also account for the degree of redundancy of important safety systems or safety-related systems. It is quite proper to insist on a higher level of reliability for the process and safety systems for those plants that have fewer safety systems.

180. The probabilistic safety evaluations that Ontario Hydro intends to undertake for its operating nuclear reactors should account for the effects of ageing on the actual reactor equipment.

181. The environmental qualification process is intended to provide assurance that essential safety equipment will be capable of operating when it is required. It is sensible to require reasonable assurance that this equipment will operate.

Appropriate audits of equipment in older plants should be undertaken, and retrofits made where equipment is found to be lacking.

#### **D. Accident Analysis**

##### **(a) Introduction**

182. Accident analysis is a useful tool to ensure the safety of hazardous technologies. It forms an essential component of the licensing of nuclear reactors. Ontario Hydro's (1985c:13-1) stated purpose in performing accident analyses\* is "to demonstrate that the station design is such that any radioactive releases predicted to occur following postulated accidents will not lead to a radiation dose to a member of the public in excess of the guidelines established by the Atomic Energy Control Board (AECB)." The purpose of this section is to determine how confident one can be of the accuracy of the above demonstration. This section describes what accident analysis consists of, and how it is performed; identifies specific areas of uncertainty in the analysis; examines how the results of accident analysis for Ontario Hydro's stations have changed with time, to see if the approach has been consistent; and considers how certain principles of quality assurance (QA) can be applied to reduce uncertainties in the analysis and increase confidence in the results. Throughout, points raised by consultants and intervenors are identified and summarised.

##### **(b) The nature of accident analysis**

183. Accident analysis is divided into deterministic and probabilistic analysis. Probabilistic safety analysis of reactors consists, in essence, of identifying all conceivable accidents that could occur to the reactor under study, then of estimating the probability and consequences of these accidents. This analysis is considered further in the next section. Some safety analyses are deterministic,

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\*Ontario Hydro now uses the term "safety analysis" rather than "accident analysis." However, the author believes the latter term is more precise.

in that they do not attempt to assess the probability of the accidents, but simply assume that certain systems or components fail.

184. In the accident analysis required by the AECB for licensing reactors, the applicant must perform deterministic analyses for the assumed failure of the process systems shown in Table II-25. Also considered is the assumed coincidental failure of a process system and one of the safety systems (Table II-26). Assessments are also required of another possible sequence that could result from external events such as earthquakes, tornadoes, and airplane crashes (Table II-27). Failures of essential safety support systems, such as electrical power supplies cooling water and pressurised air, must also be assessed, but in a probabilistic manner (Table II-27).

185. These analyses are carried out principally by the Nuclear Studies and Safety Department of Ontario Hydro. The results are reviewed internally before being submitted as part of the station's Safety Report to the AECB staff for review. Figure II-4 illustrates the process as it currently exists and shows how accident analysis fits into the broader safety design process.

186. Two reasons for accident analyses being difficult to perform are the complexities of the systems involved and the existence of interacting processes. As a result of the complexities, the analysis has become compartmentalised to allow specialists to analyse in depth their own particular fields. This compartmentalisation, however, brings with it the risk of overlooking the interacting processes. Ontario Hydro's approach to this acknowledged problem is to apply the methods of quality engineering to ensure that the needed co-ordination is achieved. Figure II-5 illustrates the interaction between different specialties needed for the analysis of a LOCA.

187. One characteristic of this approach is the division of the accident sequence into a series of steps and the assigning of goals, which may be qualitative, to the groups of specialists analysing the various steps. One example of such criteria is those used to show that the reactor core will remain intact so that the fuel can be adequately cooled. Table II-28 illustrates the

Table II-25

## Serious Process Failures Analysed as Part of Accident Analysis

## Failures of heat transport coolant piping

- failure of piping that goes outside containment
- failure of a single steam generator tube
- failure of feeder-sized pipes
- failure of a pressure tube and surrounding calandria tube
- failure of a fuel channel end fitting
- failure of large pipe headers

## Failures in heat removal

- loss of feedwater to the steam generators
- failure of a large steam pipe

## Failures in the reactor control system

- failure of the power control (regulating) system
- failure of the pressure control for the heat transport system
- failure of the pressure control for the steam generators

## Failures of coolant circulation

- loss of power to heat transport pumps
- seizure of a heat transport system pump
- flow blockage in a fuel channel

## Electrical system failures

- loss of Class IV power
- loss of Class IV, III, II, and I power

## Fuel handling accidents

- fuelling machine failures while attached to the reactor
- fuelling machine failures while transporting spent fuel

Table II-26

## Safety System Impairments Considered in Accident Analysis

Shut-down Systems

## One shut-down system (Pickering A)

## Impairment assumed (21 rods)

- two most effective rods do not fall in
- first trip does not work
- second trip does work

## Impairment considered

- complete failure of shut-down system is also considered

## Two shut-down systems (Pickering B, Bruce A and B, Darlington)

## Impairment assumed

- on SDS1, the two most effective rods do not fall in
- on SDS2, the most effective injection nozzle does not work
- reactor trip occurs on second signal from the least effective of these two systems, i.e., most effective system not working

## Impairment considered

- complete failures of both systems not considered (triple failure)

Emergency Coolant Injection System

## Impairment assumed

- first two emergency cooling pumps do not start (Pickering)
- isolation of the two heat transport loops does not occur (Pickering A)

## Other impairments considered

- ECIS unavailable
- failure to isolate the two loops (Pickering B)
- failure of crash cool-down of steam generators

Containment

## Impairments assumed for intact containment

- three of 12 pressure relief valves do not work

## Impairments of containment considered for impaired containment

- containment isolation dampers do not close
- airlock seals deflate
- loss of air cooling

Table II-27

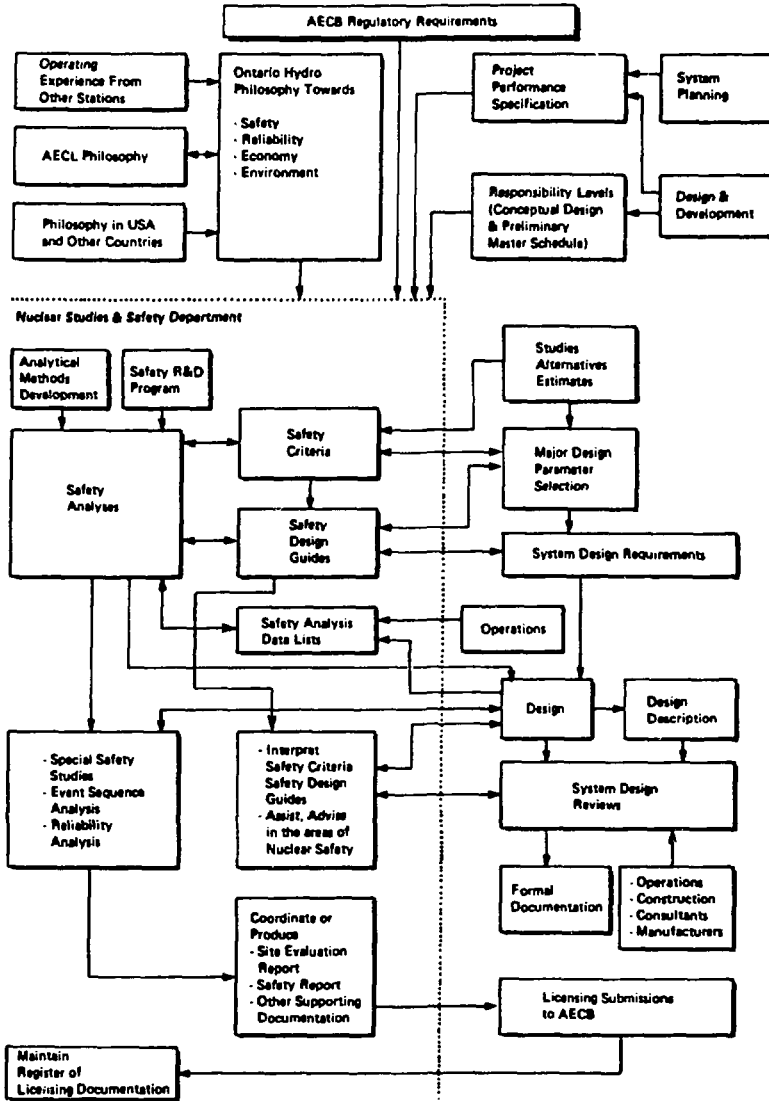
**Other Safety System Assessments****Common-mode failures (done for Darlington NGS)**

- on-site explosion
- off-site explosion
- turbine break-up
- internal fires
- tornado
- airplane crash
- toxic/corrosive gas rail line accident
- earthquake (also done for Pickering B, Bruce B)

**Safety support system assessments (done for Pickering B, Bruce B, and Darlington)**

- loss of service water
- loss of electrical power
- loss of instrument air
- loss of moderator cooling
- loss of moderator water
- loss of end shield cooling
- dual computer failure

Figure II - 4

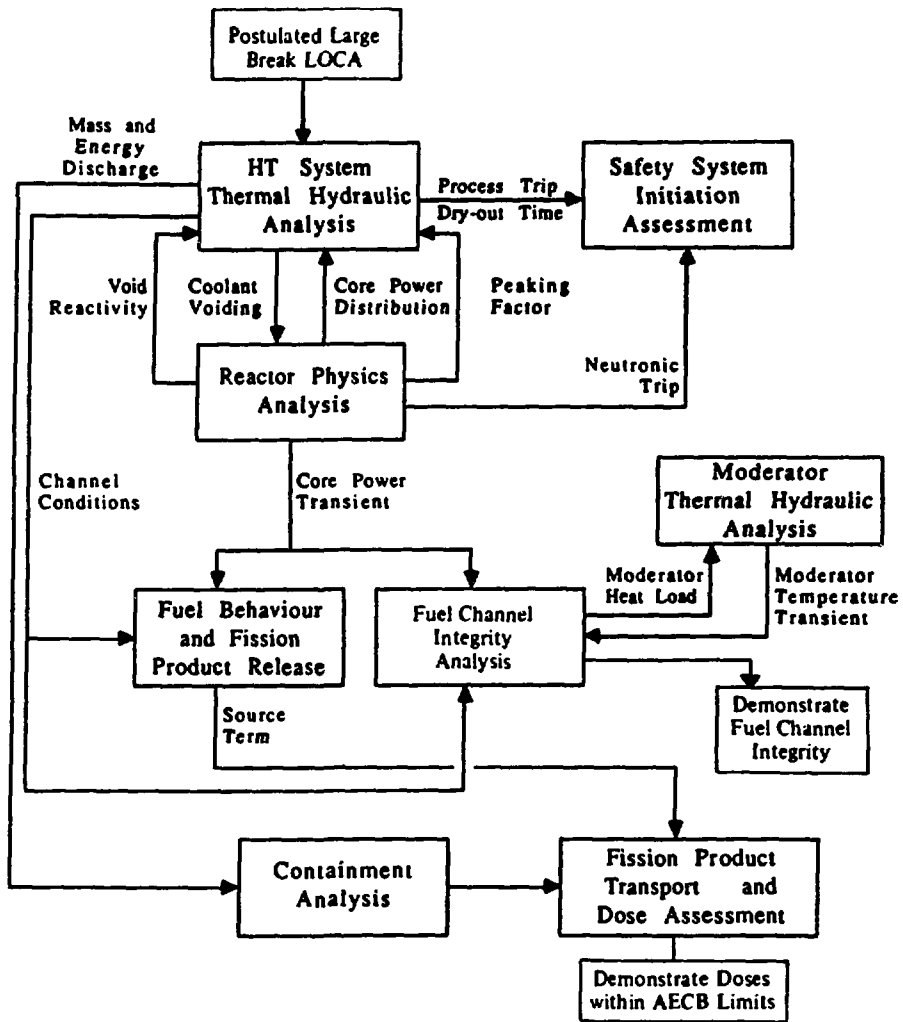


Safety Design Process for Nuclear Generating Stations

Ontario Hydro



Figure II - 5



Basic Modules Used in Loss of Coolant Accident (LOCA) Analysis

Ontario Hydro

Table II-28

**Criteria Used to Demonstrate a Coolable Fuel Geometry  
Exists Following a LOCA**

<u>Analysis</u>	<u>Criterion</u>
Reactor physics and shut-down system assessment	Fuel does not break up
Heat transport system thermal hydraulic assessment	Heat from fuel removed by discharging coolant
Moderator thermal hydraulic assessment	Decay heat removed by the moderator
Fuel behaviour assessment	Fuel does not melt
Fuel channel assessment	Fuel channel remains intact

**Conclusion: coolable fuel geometry, i.e., fuel remains in intact fuel channels.**

steps taken to demonstrate this in the analysis of an accident involving a loss of coolant without injection of emergency coolant.

### **(c) Uncertainties**

**188.** A fundamental aspect of any accident analysis is the recognition and accommodation of uncertainties. The limit-consequence approach accommodates uncertainties in the parameters by systematically over- or underestimating each of them in such a way as to make the final estimate of consequences larger than would be expected for a best estimate. Any other assumptions required in the analysis are similarly made to overestimate consequences (see Table II-29). As a result, this analysis yields an upper estimate for the consequences that might be expected.

**189.** For instance, one can estimate the consequences of the safety systems performing only as well as they are required to by the licensed design. In practice, the regular testing of the shut-down and containment systems shows significantly better performance than assumed.

**190.** The weakness of this approach is that the analysis may miss some important qualitative effect. Thus, one can depend on the limit-consequence approach only if it can be shown satisfactorily that the sequence of events assumed in the analysis is similar to the sequence of events expected.

**191.** For this reason, the AECB has required:

- improvements in the computer programs, so that the accidents can be better modelled;
- more experimental work to verify that critical aspects of the reactor behaviour are appropriately modelled in the analysis; and
- in some instances, more stringent criteria than the normal dose limits, e.g., additional requirements for no significant fuel sheath failures and that a coolable fuel geometry be maintained.

Table II-29

**Accident Analysis Assumptions and their Effect on  
Analysis Estimates**

<u>Analysis</u>	<u>Assumption</u>	<u>Effect</u>
Reactor physics	Power higher than anticipated	More heat in fuel; higher releases
Coolant behaviour	Emergency coolant does not refill channels	More fuel bundles are poorly cooled; higher releases
Fuel behaviour	Channels are grouped into seven power groups; all channels in each group are assumed to have same power as hottest channel in the group	More fuel bundles are poorly cooled; higher releases
Behaviour of radioiodine in containment	Higher fraction of iodine becomes airborne than in experiments (or at TMI)	More iodine released; higher thyroid doses
Filtered air discharge system	Filters perform less well than designed	More iodine released; higher thyroid doses
Emergency planning	No mitigative measures taken following the accident	Higher doses than if protective actions taken
Calculation of individual dose	Weather in first day of release is the most unfavourable	Higher doses than for average weather

192. Research intended to improve confidence in the input for an existing model can reveal unsuspected phenomena, necessitating development of a new model.

193. In the limit-consequence approach, it is assumed that operators take no remedial action within the first 15 min of an accident (in some cases, such as loss of all electrical power and back-up supplies, 30 min is assumed). Action by the operator is expected, in fact, to occur earlier and to be beneficial. When the pressure tube failed at Pickering, prompt intervention by the operator brought the accident under control without the special safety systems being needed.

194. However, it must be recognised that operator intervention can make the situation worse. This occurred in the failure of the pressure tube at Bruce when operator action to detect a very small leak led to a significantly more serious accident. The analysis attempts to allow for this by treating operator intervention as another cause of failure, so that such intervention affects the probability and not the consequences of the accident.

195. However, in the case of certain dual failures, operator action is required to reduce consequences. For example, the operator is assumed to isolate containment at 15 min if the automatic isolation fails.

#### **(d) Accident analysis results**

196. The accident analysis results fall into two categories. In the first, action by the shut-down system prevents damage to the fuel and to the heat transport system (thus leaving intact these two barriers to release of radioactive material). Into this category fall the following accidents: loss of power, loss of pressure control, loss of electrical power, loss of steam generator feedwater, break in a steam pipe, and any of the heat transport pump failures. In some cases, such as a loss of electrical power, back-up supplies are assumed to be made available automatically or are made available through actions of the operator 15 min after the event begins.

197. In accidents of the second category, shut-down system action is insufficient to prevent releases of radioactive material from the fuel and from the heat transport system. The accidents in this category include loss of coolant, flow blockage, end-fitting failure, pressure tube failure, small or large pipe breaks, and heat transport accidents with the fuel handling system.

### 1. Comparison of doses for the different stations

198. All estimated doses are within AECB limits. Table II-30 shows the projected whole-body doses to an individual for analysed accidents at Bruce B. Even when the improvements made to the shut-down systems and ECISs are considered, Pickering A produces the highest projected doses, for both the individual and the population, of all the nuclear generating stations surveyed (Table II-31). The principal reasons for the reduction in estimated doses for reactors built after Pickering A NGS are that:

- ECIS effectiveness has been improved;
- leakage of pressurised air from instruments has been reduced, leading to longer hold-up periods for the containment; and
- population doses are lower at Bruce NGS because of the substantially lower population density.

199. We now examine each of the accidents in the second category in more detail. This review focusses on Pickering A NGS and adopts a historical comparison, comparing the doses estimated in 1971 with those estimated in 1987, to see if any trends are discernible.

### 2. Loss of coolant accident

200. The LOCA is the accident to which the most effort has been devoted, both accident analysis and associated research. In the analysis of LOCAs, it is assumed that one of the large pipes in the heat transport system breaks so that several tonnes of water are discharged each second from the broken pipe into the reactor vault. Depending upon the size and the location of the break, it

Table II-30

**Whole-body Dose to an Individual for Analysed  
Accident Sequences at Bruce B Nuclear Generating Station**

<u>Accident type</u>	<u>Individual whole-body dose (Sv)</u>		
	<u>No impairment</u>	<u>Containment impairment*</u>	<u>ECIS impairment</u>
Fuel handling (machine not attached to reactor)	0.0028	0.027	-
Fuel handling (machine attached to reactor)	0.00037	0.11	0.017
Small LOCA (feeder pipe break)	0	0	0.017
End-fitting failure	0.00019	0.026	0.017
Pressure tube/calandria tube failure	0.0016	0.026	0.017
Large-break LOCA (nominal core)	0.00051	0.043	0.018
Large-break LOCA (pre-equilibrium core)	0.00062	0.049	0.020
AECB limits	0.005	0.25	0.25

\* To simplify the table, the containment impairment that produces the highest dose is used.

Source: Ontario Hydro.

Table II-31

**Estimated Dose for a Large Loss of Coolant Accident for  
Ontario Hydro Nuclear Generating Stations**

<u>NGS</u>	<u>Individual (whole body)</u> <u>(% of AECB limit)*</u>	<u>Population (whole body)</u> <u>(% of AECB limit)</u>
Pickering A	18	21
Bruce A	5.4	0.3
Pickering B	16	15
Bruce B	11	0.3
Darlington	4.2	2.7

\* AECB limits: individual (whole body) - 0.005 Sv; population (whole body) - 100 person-Sv.

Source: Ontario Hydro.



can almost instantly diminish the flow in about one-half of all the fuel channels in one of the coolant loops. The reactor power begins to rise because of the steam created in these channels, and this in turn triggers a reactor trip within 0.5 s. Because the steam in the poorly cooled fuel channels is not able to remove all of the heat generated during the power transient, the fuel begins to heat up. A few of the pressure tubes expand into contact with their calandria tubes. The coolant pressure falls quite rapidly.

201. As the fuel heats up, the fuel sheaths in the more poorly cooled channels begin to fail because of the high temperature of both fuel and sheath and relatively low pressure in the channel. The hot zirconium fuel sheaths begin to react with the steam in the channel, thereby generating heat and hydrogen. Fuel bundles in some of the hottest channels slump downward against the pressure tubes. Pressure tubes sag downward and contact the calandria tubes.

202. Emergency coolant injection is supposed to begin within 30 s (Pickering), but several minutes may be required until the fuel channels are filled with emergency water. Refilling these channels leads to more fission product release because of thermal shock causing cracking of embrittled fuel sheaths.

203. In summary, the following conclusions regarding the consequences of a large LOCA are made:

- (i) Shut-down prevents fuel melting and/or complete break-up of fuel bundles, although some fuel failures do occur.
- (ii) Fuel channels remain intact. Many channels are cooled by residual steam or by the emergency water. The most poorly cooled pressure tubes expand or sag to contact their calandria tubes so that they are cooled by the moderator system.
- (iii) The vacuum building prevents early releases of radioactive material and delays releases until the reactor building pressure again reaches atmospheric pressure.
- (iv) Estimated doses are below AECB limits for a single failure.

### 3. ECIS impairments

204. Analysis is also done for cases in which the ECIS is assumed to be unavailable for the first 15 min. Following that time, an operator is assumed to activate the system so that water can be injected into the heat transport system. The outcome of such an impairment is that more fuel is poorly cooled. The estimated doses for this event are about 100 times larger than if the ECIS functions. According to analysis that has been performed, the containment pressure remains below atmospheric pressure for approximately the same period as with ECIS available. The estimated doses are within the AECB dual failure limits.

### 4. Containment impairments

205. The containment impairments considered include deflated airlock seals, the failure of the ventilation system to turn off, and the failure of some of the pressure relief valves to operate. In all cases, the short-term effect of containment impairments is to increase the amount of radiation released prior to the moment when the containment pressure falls below atmospheric pressure. These impairments also decrease (to 1-2 h) the amount of time required for the containment pressure to increase to atmospheric pressure. The consequence of the impairment is that the fraction of the radioactivity released from containment is increased by about a factor of 30. The operator is assumed to close open ventilation dampers at 15 min following the accident. Doses are lower than the dual failure limits.

### 5. A historical review of the LOCA analysis for Pickering A\*

206. An analysis of a LOCA at Pickering A performed by Ontario Hydro (1971) demonstrated that Pickering A satisfied an AECB requirement that no significant fuel sheath failures would occur should one of the large pipes in the heat transport system fail. The advantage of such a strong requirement is obvious.

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\*The primary source for this historical review up to 1981 is Ontario Hydro 1981a.

If the fuel sheath does not fail, no radioactive material is released and so no harm to the public is expected. Furthermore, the result implies that no fuel channel was damaged by the accident and so there was no danger of many channels failing simultaneously. The analysis assumed that the moderator level was dumped to half its normal level, but the moderator was not needed as a heat sink. This analysis argued that sheath temperatures would remain low because flow in any fuel channel would be stagnant only temporarily in any fuel channel and fuel temperatures would be very low, well below the failure temperature of the sheath. The analysis estimated a maximum sheath temperature prior to injection of emergency coolant of only 837°C, well below the temperature at which the fuel sheath was expected to fail.

207. However, experiments at WNRE in the early 1970s showed that the assumptions made in the original analysis regarding the performance of the ECIS were optimistic. Even for small pipe breaks, the performance of the ECIS could not be shown to prevent fuel damage. Moreover, the performance of the ECIS for large breaks was expected to be inadequate.\*

208. The results of these experiments were translated into estimates of the behaviour of the low-pressure ECIS for Bruce A NGS. Ontario Hydro informed the AECB that the criterion of no significant fuel sheath failure could not be met. Ontario Hydro argued that the dose limits for the single and dual failures set out in the Siting Guide, which could be met, were the more relevant regulatory limits. AECB staff recommended that the power at which the reactors be licensed to operate should be reduced to 63%, in order that no LOCA would cause significant fuel sheath failures. The staff was overruled by the Board, which accepted Ontario Hydro's position regarding regulatory limits and the economic impact of the proposed licence restriction. The licence was granted to operate at 88% of the requested full power, with an upgrade to 100% permitted when a high-pressure ECIS was installed.

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\*For a discussion of the limitations of the old ECISs, see Slec and Rubin 1987, section 4.3.

209. The results of the Bruce A analysis led to a request by the AECB for reanalysis of the Pickering A case. The analysis showed that some fuel sheaths could fail, consistent with the Bruce A analysis but different from the original claim on which the plant was licensed. One observation made in this analysis is that if the shut-down system were altered to allow for a full moderator dump rather than a partial dump, lower releases of radioactive material from the fuel could be expected as a result of a more rapid shut-down. Ontario Hydro argued that no change was needed. The AECB insisted that the change be made.

210. However, both organisations appeared not to have allowed for the phenomenon of pressure tube ballooning. This can occur during the early stages (within 20 s or so) of a LOCA when the pressure tube has heated up and the channel pressure is still high. In such cases, the pressure tube may expand outward to contact the calandria tube, causing the calandria tube to heat up. For the calandria tube to remain intact with the moderator absent, the calandria tube must be cooled by the calandria spray system. There was a concern that such an accident could lead to fuel channel failures and hence higher releases.

211. As a result, in 1981, another analysis of ECIS performance was undertaken (Ontario Hydro 1981b), assuming that the moderator would not be dumped unless shut-down was not achieved by the shut-off rods (the current system at Pickering units 3 and 4). The estimated doses were below the Siting Guide limits, now amounting to slightly less than 1% of the limits.

212. In 1985, another analysis was done of the LOCAs at Pickering A (Ontario Hydro 1985d), assuming that the shut-down system had been upgraded to include 10 additional shut-off rods and that the ECIS had been improved by adding the high-pressure capability of the Pickering B system. In addition, the analysis assumed that the filtered air discharge system, which was not part of the original station design, was available. Still, however, the estimated releases from the fuel were larger than reported in 1981, and the estimated doses were

now a large fraction of the Siting Guide dose limits (typically 10-20% of the limits, but as high as 68%).

213. Quite simply, as new information has come available through research, the estimated releases of radioactive material from the fuel for a LOCA have continued to increase. In the original Pickering A accident analysis, expected doses from a LOCA were many orders of magnitude less than Siting Guide limits. Now they are estimated to be a substantial fraction of those limits. The increase in the estimated doses has been due largely to an improved understanding of the conditions that could occur during a LOCA. This improvement has come about primarily through research, but has been aided by the use of increasingly sophisticated tools of analysis.

214. The important question is whether the future dose estimates could be larger than current ones. Such a change could be expected if fuel channel failures occurred during a large LOCA, contrary to present expectation. At present, it is argued that no fuel channel failures would occur during the LOCA. However, it has been discovered that pressure tube failures could conceivably occur under conditions in which there is a large temperature difference around a pressure tube that is under significant internal pressure (for discussion, see Slee and Rubin 1987, section 4.4). Under these conditions, the pressure tube failure might cause calandria tube failure as well. A model of the pressure tube behaviour under these conditions has been made by Ontario Hydro. The result of the analysis based on this model is that no pressure tube failures would occur during a LOCA (Locke et al. 1987) (Annex II-1). The model is currently being compared with experiments being conducted at WNRE (So et al. 1987).

215. Other important experimental research programmes are being undertaken by AECL's research division, funded by AECL, Ontario Hydro, and other CANDU owners (Ontario Hydro 1987b:4-19). A more complete description of this research work is presented in AECL's (1987a) submission to the ONSR.

## 6. Pressure tube failure

216. Here, unlike the other cases considered in this subsection, there have been two actual events at operating CANDU stations that enable a comparison of analysis with results. In those cases, no radioactive material was released from the station, and there was no need to use the special safety systems to shut down the reactor or to supply make-up coolant.

217. Analyses of pressure tube failures in CANDU reactors have assumed that both the pressure tube and calandria tube have failed. One sees the same pattern as with the LOCA. In the original analysis, expected doses are predicted to be much less than 1% of the limits set out in the Siting Guide. The 1985 analysis (Ontario Hydro 1985d) estimates the maximum dose to be 16% of the limit. The reason for the large difference was the originally unappreciated effect of the low coolant discharge rate to containment in delaying isolation of the containment. The containment is now assumed not to be isolated until the operator activates the system at 15 min. The higher contribution to the dose comes from the release of tritium, rather than iodine, prior to isolation.

218. As discussed earlier, an issue in this area that is still outstanding is the effect of pressure tube/calandria tube failures on adjacent channels. The principal concern is whether the failure of one fuel channel could lead to the failure of another. Ontario Hydro's analysis predicts that such failures would not occur. AECB staff would like Ontario Hydro to undertake full-scale integrated tests to show that this is the case (Annex II-1).

219. A lack of propagation of pressure tube failures is a factor of fundamental importance in limiting the consequences of an accident. It is therefore essential that this claim be verified.

## 7. Flow blockage

220. A blockage of channel flow, however caused, would mean that cooling of the fuel in that channel would be diminished and the fuel would begin to heat up. If the blockage were incomplete, the flow of coolant could still be sufficient to prevent fuel melting, although the fuel might be damaged. If, however, the blockage of flow were complete or nearly complete, the fuel in the channel would begin to melt. The molten fuel could fall onto the pressure tube, causing it to burst. The steam and hydrogen mixture escaping from the broken pressure tube could break the surrounding calandria tube. As a result, fuel and the contents of the channel could be injected into the moderator system.

221. Ontario Hydro has analysed the resulting situation (Lau and Blahnik 1984; Langman et al. 1987), addressing three important issues:

- (i) the ability of either shut-down system to detect the upset conditions, to shut down the reactor, and to keep it shut down. The Ontario Hydro analysis shows that shut-down occurs and the reactor remains subcritical for at least 15 min, at which time operator action to assure shut-down can be credited. A second consideration is the damage that could be caused to shut-off rod guide tubes. The Ontario Hydro analysis concludes that although a number of guide tubes could be dented by the fuel channel failure, the number of rods still available for shut-down at Pickering B, Bruce, and Darlington was still adequate to assure shut-down (the analysis was not done for Pickering A).
- (ii) any damage to adjacent channels. The Ontario Hydro analysis shows that no failures of adjacent channels would occur as a result of this event.
- (iii) possible damage caused by moderator/fuel interaction. This is of particular concern to the AECB. The AECB commissioned a research report (Lee and Knystautas 1987) on this issue and was concerned that the influx of molten fuel to the coolant could lead to a large

amount of steam being generated. The concern was that the resulting "steam explosion" could rupture the calandria vessel and affect the integrity of other fuel channels. Ontario Hydro has argued that a steam explosion is not possible (see Annex II-1). This remains an outstanding issue between the AECB and Ontario Hydro.

## **8. End-fitting failure**

222. The end fittings are castings attached to each end of each fuel channel. For this analysis, the end fitting of one fuel channel is assumed to shear off completely, allowing the ejection of the fuel from the vessel onto the vault floor.

223. The 1971 analysis predicted that this event would produce doses to any individual member of the public of less than 0.05% of the dose limits. The 1985 analysis estimates that the doses to an individual resulting from an end-fitting failure could be as high as 14% of the limit.

224. The difference between the two analyses is principally due to the assumption concerning the isolation of the containment. An end-fitting failure need not lead to a prompt signal to cause containment isolation, thus adding to the total dose. However, the most significant increase occurs in the estimate of the long-term dose, especially from the release of tritium and noble gases.

## **9. Failures of the moderator pressure boundary**

225. Breaks in the moderator piping system are analysed but are not included in the current versions of any of the Safety Reports. Moderator piping systems are located entirely within containment. Nearly all the moderator is at low pressure, hence the rates at which water would leak out would be fairly low. Projected doses from these events are well within AECB limits.

226. One concern is that loss of moderator coolant would diminish its effectiveness as an emergency sink for heat from the fuel channels (see above



discussion on the LOCA). Another concern, still outstanding with the AECB, relates to the radiolysis of the moderator. As the moderator level falls, deuterium and oxygen gas created by radiolysis of the moderator coolant will migrate from the coolant to the cover gas space above the moderator. The concern is that these gases could be ignited by a hot adjuster rod. Apparently, such a phenomenon may have occurred during the NRX accident (Hurst 1953).

227. Ontario Hydro's position on this issue is that the mixture would be non-flammable and that the heat source to ignite a flammable mixture would not be present (see Annex II-1).

## 10. Summary

228. Accident analysis methods have improved as a consequence of improved methods and better understanding of accident phenomena. With these improvements, the dose estimates have increased. The principal significance of such a trend is that it undermines the confidence that the approach in accident analysis has been to consistently overestimate the consequences of accidents.

### (e) Quality assurance

229. Two aspects of quality assurance will be discussed. The first issue is the quality of the analysis itself, as audited by this review. The second issue is the quality of Safety Reports that are based on the analysis.

#### 1. Accident analysis

230. Although a thorough review of the quality of all safety analyses was not undertaken, one review of a particularly relevant aspect of design analysis was commissioned by the ONSR. A private US consultant, Dr. David Diamond, was asked to review the analysis of the reactor-physics computer programs used to calculate the power rise that would occur during a large LOCA. His conclusion (Diamond 1987) was that the analysis was indeed adequate, subject to two important reservations:

- (i) Analysis had not yet been completed to show that the shut-down systems would be effective in terminating the power excursion from all initial power levels and for all unusual reactor conditions. This is the subject of a forthcoming report to the AECB.
- (ii) Dr. Diamond noted that some of the computer programs lacked adequate validation to show that their results were consistent with experiments. Dr. Diamond also noted that documentation for the computer programs was not adequate and recommended that this be improved. Until such time, "the code cannot be considered by this reviewer to have been formally qualified." Apparently, the quality engineering standard proposed by Ontario Hydro (Tang and Noquiera 1986) for all computer programs used in accident analysis has not been fully implemented.

231. Another of our consultants, Dr. K. Serdula, was asked to review the accident analysis of Pickering A and Bruce A. The purpose of the study (Serdula 1987) was to evaluate and assess Ontario Hydro's accident analyses through the identification of key safety parameters used in accident analyses and to review how effectively these had been modelled in LOCAs and in loss of power regulation accidents. The modelling of reactor shut-down, reactivity, heat removal capability, and containment was examined for LOCAs and loss of power regulation accidents analysed for Bruce A and Pickering A. These accidents were then reviewed to check:

- consistency of model assumptions between stations;
- the effect of unusual conditions on the results of accident analyses; and
- possible sources of cross-links between safety system parameters, and between the operator and safety systems, which might lead to an impairment of safety systems.

232. Dr. Serdula made a large number of findings and recommendations. The following is a selection of these findings, along with Ontario Hydro's (1987a, section H1) comments:

- (i) Dr. Serdula notes that the trip setpoints for Pickering A NGS should be adjusted automatically downward during operating transients through automatic switching. Ontario Hydro has found this recommendation to be unacceptably expensive. It is committed to introducing automatic switching to Bruce A NGS (Franklin 1987), however, because the unavailability of the high power (NOP) trip leads to the unavailability of the shut-down system for a slow loss of regulation. Pickering A does not share this vulnerability.
- (ii) Pickering A ion chamber signals are used not only for the high-rate log power trip, but for the high-power trip, and as a power conditioning signal for reactor trip signals that are effective only at low power. Ontario Hydro notes that an annunciation to the operator of a faulty reading would come in and the operator could provide the appropriate correction.
- (iii) There is a concern regarding potential cross-linked failures between the regulating and shut-down systems at Pickering A. This review has already been undertaken by Ontario Hydro.
- (iv) Serdula recommends that analysis be undertaken to show that the performance of safety systems would still be adequate when the reactor is outside its normal operating envelope. Ontario Hydro has replied that the analysed events bound the permitted operating envelope for the station.

233. This final point deserves elaboration. In his paper on the lessons from Chernobyl for the nuclear design engineer, G. Brooks (1987), the Chief Engineer for AECL, has concluded that safety must be addressed from a more fundamental standpoint than is currently done in conventional accident analyses. Ideally, this would mean ensuring that the safety systems were capable for all physically possible conditions in the reactor, and not those limited by the station's operating envelope.

## 2. Safety Reports

234. The quality of Safety Reports is very important because they form the basis upon which the AECB determines whether the station should be licensed. A submission by Oxman et al. (1987) undertook an evaluation of the original Pickering A NGS Safety Report issued in 1971 and the Darlington NGS Safety Report issued in 1987. They devised a scheme for the evaluation of several types of quantitative analyses related to nuclear safety. Table II-32 lists the criteria developed and gives the results of their analyses.

235. Ontario Hydro (1987a, section A1), in its response to this submission, has said that the Oxman review of its reports was cursory, and the results of a detailed evaluation might well be different.

236. Safety Reports are written for a well-informed and specialised technical audience, specifically the AECB staff. As a result, they are very difficult for a more general audience to read and to understand. Furthermore, the organisation and numbering system for the pages, tables, and figures make the reports difficult to understand for even the most experienced AECB staff members. In addition, Safety Reports do not include all the postulated accidents that have been analysed or considered and are relevant to the licence. It would be useful if the relevant documents were at least mentioned and their conclusions summarised, in a manner similar to that in Ontario Hydro's (1987b) submission to the ONSR.

237. In recognition of these shortcomings in Safety Reports, Hydro Quebec and New Brunswick Electrical Power Corporation asked AECL to revise their Safety Reports to make them more easily read by a wider, although principally still technical, audience. The revisions of the Safety Reports were published in 1984 (Hydro Quebec 1984; New Brunswick Electric Power Corporation 1984). Many deficiencies were eliminated in these revisions.

Table II-32

## Criteria for Evaluating Quantitative Analyses

<u>Question</u>	<u>Pickering A Safety Report (1971)</u>	<u>Darlington Safety Report (1987)</u>
1. Was a precise question formulated?	Major	Major
2. Were the alternatives being considered clearly specified (and was there an incremental analysis)?	Major	Yes
3. Were any important alternatives omitted?	Major	Major
4. Were all the important causes of accidents identified?	Major	Major
5. Were all the important consequences of accidents identified?	Minor	Minor
6. Were the best available data used in the analysis?	Major	Major
7. Were the costs and consequences that were identified valued credibly?	Can't tell	Can't tell
8. Were appropriate statistical methods used?	Major	Major
9. Were sensitivity analyses performed?	No	No
10. Were the results presented unbiasedly?	Major	Major
11. Were all the relevant issues of concern addressed in the discussion?	No	No

Key: yes--criterion met,  
 minor--criterion met save minor deficiencies,  
 major--criterion addressed but with major deficiencies,  
 no--criterion not addressed.

Source: Oxman et al. 1987.

## (f) Conclusions

238. All the accidents analysed as part of the accident analysis required in the Safety Report meet dose limits set by the AECB. Pickering A NGS produces the highest estimated doses because of older equipment and higher population density.

239. There has been a trend of increasing predicted doses at Pickering A NGS with time. An improved understanding of the behaviour of fuel and fuel channels has been responsible for the increase in large LOCAs, and an appreciation of a problem with containment isolation has been responsible for increased estimates for the single-channel accidents.

240. Significant uncertainties that could affect the outcome of the analyses of more serious accidents still exist. The most important of these uncertainties are:

- fuel channel integrity during a LOCA without emergency coolant injection;
- propagation of fuel channel failures should a single channel fail;
- molten coolant fuel interaction; and
- hydrogen ignition in the calandria.

The first of these uncertainties is already being investigated with a computer model and supporting research. Owing to the fundamental importance of the issue of fuel channel failure propagation, it is important that the prediction that fuel channel failures do not propagate be verified. The last two issues are currently being debated between the AECB and Ontario Hydro.

241. A review of the reactor-physics computer program used for accident analysis was undertaken by one of our consultants. Although, in his judgement, the existing methods of analysis were adequate, he had to caution that such a finding could not be considered proven until the formal qualification of the

computer programs had been undertaken. He was particularly critical of the lack of adequate documentation in some cases.

**(g) Recommendations**

- (i) More research should be done to confirm that fuel channel failures will not propagate.
- (ii) The quality of Safety Reports and other related documents should be improved. In particular, Safety Reports should contain references to and summaries of all accident analyses performed to obtain an operating licence.
- (iii) In general, all computer programs and analytical techniques used in accident analysis should be formally qualified.
- (iv) Accident analyses should be carried out to cover the range of physically possible conditions in the reactor, rather than be limited to the range set out in the operating policies and principles in the station documentation.

**E. Probabilistic Risk Assessment**

**(a) Introduction**

242. The Probabilistic Risk Assessment (PRA) has become the most comprehensive means of reviewing the safety of the design of a nuclear power station. The PRA is an attempt to identify and quantify:

- how accidents happen, i.e., what sequence of failures can be expected to occur that lead to accidents;
- how frequently accidents are expected to occur; and
- the consequences of such accidents.

243. In light of the many reactors to be considered and the complexity of these systems, it is difficult to do a complete PRA for them all. In their most

detailed form, PRAs require years of effort and millions of dollars to complete. The full reports run to thousands of pages of text and figures.

244. The first PRA for nuclear power plants, the Reactor Safety Study (Rasmussen 1975), was done in the United States in 1975. Since that time, the US NRC, which had commissioned the original study, has placed increased emphasis on the PRA for its determination of acceptable risk (US NRC 1987). Several studies have also been carried out in Europe.

245. Although probabilistic studies of particular safety and safety support systems had been carried out in the past in Canada, only recently have comprehensive risk assessments been carried out for the CANDU reactor. Ontario Hydro has nearly completed one such study at its Darlington NGS. A second, less detailed study has been conducted by AECL and the Dutch organisation KEMA for the CANDU 600-MWe reactor designed by AECL. The ONSR has not undertaken a comprehensive review of these CANDU studies in part because these studies were still ongoing at the time the review was undertaken. However, the preliminary results are referenced in this document, and their implications are discussed.

246. Besides complexity and cost, two other characteristics of the PRA should be mentioned. First, it is a theoretical exercise, especially for the less frequent events. There simply will never be enough experience to determine the frequency of given combinations of failures with a solid statistical basis. In addition, the results are plant-specific: accurate estimates of frequencies and consequences depend on the details of design. This requires a reassessment for each station.

247. The objectives of PRAs have fallen into two categories. First, the PRA is a review of the safety of the design of the plant. Second, the PRA is used as a risk estimate, i.e., an estimate of the risk to society due to the operation of the station.

248. As a design review, the PRA has three clear advantages for safety:



- (i) The review of the station design provided by the PRA is intended to ferret out or identify previously unidentified failure modes. Awareness of such modes can lead to design corrections to reduce the chance of an accident. The PRA, however, is not the sole means of detecting these unidentified failure modes. Operating experience and other forms of design reviews have shown that original designs were defective and needed to be improved.
- (ii) Often, the PRA shows that the unreliability of one or two systems (e.g., the emergency power supplies) is critical in determining the risk. Similarly, when estimates of the reliability of a piece of equipment (e.g., as it ages) are incorporated into the PRA, the effects on the frequency of an accident leading to a release of radioactive material can be estimated. In other words, the PRA helps the designer to focus on the reliability of systems that are most important for preventing or mitigating the effects of accidents.
- (iii) The PRA provides an estimate of consequences of various accident sequences. These estimates enable one to get an appreciation of the importance of the different safety and safety-related systems in preventing or mitigating the consequences of an accident, which would lead to a release of radioactive material from the plant. Furthermore, the PRA can provide some estimate of the importance of passive features of the plant, such as the containment building, in limiting radioactive releases from the plant.

249. The second purpose of the PRA, although one that is not universally claimed, is to attempt to determine actual "risk" to society, where risk is defined as the product:

$$\text{Risk} = \text{estimated frequency of occurrence (per reactor-year)} \times \text{consequence (fatalities or collective radiation dose)}$$

In this approach, one need only sum the risks of all the severe accident sequences contemplated to obtain the total risk from the plant. If the value so

calculated is shown to be very low in comparison with other natural and industrial risks that are tolerated and accepted, it can be argued that the nuclear power plant risk ought to be accepted as well. The use of this definition of risk is also useful in a cost-benefit analysis of the cost of risk reduction measures, where it may be shown that the dollars spent per life saved are usually much larger for nuclear energy (US NRC 1987, Chap. 8) than for other technologies.

250. There are a number of objections that can be made to this position. Before enumerating these objections, it should be noted that the PRA technique is not intended to be the sole definition of acceptable and unacceptable risk of nuclear power plant safety. Safety is a judgement based upon the competence of the design, operating, and regulatory functions within the operating organisation and without. Theoretical estimates of frequencies and of consequences can provide only part of the picture. The ONSR cannot and does not base its conclusions of acceptable risk on such narrow grounds.

251. The first objection to the PRA methods is that the risk definition is too narrow: it makes high-consequence accidents seem insignificant when multiplied by their frequency. A simple example will suffice: Suppose that a chance of a severe accident is 1 in 10 000 reactor-years and, as a consequence, 10 000 people die. The risk is one fatality per year per reactor. The criticism of this definition of the risk is that it trivializes a very serious accident by showing the risk to be rather small. The response to this has been to make the permissible risk smaller for the more serious accident. For example, in the AECB single/dual failure approach, the dual failure, which was supposed to be 1000 times less likely, allowed a 100 times greater dose to the population, making the risk smaller by a factor of 10.

252. A second criticism of these methods is that risk definition does not recognise other types of serious consequences, especially for the largest accidents. Other types of injuries sustained by people (e.g., IICPH 1987), communities (e.g., Alexander 1987; cf. ACNS 1986, discussion on social costs), and biota (Hutchinson and Chouinard 1987) need to be considered; were they

considered, the conclusions about the acceptability of the risk might be different.

253. For example, the impact of a very severe accident, such as occurred at Chernobyl, is much more substantial than the number of deaths involved or the economic damages sustained. In the case of Chernobyl, a permanent disruption of a number of communities within 30 km of the station is a tragedy that cannot be measured in lives lost. The risk estimate as currently defined does not capture these qualities.

254. A third criticism of this method of risk estimation is that frequencies are likely to be underestimated. This is an argument based on a combination of actual experience and derived experience with studies such as the Accident Sequence Precursor programme in the United States (Thompson 1987). The results of this study of the more frequent failures that occur in US nuclear power stations suggest a severe accident frequency larger than estimated in theoretical studies.

255. The reasons for this discrepancy are that things that do go seriously wrong tend to be the things that are unanticipated. The two aspects that are hardest to estimate are the behaviour of people and the interaction between systems during upsets. These two aspects are identified as the weak points of PRA methods in a critical review done by the Nuclear Energy Agency (NEA 1986) of the Organisation for Economic Co-operation and Development (OECD).

256. In light of some of the criticisms outlined above, it can be agreed that the PRA does not represent a complete estimate of the risks from operating a nuclear power station. In fact, one can only consider the PRA-estimated risks as one factor to be considered in judging acceptable safety.

257. This does not, however, detract from its usefulness as a thorough design review. It can uncover possible cross-linked failures that had not been anticipated in the original design. It can be an instructive tool for reducing

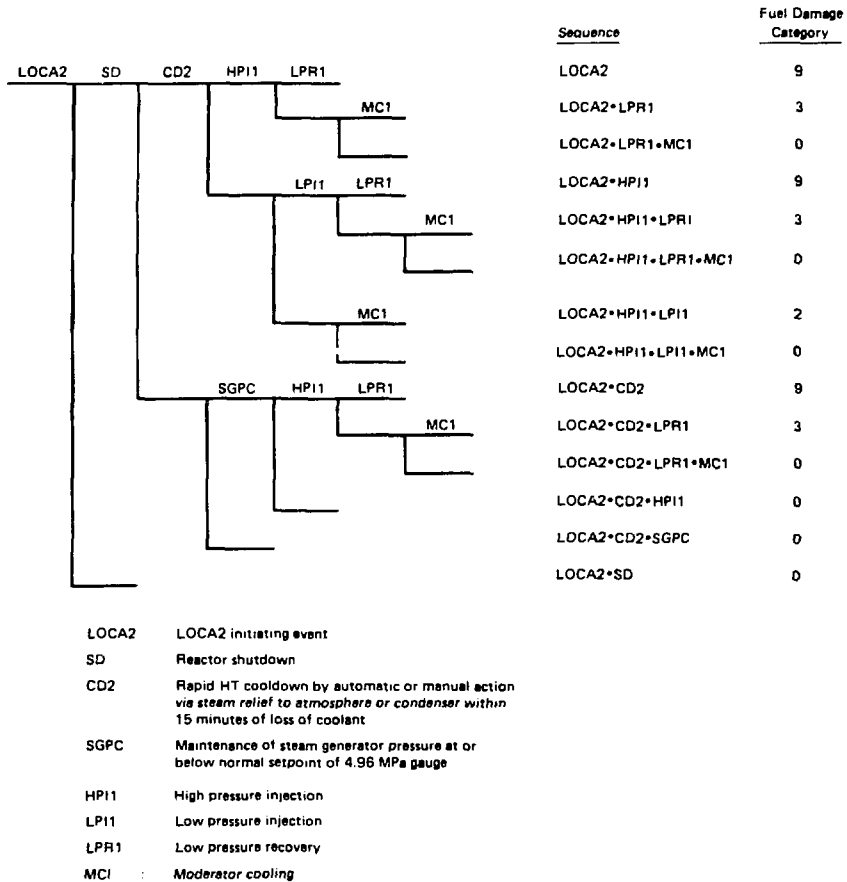
the chance of an accident, great or small. It is for this latter purpose that the PRA is discussed further.

### **(b) Methods**

258. The detailed methodology of a PRA varies from assessment to assessment. The models have become more sophisticated with the growth in computer power, experience, and money spent. There are four distinct phases of calculation involved in a PRA:

- (i) identification of the possible failures or malfunctions. These are known as the initiating events. Initiating events are identified routinely during the original design process and through the licensing requirement for analysing accidents. It should be emphasised that certain initiating events are not considered as part of a PRA. For example, the station is assumed to be capable of coping with external events, such as airplane crashes or tornadoes. The effects of earthquakes are also excluded. These analyses are done separately. Certain other failures, such as that of the steam generator shell, are not considered as possible, as it is felt that a combination of good design and regular in-service inspection will prevent such events.
- (ii) identification of the sequence of consequent or coincident failures that would affect the outcome of the accident. These event sequences are depicted pictorially by event trees. The event trees branch outwards until all possible sequences reach an identifiable end where nothing further can go wrong that would affect the outcome of the sequence of events. A simplified event tree (from King et al. 1987) shows the sequence of events relevant to a large LOCA (Figure II-6). In this figure, the horizontal axis consists of various actions to assure fuel cooling and to prevent or mitigate releases of radioactive material to the environment.
- (iii) calculation of the frequencies of particular sequences of events. Failure frequencies of particular components can be estimated from

Figure II – 6



Event Tree for Loss of Coolant LOCA 2  
Initiating Event

Ontario Hydro

available data for that particular component. By using a fault tree, the failures of individual components of a system and vital support functions to the system (such as electrical power) are combined in order to evaluate the failure rate for an entire system for that particular initiating event. Such failure rates must take into account the potential for cross-linked failures within a system or between systems. Cross-linked failures are by far the most difficult to ferret out and have been significant in a number of unusual reactor incidents. By combining the frequency of the initiating events with the chances of failures of safety and safety support systems, one can obtain an estimate for the frequency of the event.

- (iv) calculation of the consequences of particular sequences of events. The possible end-points for each reactor sequence having been identified, the states of cooling of the fuel and of the containment structure are assessed. The fission product releases from the fuel can be calculated for each sequence and hence the source term, which is the fission product release to the environment. Often, several accident sequences are grouped together into a representative category to simplify calculations.

259. Although regulatory requirements and design goals for acceptable frequencies of failures of process and safety systems have been part of the CANDU safety philosophy from the early days, it is only in the past 10 yr that comprehensive safety evaluations of safety and safety support systems have been undertaken. Work has just been completed on the first CANDU PRAs: Ontario Hydro's Darlington Probabilistic Safety Evaluation (DPSE) and the work of AECL and the Dutch organisation KEMA. The following subsections will be brief descriptions of these studies and their results. More detailed discussion of the more serious accident consequences analysed will be left to Section F.

### **(c) Darlington Probabilistic Safety Evaluation**

260. The Darlington Probabilistic Safety Evaluation (DPSE) is the first full PRA done for the CANDU reactor. It was begun in 1982 and, at the time of

writing, had just been issued to the AECB. A total of 45 person-years has been expended on the study, and the final report is expected to run to over 10 000 pages.

261. Four primary objectives are given for the DPSE (King et al. 1987:151):

- to provide a thorough safety design verification using probabilistic methods, including recommendations for design improvements--improvements are much simpler and cheaper to carry out during construction than following start-up of the station;
- to identify those initiating events and accident sequences that dominate public and economic risk--recommendations for design improvements may follow in order to reduce these risks;
- to aid operating staff in the preparation of operating procedures, especially those to deal with abnormal situations; and
- to provide system reliability estimates that are required as part of the licensing process.

262. Although the DPSE follows the pattern of the general PRA described above, there are a few points worth noting:

- (i) Operating experience was used wherever possible to determine the frequency of initiating events and the reliability of important station services (such as electrical power and cooling water).
- (ii) The state of the reactor fuel was classified into ten different fuel damage categories (FDCs). This greatly simplifies the task of analysing the event consequences by eliminating the need to analyse two events that have very similar outcomes. The fuel damage expected from each of such a large number of events can be classified into a smaller number of categories. For nearly all the cases, the extent of fuel damage can be characterised by the effectiveness of removing heat from the fuel and ultimately from the reactor building to the environment. At one extreme, the most effective heat removal systems are available, and there are no

fission product releases from the fuel. A less effective means of cooling the fuel is the emergency core cooling (ECC) system, which nevertheless keeps the release down to a very small fraction of the core inventory. In some cases, the moderator is relied upon to act as a heat sink. This is less effective, but would appear to prevent widespread melting of the fuel, provided that the moderator cooling system is working as well.

- (iii) Different impairments of containment were analysed to produce different ex-plant release categories (EPRCs).\*
- (iv) Special models were developed to identify and quantify the effects of human error.
- (v) In addition to detailed modelling of the process and safety systems, the modelling of safety support systems was very detailed in order to try to determine the dependence between the two systems.
- (vi) A special procedure was employed to classify severe accidents. Severe accidents are defined in the report as those sequences of events that "lead to physical damage of the core structure such that fuel cooling cannot be assured."\*\* One example of a severe accident is the LOCA with failure of the shut-down systems to operate (see Section F). In the DPSE, any severe accident was classified into FDC 0. It was then shown that the frequency of this class of accident was "acceptably small (so that) no detailed assessment of consequence would be considered necessary to demonstrate design adequacy" (King et al. 1987:152).
- (vii) Economic risk analysis was performed. For all the categories for which consequence analysis was performed, relatively little radioactive material escaped off-site, and so nearly all the economic damage incurred will be to the station itself.\*\*\*

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\*The origin of this unfortunate jargon appears to be American.

\*\*In the main report of the ONSR, "severe accident" is defined as an accident that leads to a large release of radioactive material from the station.

\*\*\*This result is consistent with the work of our consultants Goble and Lonergan.



263. The 10 FDCs are defined as shown in Table II-33, which includes estimates of the total frequencies of all the events in that category. The 10 different FDCs are condensed into four different EPRCs for releases of radioactive material from the plant. Estimated frequencies and doses for each EPRC are shown in Table II-34. The maximum dose sustained by any member of the public is approximately 0.24 Sv, for the analysed cases.\* However, doses arising from EPRC 0 could be significantly larger.

**(d) AECL-KEMA source term report**

264. The Dutch government, in its effort to evaluate different proposals for nuclear reactors, requested that a PRA be done for each of the reactors under consideration to estimate both the probability and the magnitude of release of radionuclides from nuclear generating stations. At the time, numerous PRA studies existed for various light-water reactor (LWR) types. However, no such study existed for the CANDU reactor.

265. AECL, in co-operation with the Dutch organisation KEMA, has done a limited PRA for the CANDU 600-MWe design in order to provide a comparison. The study has been made available to the ONSR in draft form (AECL/KEMA 1987).

266. The calculation of accident frequencies was done using the methodology employed in the Reactor Safety Study in the United States, modified and extended to include events not considered in that study. The calculation of consequences was carried out using AECL analysis methods to estimate releases of radionuclides from the fuel and the pressure within the containment system. The estimates of releases from the containment, however, were calculated using the CORRAL code used in the Reactor Safety Study.

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\*This individual is supposed to be a young child, living at the boundary fence for a three-month period following the accident, and being fed local produce. Doses to older children and to adults would be much lower, and, if evacuation actions were taken, much lower still.

Table II-33

## DPSE Fuel Damage Category Frequencies

<u>FDC</u>	<u>Characteristics</u>	<u>Mean frequency (per reactor-year)</u>	<u>Uncertainty factor</u>
0	Possible loss of core structural integrity	$3.8 \times 10^{-6}$	6
1	Moderator required as a long-term heat sink within 200 s after reactor trip	$2 \times 10^{-6}$	6
2	Moderator required as a long-term heat sink between 200 s and 1 h after reactor trip	$8 \times 10^{-5}$	6
3	Moderator required as a long-term heat sink more than 1 h after reactor trip	$6 \times 10^{-4}$	4
4	Large LOCA, early flow stagnation	$3 \times 10^{-5}$	10
5	Large LOCA, delayed flow stagnation	$1 \times 10^{-4}$	10
6	Single-channel event and containment overpressure	$2 \times 10^{-3}$	10
7	Single-channel event, no containment overpressure	$3 \times 10^{-3}$	5
8	Loss of cooling to irradiated fuel in fuelling machine	$2 \times 10^{-3}$	10
9	LOCA, no significant fuel failures	$2.3 \times 10^{-2}$	3

Source: Ontario Hydro.

Table II-34

## DPSE Ex-plant Release Categories--Frequencies and Doses

<u>EPRC</u>	<u>Mean frequency (per reactor-year)</u>	<u>Mean individual dose (Sv)</u>	<u>Mean population dose (person-Sv)</u>
0	$4.4 \times 10^{-6}$	NE*	NE
1	$9.2 \times 10^{-6}$	$2.4 \times 10^{-1}$	$1.3 \times 10^3$
2	$5.7 \times 10^{-6}$	$5.9 \times 10^{-3}$	$3.2 \times 10^2$
3	$1.7 \times 10^{-5}$	$1.2 \times 10^{-3}$	$2.9 \times 10^1$
4	$1.5 \times 10^{-4}$	$7.0 \times 10^{-5}$	$1.9 \times 10^0$
5	$3.1 \times 10^{-2}$	NE	NE

\* NE = not estimated quantitatively.

Source: Ontario Hydro.

267. As with the DPSE, the various possible outcomes were classified based upon fuel damage. In contrast, however, this assessment did not exclude analysis of the outcome of any event based upon its low frequency. Thus, severe accidents that were excluded from consideration in the DPSE have been analysed. Power runaway accidents (called early core disassembly) and core melting accidents (late core disassembly) were considered, along with the other accidents previously analysed. These accidents are discussed further in Section F.

268. A distinctive aspect of the AECL study is its estimate of "core melt frequency," which was undertaken in order to compare it with the PRAs for LWRs.

269. Thirty-one different initiating events were considered, of which it was found necessary to analyse eight different sequences. The core melting events have been combined into a melt-down frequency for the CANDU 600-MWe reactor of  $4.6 \times 10^{-6}$  per reactor per year.

270. This result compares favourably with US light-water reactors. However, it should be pointed out that the CANDU 600-MWe reactor considered is the latest version and has already undergone significant probabilistic safety assessments as part of its design review. The US reactors with which they are compared were built in an era before systematic probabilistic assessments were undertaken.

#### **(e) Conclusions and recommendations**

271. This concludes the discussion of the PRA and its use in the present CANDU reactor. In it, there has been a glimpse of the more severe accidents that could occur. A discussion of these types of accidents and how they might apply to CANDU reactors is considered in Section F.

272. The recommendations are:

- that probabilistic safety evaluations should be undertaken for operating reactors, beginning with Pickering A--we understand that Ontario Hydro is already committed to performing such an exercise; and
- that an analysis of severe core damage events be undertaken as part of probabilistic safety evaluations.

## **F. Analysis and Consequences of Severe Accidents**

### **(a) Introduction**

273. A severe accident is defined as a nuclear reactor accident in which damage to the reactor core occurs so that fuel cooling cannot be assured. Two types of reactor accidents can be classified as severe: the power runaway accident, such as occurred at Chernobyl, and the core melt-down accident, of which the TMI accident is the closest example.

274. This section considers both the analysis and consequences of severe accidents. Analyses of severe accident consequences to determine the amount of radioactive material that could escape have recently been undertaken by AECL and, at our request, by Ontario Hydro. Results of these analyses are discussed in this section. Included is a description of the analysis, undertaken by Ontario Hydro and Argonne National Laboratory in the United States, of the consequences of a power runaway accident at the Pickering A NGS.

275. Also considered are the consequences of a release of a substantial fraction of the Pickering A NGS inventory on the health and on the property of citizens nearby. Following this, the environmental and social damages that such a large release would cause are also considered.

(b) Power runaway accidents

1. Ontario Hydro/Argonne analysis of a power runaway accident at Pickering A

276. The Chernobyl accident was an example of a power runaway accident, in which reactor power increased until the reactor was destroyed. There is one troubling similarity between the CANDU reactor and the RBMK reactor type used at Chernobyl. The positive void reactivity effect was clearly an important factor in the reactor accident and is shared by the CANDU design. It was natural to want to find out what assurances exist in the design and in the supporting analysis that this effect will not lead to a catastrophic accident in a CANDU reactor in Ontario.

277. As has been pointed out, with the exception of Pickering A, the CANDU reactors in Ontario have two independent shut-down systems. Each system has been shown through analysis to be capable of shutting down the reactor under a wide variety of upset conditions. These systems were considerably better than those at Chernobyl: if a shut-down system with the capability of a CANDU shut-down system had been available to the operator of the Chernobyl reactor, the accident would not have occurred.

278. Nevertheless, Pickering A stands out as the important exception to the rule at currently operating CANDU generating stations. Analysis of an event in which the failure of the shut-down system was assumed to occur following a large pipe break (which would induce the largest possible power excursion) was done as part of the original licensing analysis of the station. At that time, it was argued that this severe accident would not cause a failure of containment. Thus, doses would be small compared with the dual failure release limits. However, it was admitted at the time that the analysis was speculative, and so its conclusions were not as firmly based as those of other analyses.

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\*This analysis is also discussed extensively in the main report.

279. Methods to analyse power excursions have improved greatly since 1971. Considerable work has been undertaken in the United States to analyse the consequences of such excursions in fast breeder reactors. Many nuclear safety researchers world-wide have attempted to analyse the Chernobyl power excursion.

280. In Canada, safety analysis methods have improved greatly. As noted in Section D, the improvements have, in some cases, been accompanied by increases in the estimated doses to members of the public.

281. In light of the improvements in capability, and considering the fact that these improvements have, in some cases, meant that the estimated doses have increased, it was concluded that a reassessment of the consequences of the loss of coolant with loss of shut-down event was desirable. Furthermore, the outcome of this analysis would affect our recommendations on the necessity of a second shut-down system at Pickering A.

282. It was the initial intent of the ONSR to carry out this assessment as independently of Ontario Hydro staff as possible. Staff at the Reactor Safety Division of the Argonne National Laboratory, who had recently completed a similar exercise to simulate the Chernobyl accident, were asked to undertake the reassessment. This arrangement was not found to be feasible in light of the limited time available and the enormous amount of knowledge concerning the CANDU design that would need to be transferred. When this became apparent, a co-operative arrangement was undertaken between Ontario Hydro and Argonne National Laboratory. Argonne undertook its own analysis using its own methods, but relied extensively on Ontario Hydro for input data. Ontario Hydro also did its own analysis and incorporated a detailed assessment of the containment response to such an event.

283. The results of this assessment are reported in an Ontario Hydro (1987b) report to the ONSR. The Argonne analysis was included as an appendix to the Ontario Hydro report and has been published under separate cover.

## 2. Description of the event

284. The large pipe break results in the reduction of the coolant pressure in the reactor system. The coolant at the end of the fuel channels is already at the boiling point, the reduction in pressure and in flow results in a further increase in boiling, and the channels in the loop in which this occurs become voided. This adds positive reactivity (and hence increases power) to the CANDU reactor. The coolant with the broken pipe voids more rapidly, and so the power rise on that side of the reactor is more rapid (Figure II-7). Within 2.2 s, the reactor goes super-prompt-critical, meaning that the prompt neutrons themselves can sustain the increase in power. The difference in power between the two sides has reached 40% in 3 s. By 3.1 s, fuel has begun to melt in the centre of the fuel elements of some of the hottest channels. By 3.3 s, the molten fuel squirts through cracks in the fuel sheaths and hits the pressure tube wall (Figure II-7). The hot fuel droplets cause the pressure tube to strain and rupture at 3.6 s. The hot fuel droplets then impinge on the surrounding calandria tube, causing it to rupture as well.

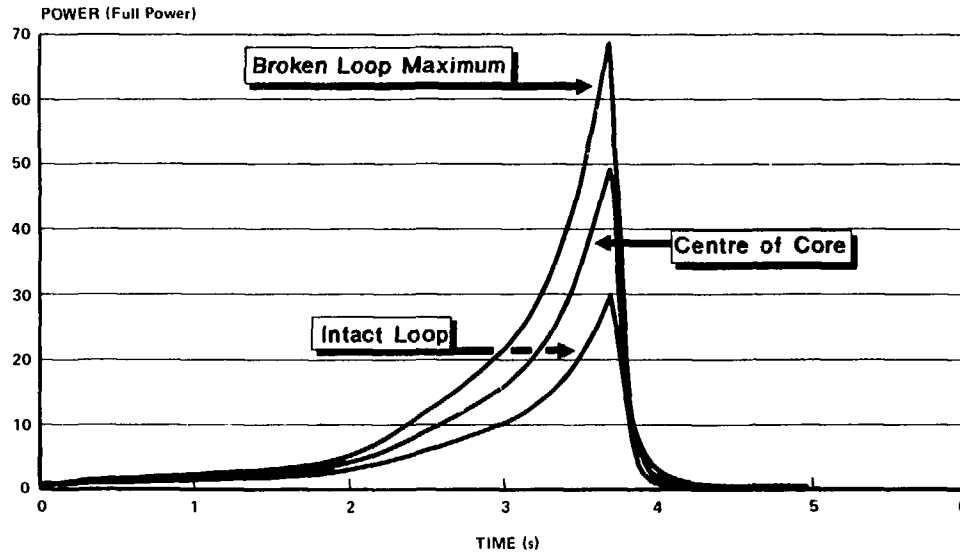
285. As fuel channels fail, steam, molten fuel droplets, and other channel debris are ejected into the moderator. A large steam bubble forms inside the calandria, displacing the moderator and thereby shutting down the reactor by 3.8 s. The pressure rise leads to the opening of pressure relief ducts on the calandria and the failure of the calandria vessel itself.

286. The large quantity of steam and water ejected from the broken pipe and from the broken calandria vessel leads to a rapid pressurisation of the reactor vault. Concrete plugs on the reactivity mechanisms deck are lifted. The pressure in the boiler room is sufficiently high that cracking of the concrete is possible, although these cracks reseal as the pressure falls. Furthermore, once the vacuum building responds to reduce pressure in the reactor building below atmospheric pressure, the discharging coolant will cool the fuel that has fallen into the calandria.



Figure II - 7

# FUEL POWER TRANSIENTS BASE CASE



Ontario Hydro

287. At the end of the early phase of the accident, the predictions suggest that 190 out of 390 channels would have failed, with the contents of the channels dumped into the calandria vessel. The moderator has been expelled through the relief ducts or through the failed weld of the calandria vessel into the calandria vault.

288. About 35% of the fuel has melted during the power excursion. The pressure relief of the vacuum containment system causes the pressure inside the reactor building to fall. Emergency coolant should still be available to inject into the heat transport system. In the long term, emergency coolant would be recovered off the floor of the reactor vault and pumped back into the heat transport system.

### 3. Main conclusions of the analysis

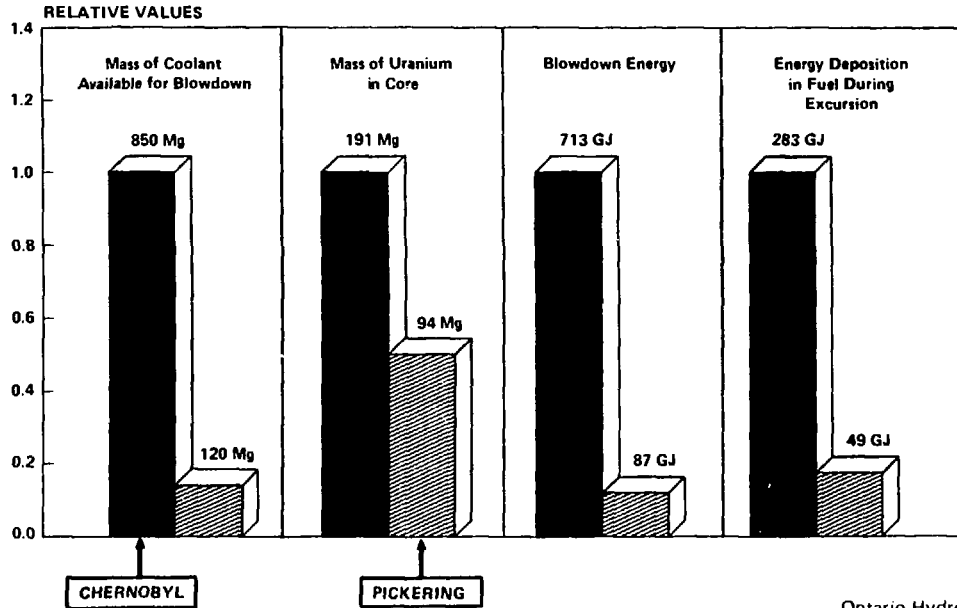
289. The principal conclusion of this analysis is that a power runaway accident will leave the containment intact and the emergency cooling system operable. This implies that the accident would be much less severe than that at Chernobyl. Doses are estimated to be less than the dual failure limits.

290. Three reasons are given for the less severe outcome. Energy release is the first important difference between CANDU and RBMK. CANDU fuel elements would fail at lower energies, transferring less energy to the coolant than at Chernobyl. There is also much less energy available in the hot coolant itself. As a result, the energy released to containment is an order of magnitude lower than the Chernobyl increase (Figure II-8).

291. Second, the CANDU reactor would shut itself down more readily through moderator displacement. The relatively early failure of fuel channels causes the moderator to displace and shuts the reactor down. Graphite moderators are not easily displaced. Furthermore, at Chernobyl, the failure of the first few channels caused the failure of a large number of channels, greatly magnifying the effects of the power excursion.

Figure II - 8

# MASS AND ENERGY COMPARISONS BETWEEN CHERNOBYL AND PICKERING



Ontario Hydro

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292. Third, the containment is a much more robust structure than that at Chernobyl. Although it is debatable whether any containment structure could have withstood the energy released during the Chernobyl accident, it is doubtful whether the Chernobyl structure could have withstood the much less powerful event in a CANDU reactor.

293. The ONSR asked J.T. Rogers to review the results of these assessments.\* His principal conclusion is that the analysis conclusions are firmly based, and that any uncertainties that he has identified would not likely change the conclusion. He does, however, identify a number of omissions in the report. These omissions form the basis for the first four recommendations given at the end of this section.

294. J.A.L. Robertson has also commented on the results of this analysis. He notes that the satisfactory result is critically dependent on the mechanism for fuel sheath failure. He is concerned that the model for the fuel and fuel sheath behaviour has not been properly validated by a comparison of computer estimates with appropriate experiments. He has informed us of another model for fuel behaviour, which has been employed by AECL. In this model, the time required for the fuel sheath to fail may well be delayed, although the amount of delay may not be significant.

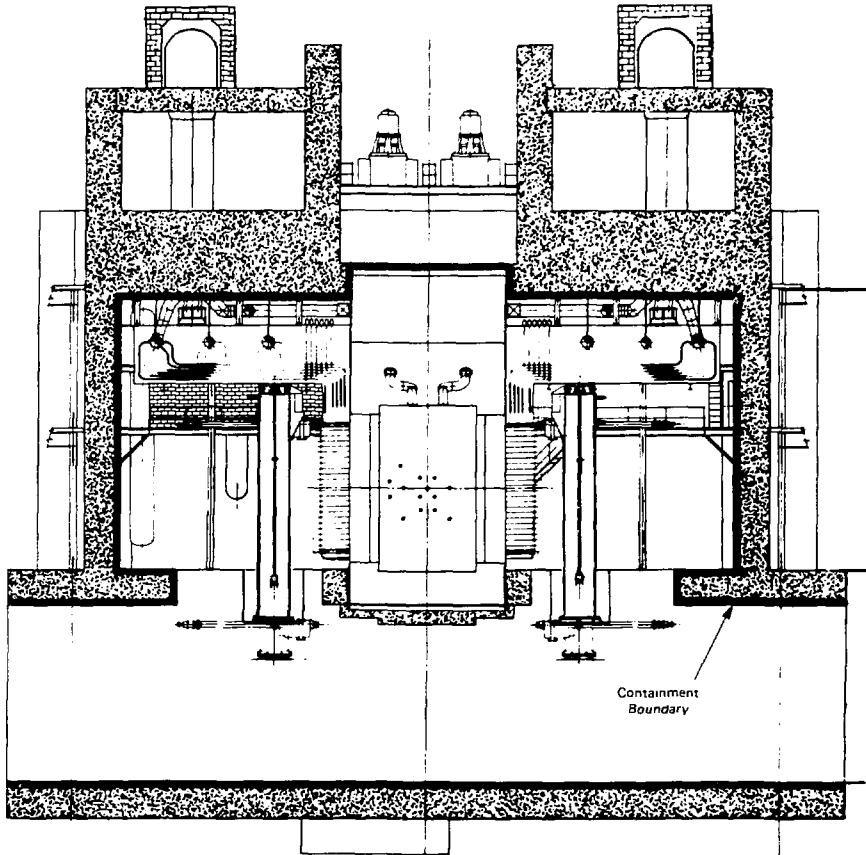
295. The Ontario Hydro analysis was performed for the Pickering A reactors because these reactors each have only a single shut-down system. Subsequent reactors employ two independent systems.

296. The conclusions of this analysis cannot be generalised to Ontario Hydro's other reactors. In particular, because of the substantially different containment designs at Bruce and Darlington (Figure II-9) in which the containment boundary is adjacent to the calandria vessel, the intactness of the containment following such an event is doubtful.

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\*His report has been published with the ONSR report.

FIGURE II-9  
Bruce GS B Reactor Building Containment



Ontario Hydro

297. AECL and KEMA (AECL/KEMA 1987), as part of their source term report, have analysed the consequences of a power excursion. In their analysis, it is assumed that CANDU fuel would fail by fragmenting, and at higher energy than assumed by Ontario Hydro. In addition, the heat generated by the oxidation of zirconium and the combustion of hydrogen is assumed to be added 15 s into the accident. As a consequence, the energy added to the containment is estimated to be three times higher than that estimated by the Ontario Hydro analysis.

298. Furthermore, the AECL analysis assumes that all the fuel has been severely damaged by the power excursion, leading to much higher initial releases of radioactive material from the fuel. All the fuel is assumed to fall to the bottom of the calandria, where it forms a molten mass. This mass is assumed to quickly release much of its fission product content to the containment.

299. When the molten mass melts through the calandria into the shield tank, which in the CANDU 600-MWe reactor is water-filled, a steam explosion is assumed to occur, which results in a very high leakage of radioactive steam from the containment.\*

300. The radically different conclusion of the AECL analysis depends upon the behaviour of the fuel and the fuel channels during the power excursion, the effectiveness of cooling the fuel and fuel fragments following the power excursion, and the possibility of a later pressure excursion, which might subsequently damage the containment structure. On the whole, the Ontario Hydro analysis is far more thorough, and its results are consistent with the Argonne analysis results. However, the comparison with the AECL analysis makes clear the necessary steps that need to be taken to confirm the Ontario Hydro result. These necessary steps are contained in the recommendations at the end of this section.

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\* At Pickering A, the shield tank is not water-filled, and so such a steam explosion is not possible.

**(c) Melt-downs**

301. Even with the reactor shut down, reactor fuel can melt if the heat generated by the fission products within the fuel is not removed. In a LWR, this could occur in the case of a loss of coolant without emergency cooling. This occurred at TMI, with the consequence that approximately 35% of the fuel melted before emergency cooling was restarted.

302. In a CANDU reactor, as has been pointed out in Section D, the LOCA without emergency cooling will not lead to fuel melting, because heat generated by the fuel will be removed through the moderator system. If moderator cooling is unavailable, however, then fuel melting and fuel channel failure can be expected.

303. The sequence of events for such an accident at the Bruce A reactor has been analysed by J.T. Rogers of Carleton University. He predicts that the fuel channels will fail because of the inadequate cooling and leave a molten mass at the bottom of the calandria. He concludes that the capacity of the water-filled shield tank surrounding the calandria is sufficient to prevent this mass from melting through the calandria. The reactor vault pressure would be much higher than atmospheric pressure, although low enough that no damage to the containment structures is expected. Release estimates are not given.

304. AECL has estimated the frequency of such an event to be  $1.2 \times 10^{-6}$  per reactor per year.

305. A loss of heat removal would also occur if all electrical power and/or cooling water were lost to the reactor. Special protection for this type of event is provided by emergency water and emergency power supplies at all reactors except Pickering A and Bruce A. In this case, shut-down would be immediate, but the heat generated within the core would be removed by the boiling of steam generator coolant, heat transport coolant, and moderator coolant. The process is much slower in a CANDU reactor than in a LWR, because of the relatively large amount of coolant per unit power. This boil-off

process of steam generator coolant and heat transport system coolant is expected to take several hours, allowing a great deal of time before any operator action is needed to prevent fission product releases from the containment. Ultimately, if no action is taken, fuel channel failures occur. These failures would be expected to occur prior to fuel melting in the fuel channels. Fuel and fuel channel fragments could end up at the bottom of the moderator. Moderator would be expelled, but the large heat capacity of the shield tank would delay further heat-up of the fuel for a few tens of hours.

**306.** Ontario Hydro has argued that operator action can reasonably be expected to have occurred long before any fission product releases would occur from the containment.

**307.** AECL performed an assessment of this event, assuming that no operator action is ever taken. If no operator action has been taken at 26 h into the accident, the core debris at the bottom of the calandria would finally begin to melt through the calandria and fall into a pool of water beneath it. This gives rise to a steam explosion, resulting in a large increase in pressure and release of fission products. The containment cracks but does not collapse. Steam and radioactive material escape through the small cracks. The steam blown out relieves the high pressure inside the building.

**308.** The mass then melts through the shield tank to the reactor building floor, which is covered with water from the heat transport system. This is presumed to cool the heated mass, and, because of the very low heat being generated by it, the core material is assumed to move no further downward.

**309.** Fission product releases from containment as estimated by AECL are given in Table II-35. Releases are smaller than those predicted by AECL for a power runaway event, but much larger than the dual failure dose limits. The estimated frequency of this event is  $5 \times 10^{-7}$  per reactor per year. The AECL study's predictions of the fractions of material released were based upon the methods used in the US Reactor Safety Study (Rasmussen 1975). AECL has recently published (Howieson et al. 1988) a new analysis based upon the US



Table II-35

Releases of Noble Gases, Iodine, and Cesium from Containment  
from Postulated and Actual Reactor Accidents

<u>Accident</u>	<u>% of core inventory</u>		
	<u>Noble gases</u>	<u>Iodine</u>	<u>Cesium</u>
PWR-2 (Case A) <sup>1</sup>	90	77	50
LOCA/LOECC impaired FADS* (Case B) <sup>1</sup>	2.2	0.005	0.005
AECL source term study (power runaway) <sup>2</sup>	81	30	20
AECL source term study (melt-down) <sup>2</sup>	45	15	9
AECL source term study (melt-down) revised value (1988) <sup>3</sup>	90	0.1	0
Three Mile Island <sup>4</sup>	1.7-25	0.00002	0
Chernobyl <sup>5</sup>	100	40-60	30-50

\* LOECC = loss of emergency core cooling system; FADS = filtered air discharge system.

1. Lonergan et al. 1987.
2. AECL 1987b (used same assumptions as PWR-2).
3. Howieson et al. 1988.
4. These numbers are extracted from the report by Beyea (1984). Noble gas numbers reflect uncertainties, which Beyea considered unresolved. Although some inconsistencies also existed in the iodine releases, Beyea considered the bulk of the data to be consistent with the value given here.
5. As given by Warman (1987). Warman notes that official Soviet estimates underestimate the releases of cesium and iodine by a factor of two to three, because of their neglect of releases outside the Soviet Union. Releases took place over 9 d, with about 10% of this total released during the first day.

NRC Source Term Code Package. Using this package, AECL predicts that iodine releases from containment for the melt-down sequence will be one-tenth of 1% or less of the total iodine inventory, compared with 15% of the iodine released using the Reactor Safety Study methods. A similar reduction in released iodine is expected for the revised analysis for the power runaway event.

310. Although the new, lower values for iodine release are supported by the bulk of experiments performed and by the small releases of iodine at the TMI accident, a recent US NRC (1987) report, intended as an updated version of the original Reactor Safety Study, refuses to endorse these substantial reductions in predicted iodine releases for US reactors because of outstanding uncertainties.

311. The applicability of the AECL results to the Ontario Hydro reactors has not been assessed.

#### **(d) Health effects and economic consequences**

312. The health effects of a reactor accident on the nearby populace are an aspect of nuclear safety to which the most attention has been paid. However, other aspects, such as the environmental, social, and economic damage, should also be considered. Ontario Hydro has asserted for all but the most severe accidents that economic harm to the members of the public would be small. Those cases that could lead to significant economic damage to the public require a combination of failures that must be considered so improbable that these accidents need not be considered.

313. Nevertheless, the ONSR commissioned a study of the off-site economic consequences of a serious reactor accident at the Pickering NGS (Lonergan et al. 1987). The study was carried out by two groups: a group at Clark University, in Worcester, Massachusetts, headed by Dr. R. Goble, was responsible for calculating the dispersion of radioactive material from the reactor in the case of a serious accident, in particular the ground contamination and the health effects that this caused; Dr. S. Lonergan, an economic geo-

grapher at McMaster University, was responsible for evaluating the economic impact of the ground contamination and the health effects.

314. The group at Clark University used the Melcor Accident Consequence Code System (MACCS) computer program to estimate the dispersion of radioactive material following an accident. The program accepts as input the postulated release of radioactive material and assumes this material is dispersed in one particular wind direction over a given time period (in this case 24 h). The program then calculates the dispersion of the material, the health effects caused thereby, and the direct costs incurred by the public in such an event. These direct costs included: health costs, population evacuation and relocation costs, crop and milk disposal costs, decontamination costs, and land area interdiction costs.

315. The computer code MACCS assumes that the wind is constantly blowing in one direction. The study was carried out for three different sectors: one sector where the wind blows towards the region of highest urban land use (sector D); a region where the wind blows towards the section of highest nearby population density (sector H); and a rural sector representative of the most common wind direction (sector L).

316. The main difficulty in applying the MACCS code to a CANDU reactor is the application of an appropriate model of the release of radioactive material. Part of the difficulty faced by the researchers was a lack of studies that had been carried out comparable to the US Reactor Safety Study of what might occur in the most severe accidents. In its place, they have substituted one of the most severe postulated accidents in a US pressurised water reactor (PWR), known as PWR-2 (Case A).

317. The use of this release would appear to be an overestimate of the possible consequences for the releases from a CANDU reactor based upon the studies recently done by AECL and Ontario Hydro (Table II-35). Ontario Hydro has argued that the filtered air discharge system would be used to drastically reduce releases from a severe accident, particularly releases of iodine. Thus, it

argues that a more appropriate release, such as a loss of coolant with a loss of the ECC system (abbreviated LOCA/LOECC) with a degraded fan efficiency is a more appropriate upper limit to consider in the case of a CANDU reactor. The researchers also analysed the consequences of this case (Case B) for the purposes of comparison. This release should be consistent with the LOCA with the failure of the shut-down system considered in a previous section.

318. Table II-35 shows the fraction of fission products assumed to be released from the containment for these two cases, from the AECL studies, from the TMI accident, and from the scenarios that were considered for the release from the Chernobyl accident. It is worthwhile to note that the PWR-2 release is larger than the Chernobyl release, and that the Chernobyl release took place over 9 d, with only about 10% of the total release occurring in the first day.

319. Health effects are summarised in Table II-36. Even without any emergency measures such as evacuation, relocation, and iodine blocking, Case B leads to no early injuries or fatalities to members of the public, and six or fewer cancers. Property damage caused by the accident is less than \$1.4 million in the worst case (Table II-37), and substantially less in most other cases.

320. In Case A, the prompt fatalities from radiation injuries to the public could be anywhere from zero (as at Chernobyl) to a few thousand, depending upon the weather conditions. The latent cancers induced could be as few as a few hundred or as many as 13 000. Timely evacuation of the affected area surrounding the reactor could substantially reduce the predicted health effects.

321. The economic damage (Table II-37) is quite sensitive to the direction of the wind and the weather conditions. Property damage could be as low as \$100 million for relatively favourable weather conditions or as high as \$12 billion. This figure is consistent with Ontario Hydro estimates. Costs would be much higher if a stricter level of decontamination were used.

Table II-36

## Health Effects Resulting From Different Accident Sequences

	<u>Accident sequence</u>	<u>Prompt fatalities</u>	<u>Early injuries</u>	<u>Latent cancers</u>
Case A	TD1	0	5.12	6 700
	TD2	0	2.9	13 600
	TD3	37.5	6011	9 700
	TD4	0	0.143	6 390
	TD5	14.3	2028	9 450
	TH1	1420	355.8	415
	TH2	1340	722.7	1 180
	TH3	3370	5990	1 020
	TH4	0	196.8	570
	TH5	3410	7098	1 490
	TL1	216	189.5	1 080
	TL2	333	4153	2 130
	TL3	569	8780	2 250
	TL4	0	27.15	1 010
	TL5	613	4958	2 200
Case B	CD1	0	0	0.835
	CD2	0	0	3.94
	CD3	0	0	3.80
	CD4	0	0	2.25
	CD5	0	0	4.25
	CH1	0	0	0.610
	CH2	0	0	2.13
	CH3	0	0	2.42
	CH4	0	0	1.24
	CH5	0	0	5.52
	CL1	0	0	0.311
	CL2	0	0	2.54
	CL3	0	0	2.73
	CL4	0	0	1.02
	CL5	0	0	3.17

Source: Lonergan et al. 1987.

Table II-37  
 Components of Property Damage--Case A

<u>Component</u>	<u>Value (expressed in 1981 \$ million Cdn.)</u>				
	<u>TD1</u>	<u>TD2</u>	<u>TD3</u>	<u>TD4</u>	<u>TD5</u>
<b>Decontamination</b>					
Farmland	0	0	0	0	0
Non-farm	284	11 600	8 450	9.52	802
<b>Interdiction</b>					
Farmland	2.52	88.7	39.9	3.11	7.67
Non-farm	0.383	4.06	2 070	0.004	1170
Emergency action	0.134	26.5	53.2	0	40
Milk loss	2.35	0	1.43	2.32	2.36
Crop loss	90.7	0	51.1	89	7.98
Total cost	380.1	11 719.3	10 665.63	103.95	2030.01

Source: Lonergan et al. 1987.

**(e) Social consequences**

322. Several intervenors have criticised the neglect of the non-quantifiable consequences that would be experienced should a large release of radioactive material from a nuclear generating station occur. D. Alexander (1987) and A. Slater\* both argue that such an accident would be very damaging to the way of life of the rural communities near the affected reactor. Alexander points out that a much smaller release of radioactive material at TMI has significantly polarised the nearby communities and undermined the faith in the utility and in the nuclear regulatory body.

323. There is no doubt that a very large release of radioactivity would have serious adverse social impacts. The damage to agriculture and the long-term evacuation of most of the communities within 30 km of Chernobyl have been very traumatic, as Alexander has documented.

324. Whether the social impact of a release similar to the TMI accident would prove as traumatic would depend upon the particular circumstances of the accident. The misleading and contradictory information that was given to the public following the accident certainly undermined any credibility that the utility and the regulator had earned.

**(f) Environmental consequences**

325. A large release of radioactive material would produce serious adverse effects on the ecosystems of Ontario. A brief on the environmental consequences of such accidents was presented to the ONSR by Science for Peace (Hutchinson and Chouinard 1987). The brief discusses the ecosystems that could be affected should a very severe accident occur at one of Ontario Hydro's nuclear generating stations. A very severe accident, such as that hypothesised in Case A above, could cause serious contamination of drinking water and cow's milk and could disrupt agricultural production for years.

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\*Slater represented the Church and Society Committee, London Conference, of the United Church of Canada at the ONSR Workshop.

Although natural ecosystems are not likely to be seriously disrupted, contaminants could congregate in wetlands or river estuaries.

**(g) Recommendations**

**1. Power runaway accidents**

326. The Ontario Hydro/Argonne analysis should provide further information on the behaviour of fission products in containment and their releases from containment, using the best available information on fission product behaviour. In addition, calculations of individual and collective public doses resulting from the accident for both average and worst weather conditions should be made.

327. The present analysis should be extended as needed to demonstrate that long-term cooling of the damaged and undamaged fuel can be maintained and that the repressurisation of the containment does not result in an uncontrolled release of fission products from containment. This assessment should take into consideration possible damage to containment suffered during the early stage of the accident.

328. Results of the analysis for a LOCA with failure to shut down with the reactor at very low power should be documented.

329. Analysis should be undertaken of a loss of regulation accident with the failure of the shut-down system to operate.

330. Ontario Hydro should adequately validate the predictions of its fuel element and fuel channel model for severe power excursions.

331. The Ontario Hydro/Argonne analysis of a LOCA with a failure of the reactor shut-down system predicts that the consequences of this accident would not pose a significantly larger hazard to the public than for other serious accidents. The great expense of adding a second fully independent shut-down system is not justified in this analysis. If, upon the further review and



analysis recommended above, it is shown that the Ontario Hydro and Argonne analyses have substantially underestimated the public hazard of an accident of this type, then upgrading to two independent shut-down systems is fully justified.

332. In any event, it is clear from the work of one of the consultants (Serdula 1987; see Section D) that further improvements to the single shut-down system at Pickering A NGS can and should be made without further delay. At the very least:

- (i) any cross links between the reactor regulating (power control) system and the shut-down system must be eliminated; and
- (ii) additional detectors should be installed to activate the shut-down system when the reactor power (or the rate of reactor power increase) is too great.

333. In addition, consideration should be given to dividing the shut-off rods into two distinct groups (of 11 and 10), each with its own set of detectors. This would provide Pickering A reactors with the protection of two shut-down systems, albeit without the physical separation between systems that exists at other stations.

## 2. Melt-downs

334. The AECB should complete Rogers' analysis of a LOCA with a failure of the ECIS and of the moderator cooling system by estimating the doses to the public in such an event.

335. The probability and consequences of this type of accident at Bruce A and Pickering A should be analysed. The results should be used to consider if an emergency power supply or emergency water supply is needed at Pickering A and Bruce A.

Annex II-1  
**Uncertainties in the Loss of Coolant Accident Analysis\***

1. The analyses of large-break, stagnation LOCAs carried out by Ontario Hydro are based on the limit-consequence approach. In this approach, a number of simplifying but conservative assumptions are used to characterise phenomena that cannot be accurately described by existing computer codes. For example, because of the uncertainties associated with the fuel rewet and channel refill process following a stagnation LOCA, credit for early emergency coolant injection effectiveness is not taken. The AECB has expressed concerns related to the adequacy of the experimental database used to support some of the assumptions and to the fact that this approach does not show the degree to which various safety barriers are protected by the ECIS.

2. Ontario Hydro's position can be summarised as follows:

- (i) The limit-consequence analysis is demonstrably conservative with respect to the expected safety systems performance and thus is suitable for licence applications.
- (ii) The analysis is supported by relevant experimental data; an extensive experimental and theoretical programme was put in place several years ago and is continuing; the AECB Advisory Committee on Nuclear Safety (ACNS) concluded that the analysis is firmly based.

3. Recognising, however, the advantages in terms of both operating limits and licensing basis of a more realistic approach, Ontario Hydro agreed to continue the development and validation of a "best effort" type of methodology. A unit within the Nuclear Studies and Safety Department (NSSD) is dedicated to developing the necessary analytical tools. This unit is supported by other specialists within NSSD, as well as by staff of the AECL-WNRE. Ontario Hydro has submitted two reports so far detailing results of studies performed for the

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\* Attachment to Franklin 1987.

Bruce B NGS reactors. Development of a two-fluid computer code and other relevant models is continuing. An interim reactor calculation for Darlington NGS is scheduled for the end of 1987. The AECB is kept informed of the progress.

**(a) Hydrogen ignition in containment**

4. Analysis of severe accidents, such as loss of coolant with coincident failure of emergency coolant injection, shows that significant amounts of hydrogen can be generated, mainly as a result of zirconium oxidation. The burning and/or detonation of the hydrogen/oxygen mixture could compromise containment integrity or the survival of important post-accident equipment. Recognising this, Ontario Hydro provides a hydrogen ignition system in all the nuclear generating stations. Analysis shows that the system is effective in limiting potential pressure excursions in containment below failure limits.

5. The only outstanding issue with the AECB is related to the possible formation of flammable gas layers or pockets. Ontario Hydro's position is as follows:

- strong gas circulation is expected in a post-LOCA environment precluding the formation of such gas layers or pockets; and
- analysis indicates that the ignition of such pockets, should they occur, does not lead to overpressure greater than that calculated for the case of homogeneously distributed gas mixture.

This analysis is being finalised for submission to the AECB.

**(b) Hydrogen ignition in the calandria**

6. During postulated loss of moderator accidents, the cover gas system depressurises, leading to release of radiolytic deuterium and oxygen from the moderator. At the same time, in-core structures no longer cooled by the moderator may heat up, providing a potential ignition source.

7. Concerns have been expressed that a hydrogen burn may take place within the calandria vessel leading to in-core damage. Considerable analytical and experimental work has been undertaken in the past few years to address this concern. This work is not yet complete. Preliminary results indicate, however, that temperatures high enough for ignition are not achievable. Furthermore, even if they were, the possible calandria gas mixtures are non-flammable.

**(c) Effect of pressure tube/calandria tube failure on adjacent channels**

8. One of the key concerns in an in-core LOCA (i.e., combined failure of pressure and calandria tubes) is that of channel failure propagation. The issue is briefly discussed in section 9 of Ontario Hydro's (1987b) submission to ONSR.

9. To address this concern, Ontario Hydro has carried out a detailed evaluation of various damage mechanisms that may arise. These are:

- transient hydrodynamic loading;
- impingement loads due to fuel projectiles and pipe whip; and
- jet and thrust forces due to coolant discharge from the break.

10. The structural responses of the adjacent channels to these damage mechanisms were evaluated, and it was concluded that adjacent channels do not fail (Frescura 1981a, b; Kundurpi et al. 1984; Muzumdar and Frescura 1987). These analyses were presented to the AECB. An AECB assessment (Jarman 1982) also concluded that channel failure propagation is not credible. However, AECB staff identified a number of relatively minor issues, which were subsequently resolved (Morison 1983). Recent correspondence between the AECB and Ontario Hydro (AECB 1985; Irvine 1987) reiterated the existing position.

11. AECB staff generally agree with the analyses presented by Ontario Hydro, but have requested large-scale fully integrated tests to provide overall confirmation. Ontario Hydro does not believe such tests are necessary, but the Board has agreed with its staff at AECB that such information is necessary.

AECL and Ontario Hydro are looking into options to provide the AECB with the information requested.

**(d) Pressure tube integrity following a large-break LOCA**

12. The issue of pressure tube integrity during large-break LOCAs has been addressed in station-specific analysis. The analysis addresses the integrity both during ballooning and after ballooning contact with the calandria tube. The integrity of the fuel channels after ballooning contact with the calandria tube is demonstrated by ensuring that the calandria tube does not go into sustained dryout, thereby maintaining acceptable pressure tube temperatures.

13. The AECB assessment of the limit-consequence analysis for Pickering NGS and Bruce NGS (Marchildon 1984) indicated that it is in agreement with the general methodology used. The additional information requested by AECB was regarding the accident scenarios for which pressure tube rupture might occur before the pressure tube contacts the calandria tube. This has been presented in Archinoff and Kundurpi (1984), and it has been concluded that only when the pressure tube experiences large temperature gradients is it likely to fail before ballooning contact. The large temperature gradients needed are inconsistent with the large-break LOCA scenario, where rapid voiding of the channel occurs and the pressure decreases relatively rapidly. AECB assessment of the above submission (Power 1985) queried only the failure criterion used, the applicability of the creep equations, and the validation of the computer code used. A reply to all these queries has been given in Penn 1986. Experiments are also currently under way to further support the analytical prediction that large circumferential temperature differences do not occur under such conditions.

14. In summary, Ontario Hydro has developed the methodology for demonstrating fuel channel integrity during the pre- and post-ballooning phases of large LOCAs. All other queries raised by the AECB have been resolved (Penn 1986). Hence, Ontario Hydro does not consider that pressure tube integrity during large LOCAs is an unresolved issue.

**(e) Moderator-fuel interaction following a flow blockage**

15. Ontario Hydro has submitted a detailed assessment of the consequences of a complete flow blockage in a fuel channel (Lau and Blahnik 1984). The AECB identified the issue of fuel-moderator interaction in its assessment report (Marchildon 1986), along with other possible concerns. All these were addressed by Ontario Hydro in Penn 1987 and in a subsequent informal meeting. Ontario Hydro's position on this issue is as follows:

- (i) In the event of a severe flow blockage, the analysis shows that channel failure occurs before the fuel reaches the melting temperature. Ontario Hydro has also analysed a limiting case that ignores all early channel failure mechanisms. In this case, molten fuel would be ejected into the moderator.
- (ii) Despite the presence of the molten material in the moderator, a classical steam explosion will not result because of the following unique circumstances:
  - development and maintenance of a steam bubble at relatively high pressure around the rupture area while the heat transport system depressurises;
  - as a result of the high pressure in the fuel channel, the molten fuel material would be in the form of very fine particles, which are ejected into contact with the moderator at high velocity; this mode of contact between the fuel particles and the moderator is termed "forced" contact as opposed to the "free" contact mode (which is more susceptible to steam explosion) resulting from pouring molten fuel into the liquid under gravity or low-pressure differential; and
  - good heat transfer medium of fog and mist would be present to rapidly cool these fine molten fuel particles, thus removing all their heat content within the steam bubble.
- (iii) The available experimental data under conditions relevant to CANDU situations do not indicate that a steam explosion is possible for the case of channel blockage considered.

**16.** This position has been outlined clearly to the AECB (Brown 1987). We are currently awaiting response from the AECB regarding its position on this topic.

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## Appendix III.1

### An Evaluation of Operational Safety in Ontario Hydro Nuclear Generating Facilities

by  
W.J. Keough

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## A. Introduction

1. The term "operational safety" refers to the set of activities involved in the organisation of tasks and the operation of complex systems of humans and machines in a reliable manner. As the name implies, the major thrust of activities stresses the procedures and actions that are necessary so that facilities are operated in such a way as to assure minimum deviations from expected performance and safe shut-down of facilities in the event of major deviations.
2. The objective of operational safety is to assure that no harm is done to workers, to the public, or to facilities. The objective is therefore both protection of people (humanitarian) and protection against financial loss (environmental damage or destruction of plant facilities). Some corporations look upon operational safety as part of a total loss control concept, which encompasses the total social and economic impact of deviations in operations.
3. In this report, the definition of operational safety has been narrowed to focus investigations on activities that bear directly on the operation of nuclear facilities that can potentially contribute to radiation in the environment.
4. The objective that I stated for this survey of operational safety in Ontario Hydro nuclear operations was "to evaluate the safety culture within Ontario Hydro and assess the effectiveness of systems and procedures used to monitor operations and to respond to deviations in normal operating standards."
5. Annex III.1-1 lists the reports of consultants who were commissioned to investigate operational safety. These reports are cited throughout this appendix.
6. Annex III.1-2 lists reports provided by Ontario Hydro, the Atomic Energy Control Board (AECB), the Nuclear Regulatory Commission of the United States (US NRC), and other agencies that contributed to the information used in this assessment.



## **B. Methodology of the Survey**

7. There is an old management slogan that says "If you can't measure it you can't control it." Techniques for measuring the effectiveness of safe operations in North American industry have been the subject of intensive research and attention in the last decade, as society and governments have insisted on greater protection for workers and the public. Industry has also become acutely aware of the economic impact of unsafe operations in the area of public liability costs associated with damage caused by industrial mishaps. Agencies such as the Industrial Accident Prevention Association, the International Loss Control Institute, Atomic Energy of Canada Limited (AECL), AECB, Institute of Nuclear Power Operations (INPO), and International Atomic Energy Agency (IAEA), to name just a few, have developed very disciplined procedures for auditing and checking operational practices and rating the degree of safety in operations. Some systems such as that of the International Loss Control Institute produce a graded rating on safety.

### **(a) Personal injury measures**

8. In the area of conventional safety, which concerns those activities common to a whole host of industries, including conventional power generation, there is a well-established method for reporting injuries incurred by workers so that safety performance at one location can be compared with others in the same industry group.

9. Personal injury statistics are reported in terms of the number of disabling injuries experienced for a specified number of hours worked. In most industries in Canada and the United States, for instance, this statistic is reported in terms of injuries per 200 000 person-hours worked. The statistic allows a reasonably accurate comparison of the safety performance of one work unit with another and one industry with another. It has proven to be an excellent method of measuring and improving industrial safety. In many cases, it is universally applicable and can be used to compare performance with operations in Europe, Japan, and other industrialized countries.

10. Personal injury statistics are also used to calculate a severity index. This statistic evaluates the type of injury and the days of work lost. In some cases, weighting coefficients are applied to each class of injury so as to accumulate a total factor that represents an index of severity of the accidents that have occurred. It is a valuable measure of the seriousness of incidents occurring in an organisation.

#### **(b) Operational incident measures**

11. Measures of operational incidents other than personal injury are not as universally established as the personal injury statistic. Individual industries have established their own unique systems for measuring operational incidents. The petroleum-refining industry, concerned with the escape of hydrocarbon vapours and liquids and the ignition of these substances, measures performance in terms of the frequency of fires in refineries and the percentage of product lost through leaks and flares.

#### **(c) Nuclear generating industry measures**

12. Operators of nuclear power generating stations have developed measures of performance to monitor progress towards operational excellence. Ontario Hydro has several operational indicators that provide the management of the corporation as well as the public with an insight into the operation of nuclear power generating facilities. Some of the performance measures are required within the terms of the operating licence and are prescribed by the AECB. Others are performance standards developed by Ontario Hydro to ensure excellence in operation and, additionally, to improve overall safety both in plant operations and in the community.

#### **(d) Incident reports**

13. A key document produced by Ontario Hydro to record deviations in operations is the Significant Event Report, or SER. Events that must be classed as significant are clearly and precisely defined. Each event is

documented and investigated, and corrective actions are taken to improve performance. The corporation reports on significant events to the Government of Ontario in an annual report.

**(e) Radioactivity measures**

14. Other important measures of performance are the radiation dose recorded for the work-force and for individuals and the radioactive emissions to the environment. These indices of operation are closely monitored by Ontario Hydro and AECB. They form a valuable source of information for the investigation of operational safety.

**(f) Qualitative measures**

15. In addition to the statistical measures of operating indicators, it is also necessary to make qualitative judgements on the practices and procedures used in operations at Ontario Hydro. Qualitative judgements from well-qualified authorities are valuable measures of operational safety. To this end, the Ontario Nuclear Safety Review (ONSR) recruited investigative personnel (Annex III.1-3) with a variety of background experience in human behaviour, nuclear power generation, operational management in the nuclear industry, and engineering and safety management in the chemical industry. This group brought a well-balanced perspective to the survey of operational safety and allowed the practices from other industries to be compared with those of Ontario Hydro's Nuclear Generation Division.

**(g) Comparing Ontario Hydro with ideal standards**

16. The technique used for the survey of operational safety at Ontario Hydro was based on the premise that a high-performing operating organisation exhibits certain characteristics that account for excellence in operations. The correlation of these characteristics with excellent safety operations has been validated in a variety of North American industries. A summary of the

characteristics of a high-performing operations-oriented organisation is shown in Annex III.1-4.

17. The survey of operational safety performance was divided into two sections. One section dealt exclusively with the human factors involved in generating station operations. This investigation examined the aspects that directly affect the quality of operations, with particular emphasis on the areas of:

- policy and information transfer systems and the effectiveness of communications to and within the operating crew;
- total work environment, including stress factors, work culture, performance evaluation, attitudes, and work schedules;
- training and testing of operations staff, an assessment of the effectiveness of training programmes and facilities; and
- decision and action authority, the degree of understanding of authority at the operating level.

18. The second section of the survey addressed the subject of operational safety practices in the control, maintenance, and management of a generating station. This phase of the investigation concentrated on the areas of:

- safety philosophy of Ontario Hydro: management philosophy, personal injury control, work permit system, safety education, and communications;
- radiation protection: employee protection, community protection, and performance versus standards;
- nuclear emergency plans for on-site action;
- performance control and records: operating reports, internal audits, external audits, and event reports;
- special safety systems: safety systems, emergency procedures, and abnormal incident manuals;
- incident investigation: a review of the SER follow-up procedure; and

- training: an assessment of the need for post-graduate training in a fully qualified operating organisation.

19. The reports of the two investigative teams (Annex III.1-1) can be made available to interested parties.

### **C. Safety Philosophy at Ontario Hydro\***

20. In its statement of corporate direction, Ontario Hydro has made a commitment to "have prime concern for employee health and safety" and "have a prime concern for the safety of the public." This philosophy is imbedded in the attitudes and activities of all staff contacted in the course of the survey.

21. In addition to the general safety philosophy, the Ontario Hydro Nuclear Generation Division has established the philosophy of defence-in-depth. This philosophy has a significant impact on operating staff and fosters a policy that provides parallel yet independent safety systems or procedures that provide a secondary or in some cases tertiary option to control the operating facilities or to contain emissions. The defence-in-depth philosophy permeates the operating organisation and has a very beneficial impact on the attitude of the operating staff.

### **D. Human Factors in Operational Safety**

22. The CANDU nuclear generating plants in Ontario Hydro are equipped with advanced-technology control and emergency systems, with sophisticated logic controls to handle unexpected operating excursions. Notwithstanding the reliability and effectiveness of these systems, it is recognised that human operators form the primary line of operational control and safety.

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\* Compare with the statement of Canadian safety philosophy by J.A.L. Robertson and D.G. Hurst in Appendix IV.

23. The operating crew of a nuclear plant can be likened to an airline crew. The operating crew must be licensed by a government agency; the maintenance crew must keep precise records and use only inspected approved materials; and, under emergency conditions, the crew must respond with precise, well-disciplined reflex actions. The time element for crew reaction is abnormally short in nuclear operations, as well as in aircraft operations.

24. The importance of a well-trained operating staff is highlighted in a large percentage of world-wide operational incident reports. The message from these reports is very clear: a well-trained crew can contain and control a problem situation, whereas a poorly trained crew has the potential to turn a small problem into a calamity.

25. The interviews conducted with the members of the operating staff at the Pickering Nuclear Generating Station (NGS) and a review of documents provided by Ontario Hydro did not identify any serious safety implications, but did identify a number of areas in which improvements in the system should be made (see [a], [b], and [d] in this section).

**(a) Policy and information transfer**

26. The policies of Ontario Hydro, particularly the safety policies, were found to be clearly stated and well understood by the operating crews. There is always a dilemma for an operating crew to integrate the goals of safety and production. The interviewed crews seemed to have an accurate interpretation of both safety and production responsibilities.

**(b) Suggested improvements to instructions**

27. Communications between the operating crews and station management are considered adequate for routine directives on operating objectives, component test schedules, maintenance priorities, and other daily operating instructions. There is a continuous need to evaluate the system of communicating daily instructions from the operating crew of one shift to that of the succeeding

shift, and the effectiveness of the present system must be reviewed by management and operating personnel on a regular basis, with the objective of continually improving the quality of information transfer.

28. One area of communication received general criticism from operating personnel: i.e., feedback from management on suggestions and concerns on operating procedures that are brought up by operating crews. Although there are informal, indirect paths for management comments to reach the operating crews, there is a fairly strong feeling that the formal system of dealing with operating suggestions should be strengthened, and that management should provide a more direct and properly recorded response to suggestions made by the operating crew.

29. Communications within the operating crew for normal operating activities were found to be open and accurate, and routine procedures are well known. In the review of the operating manuals of the station, however, there is a need for a substantial updating of procedures in both form and content to avoid possible confusion and delay in reacting to abnormal operating emergencies.

30. The recommended review of all operating manuals should address the following improvements:

- policies and procedures should exist for operating with an incomplete crew;
- where specific time-related actions are directed, the action with the shortest lead time should be presented first;
- where diagnostic action is required, the specific diagnostic steps should be an integral part of the procedure;
- system-oriented rather than event-oriented approaches should be used for specific procedures; and
- maintenance procedures should specify an independent check of work completion.

**(c) Total work environment**

31. Ontario Hydro nuclear stations are characterised by exceptionally clean, spacious, and functional work areas. Equipment is adequately identified, and control room indicators are accurately labelled with numeric and descriptive labels.

32. Operating crews, comprising technical, operating, maintenance, and AECB representatives, operate in harmony, with common goals and clearly defined areas of responsibility and authority. Each member of the crew is trained to a specific level of competence for personal radiation monitoring and control. This is a practice unique to Ontario Hydro and one that is commended as especially appropriate for a safe operation.

**(d) Shift-work stress**

33. Shift operating crews operate on a 12-h shift schedule. Similar schedules are common in many Canadian industries, including petroleum refining, chemicals, and health care. There is no evidence to condemn the 12-h shift as unsafe, but there is evidence to alert operators to the conditions of mental alertness that can be degraded by long hours of work. Night-shift workers are particularly vulnerable to the stresses of long hours of work because of the human body's natural desire to sleep and the lack of normal day-shift activity to provide stimulation. Overtime work exacerbates the stresses caused by fatigue. The effects of fatigue, particularly the occurrence of drowsiness or sleep in the operating crew, are potential hazards in the nuclear station and need to be addressed for additional action.

**(e) Alcohol and drug policies**

34. Ontario Hydro has a clear and well-communicated policy on prohibiting the use of intoxicants or non-medical drugs on the station premises. No evidence was found that this rule is violated. However, a difficult judgement is required in the interpretation of the condition that may exist for an employee



who enters the plant after consuming a quantity of alcohol. No specific guidelines or rules are set out for the time lapse between drinking alcohol and reporting for work. Most airlines have a rule that pilots cannot fly if they have consumed alcohol within 8 h of command--a rule that is commonly referred to as "8 hours bottle to throttle." Bus drivers have similar regulations. A recent study at Stanford Medical School on US Navy pilots found significant impairment of performance at an interval of 14 h after alcohol consumption. The body of evidence on the effects of alcohol on the central nervous system indicates that Ontario Hydro should address the subject of alcohol consumption and clarify the rules and guidelines for reporting for work.

#### **(f) Recommendation**

35. It is suggested that Ontario Hydro management and employee union management should develop a work culture that does not allow alcohol consumption at least 8 h prior to reporting for work.

#### **E. Training and Evaluation of Operations Staff**

36. Training and the demonstration of learned skills are elements of safe operations that must be placed at the top of the hierarchy of safety prerequisites. In Ontario Hydro, the direction for the training programme comes from the joint effort of the corporation and AECB. Graduates of the training programme must meet the standards and pass exams set by AECB and, in addition, meet the requirements set by Ontario Hydro. Training programmes are presented for shift supervisors, shift operating supervisors, first operators, second operators, assistant operators, mechanics, maintenance technicians, and control technicians. Training curricula include classroom theory classes, laboratory classes, simulator operations classes, and on-the-job experience under the direction of qualified supervision. Ontario Hydro operates two facilities as Nuclear Training Centres at Pickering and Bruce stations. The centres have the appearance of a technical college, complete with control simulators that duplicate the actual station control rooms.

**(a) Licensed positions**

37. The shift supervisor, shift operating supervisor, and first operator must pass the requirements and be licensed by AECB. Each shift must have these qualified personnel in charge.

38. Shift supervisors often have a degree in science or engineering with at least 7 yr of experience in nuclear operations. First operators generally have some post-grade 13 education and 5 yr of experience at the Assistant and second operator levels.

39. Mechanics and control technicians have similar training schedules and experience levels to reach the fully qualified status.

40. The training facilities, course content, and practical experience afforded students can be classed as excellent. The qualification standards set by AECB and Ontario Hydro produce operators and supervisors with exceptional knowledge of the control and operation of the system.

**(b) Post-licence training**

41. After an operator or supervisor is licensed by AECB, there is no formal requirement for validating the licence on a regular basis. There is an ongoing retraining schedule to keep skills at an acceptable level, but this programme lacks the precision and dedication required in the original training.

**(c) Training recommendations**

42. It is recommended that the simulator training programme for trainee operators be expanded to provide more training and evaluation of procedures in the handling of abnormal incidents. The instructional time on the simulator and the operator's opportunities to practise on the simulator should be significantly increased.

43. The refresher training programme for operators and shift supervisors requires immediate upgrading. A formal programme for training licensed operators and supervisors on a regular basis should be prepared and implemented. The programme should include a method of evaluating the results of the training. Evaluations of the training should be used to assure that the competency of the operating crew is maintained.

#### **F. Conventional Safety Performance**

44. "Conventional safety" in this report means the safety performance associated with general industry in North America. Reporting systems for this performance are well defined, and therefore it is a simple matter to match the safety performance of one operation with that of another. The common method of measuring safety performance is to record all injuries sustained by the workforce and categorise the injuries as minor (first aid), serious (medical attention but no time off work), or severe (time off work, permanent or partial disability or fatality).

45. The key statistic monitored by industries for performance evaluation is the frequency of severe (time off work) injuries. The general industry standard expresses this performance in terms of injuries for every 200 000 person-hours worked. Ontario Hydro reports performance as injuries per million person-hours.

46. In order to develop a broader statistical base for the purposes of safety control, it is also common practice to record all injuries (minor, serious, and severe) and report a recordable injury rate for person-hours worked.

##### **(a) Ontario Hydro performance compared with that of the Canadian chemical industry**

47. A comparison of Ontario Hydro performance with those of the three best-performing chemical companies in Canada is shown in Tables III.1-1 and III.1-2. The injury rate at Ontario Hydro nuclear operations is much higher than those

Table III.1-1

## Severe or Disabling Injury Rate per 200 000 Person-hours

<u>Industry</u>	<u>1984</u>	<u>1985</u>	<u>1986</u>
Ontario Hydro Nuclear Generation Division	0.26	0.60	0.66
Dow Chemical	0.05	0.00	0.00
Dupont	0.00	0.06	0.11
Esso Chemical	0.06	0.25	0.00

Source: Ontario Hydro; Canadian Chemical Producers Association (CCPA).

Table III.1-2

## Recordable Injuries per 200 000 Person-hours

<u>Industry</u>	<u>1984</u>	<u>1985</u>	<u>1986</u>
Ontario Hydro Nuclear Generation Division		4.00	3.94
Dow Chemical	1.58	1.58	1.89
Dupont	0.72	0.59	1.89
Esso Chemical	0.88	0.63	0.34

Source: Ontario Hydro; CCPA.

at the top three chemical companies, even though the Ontario Hydro work facilities and skills training programmes for employees are clearly superior to those of most industries, and the disciplined approach to safety in operations is a part of the work culture.

48. Because there is frequently a direct correlation between personal injury statistics and the safe performance of tasks, there is also a correlation between personal injury statistics and general compliance with operating standards.

**(b) Recommended changes in conventional safety programme**

49. It is recommended that the Ontario Hydro Nuclear Generation Division establish new, lower targets for personal injuries and evaluate programmes used in other industries. Some specific recommendations are:

- to use the same reporting standard as the chemical and petroleum industries and compare performance with these industries;
- to use the all-injury reporting system for early-warning control;
- to institute a 24-h safety concept for safety at work, at home, and on the road; and
- to educate the work-force in the meaning of statistical targets and how they can be used to motivate high performance.

**G. Radiation Protection**

50. Ontario Hydro has developed what appears to be a unique system of radiation monitoring, in which each worker is trained and qualified to handle personal radiation protection. The results of this programme are encouraging, with the average individual dose being below 10% of the international standard for occupational exposure. At the newer plants such as Bruce B, where there is very little need to enter the containment structures, the radiation dose is extremely low for the average of the work-force and for the highest individual exposure.

**(a) On-site radiation measurements**

51. Surveys of station radiological conditions are performed routinely and reported to the Plant Manager. A detailed summary of the radiological measurements for specifically identified areas of the station is included in the Quarterly Technical Report of station operations. Any exposure to individuals in excess of regulatory limits is reported, and appropriate protective actions are taken.

**(b) Off-site radiation measurements**

52. The on-site radiation control programmes are intended to protect work staff and also to prevent the release of radioactive materials to the off-site environment. The emissions of radionuclides to the environment are monitored by off-site monitoring stations, and results are cross-checked by calculations of on-site emission data. The data reviewed during the course of the safety survey indicated that radiation releases to the environment from current operations are less than 1% (derived emission limit) of the AECB allowable release limits for individual radionuclides, and less than 20% of the limit recommended by the International Commission on Radiological Protection (ICRP) (see Appendix III.2).

53. The data reviewed on radiation control and experience show steady progress in the year-to-year reduction of radiological emissions and dramatic reductions in the radiation exposures in the operation of the newest facilities.

**(c) Recommended actions on radiation control**

54. In the course of the survey, it was noted that some complacency appears in the attitudes and work practices associated with radiation protection. There was evidence of a relaxed policy in requiring full radiation qualification for maintenance workers and safety staff and noticeable degradation of the standards for maintaining "rubber areas" (for the storage and handling of protective clothing) at one station (Pickering A).

55. In order to protect the excellent record on radiation control, Ontario Hydro should be alert to any signs of the lowering of standards and take corrective action. Specifically:

- radiation qualifications for all operating staff should be brought up to standard immediately; and
- "rubber areas" at Pickering should be upgraded to meet standards; regulations should be reviewed and training instituted where appropriate.

## **H. Operating Practices and Control Functions**

### **(a) Operating authority and manuals**

56. Operating crews have excellent understanding of the scope of authority for each of the operating positions, and there was no evidence that this authority was being neglected or transgressed. The discipline of role authority was clearly superior to that in some other industries known to the surveyors.

57. Extensive operating manuals are provided and used for normal operations. In the course of the survey of safe operations, the surveyors referred to the station operating manuals and the maintenance procedure and testing manuals and gave particular attention to the Abnormal Incident Manuals (AIMs).

### **(b) Control and safety systems**

58. Control and safety systems for the operation of nuclear facilities are separated in Ontario Hydro nuclear stations so that safety systems are not operated on common control signals from the operating monitors. This separation of control and safety functions is not common in the nuclear industry and is part of Ontario Hydro's defence-in-depth philosophy.

59. Both control and safety systems have duplicate or triplicate back-up systems for each control loop. Extensive testing procedures are in place to assure that each safety system control loop is in operable condition; if the test does not meet certain criteria, the control loop is corrected and the problem is reported, analysed, and used to predict the overall state of readiness of the safety system.

**(c) Management reports on station condition**

60. Monthly and quarterly reports are prepared to document the operating parameters of the station and report on the adherence to licence standards and to Ontario Hydro standards.

**(d) Compliance with procedures**

**1. Internal audits**

61. Internal audits of operations are carried out by a number of groups in Ontario Hydro, and the Radioactivity Management and Environmental Protection Department (RMEP) has a section that specialises in developing and organising audit techniques to monitor plant practices. Some of the internal audits performed on a regular schedule are:

- quality assurance audits;
- conventional safety audits;
- joint Safety and Health Committee audits;
- RMEP audits;
- INPO-style audits, conducted by RMEP; and
- Personnel Assessment Audits (morale and attitudes).

**2. External audits**

62. An external audit of operations is conducted by AECB on a continuing basis by representatives who are resident at the station. The AECB sets



standards of operation, reliability, and safety that must be met to allow continuing operation of a nuclear station, and these standards are closely monitored.

63. In addition, the operating practices at nuclear stations may be audited by a variety of external organisations, including the Ontario ministries of Labour, the Solicitor General, and Commercial and Consumer Relations.

**(e) Recommendations**

64. The survey of practices and procedures did not identify any deficiencies that would compromise the safe operation of the nuclear generating facilities. The reviewers were impressed with the knowledge and discipline of operating personnel, the multiple back-up of safety systems, and the comprehensive audit and testing procedures used on the normal operating and safety systems.

65. There are, however, a few areas in which improvements in practices and facilities should be instituted to reduce the potential for errors in operations. These areas for improvement are as follows:

- (i) Operating Manuals: As previously mentioned, the format of operating manuals should be changed to include diagnostic procedures in the body of procedures and to present time-sequenced activities in chronological order. The operating procedures in many cases are also modified by Operating Memos, a separately filed temporary instruction that modifies a procedure. There are currently too many temporary instructions outstanding, and this complicates the work-load of shift supervisors and operators as well as increases the opportunity for operating errors. Procedures should be revised to incorporate the Operating Memo alterations.
- (ii) Operating Crew: In some interviews, the operating crews expressed a reluctance to disclose any operating errors that occurred if there had been no subsequent event that would show the error. It would be most beneficial to have these near-miss events recorded and

analysed by the operating crews, and Ontario Hydro should develop a system for encouraging this form of crew self-audit.

- (iii) Shift Supervisor Term of Office: The operating effectiveness of the control team would be improved by extending the term of office of the shift supervisor so that long-term multi-year programmes could be instituted and directed by the shift supervisor. At present, the turnover rate of the shift supervisor position is excessive and inhibits the implementation of long-range plans.
- (iv) Audits: Ontario Hydro's plan to set up an INPO-style audit function and peer-review teams is strongly supported. It is also recommended that Operational Safety Review Team (OSART) style external audits be continued.

## **I. Maintenance**

### **(a) Maintenance organisational structure**

66. Maintenance functions are divided among a number of departments at the station level and the head office. The Production Department at the station is responsible for directing the maintenance work. Job planning is organised by a clerical section in the Planning Organization. Engineering support is provided by the Technical Department. Quality control (e.g., inspection and radiographic examination) is provided by an on-site representative from Central Nuclear Services at head office. Analysis of maintenance trends is conducted by the Equipment Reliability Department of Central Nuclear Services at head office.

67. The maintenance facilities and organisation at Bruce B site were the only ones examined in this survey.

### **(b) Maintenance performance**

68. At the job performance level, the maintenance function is directed by a mechanical maintenance supervisor and a control maintenance supervisor. Each

supervisor directs a group of shift supervisors. Maintenance crews work all shifts. Preventive maintenance is planned by the Technical Department, and most of this is performed by the day shift.

69. Maintenance supervisors maintain very extensive drawings and records of plant equipment and are responsible for keeping these records updated with physical changes that have taken place.

70. Maintenance shop facilities and records-keeping areas are outstanding in their degree of organisation, cleanliness, and equipment.

71. A good example of Ontario Hydro's defence-in-depth philosophy is the facility provided for the maintenance of heat transfer pump seals and the design of the seal system itself. Seal repair facilities are housed in a clean room where precision equipment can be used to inspect all new parts before they are accepted for installation or to repair parts that do not meet tolerances. The design of the seal provides for dual seals with continuous monitoring for any sign of leakage from the primary seal. The improvement in the reliability of pump seals has been impressive.

**(c) "Jumpers"—temporary changes**

72. Temporary repairs or changes to equipment are identified by tagging the equipment and making a change to records and drawings. These temporary alterations are referred to in the nuclear industry as "jumpers." A jumper is resolved by returning the equipment to its original condition with a permanent repair or by making the jumper alteration a permanent configuration. The latter case requires extensive engineering analysis and approval. The survey of maintenance indicated an unusual backlog of jumpers in effect.

**(d) Work backlog**

73. It is normal to have a backlog of maintenance jobs, which are arranged in order of priority and worked into the job planning process. However, in the

survey, it was judged that the maintenance backlog was excessively high, and this was confirmed by station management.

**(e) Recommendations for improvements in maintenance**

74. Although the quality of personnel, the quality of work, and the facilities provided for maintenance were all judged to be superior, there is considerable room for improvement in the planning, control, and direction of work, and this would indirectly improve the safety of operations.

75. The following recommendations are suggested:

- strengthen the administration of the maintenance function by appointing an overall superintendent with direct access to dedicated engineering support; and
- organise a task force to reduce jumpers and work backlog.

**Annex III.1-1**  
**Operational Safety Reports Commissioned by ONSR**

1. **An Assessment of Human Factors Issues in the Safety of Ontario Hydro's Nuclear Generating Stations**

**Prepared by: Human Factors North, Inc.**  
**118 Baldwin Street**  
**Toronto, Ontario M5T 1L6**  
**14 September 1987**

2. **Safety Evaluations for Ontario Nuclear Safety Review**

**Prepared by: K.V. Biron (Biron Engineering Ltd.)**  
**J.W. Richmond (James Richmond and Associates)**  
**September 1987**

## Annex III.1-2

**Reports Used in Preparing the Evaluation  
of Operational Safety**

1. Human Factors North, Inc.  
An Assessment of Human Factors Issues in the Safety of Ontario  
Hydro's Nuclear Generating Stations  
14 September 1987
2. K.V. Biron and J.W. Richmond  
Safety Evaluations for Ontario Nuclear Safety Review  
September 1987
3. CUPE Local 1000  
A Brief to the Ontario Nuclear Safety Review  
8 September 1987
4. US Nuclear Regulatory Commission (NRC)  
1985 Annual Report
5. Ontario Hydro  
Bruce Nuclear Generating Station "B"  
Quarterly Report, Fourth Quarter 1986
6. Ontario Hydro  
Pickering Nuclear Generating Station "A"  
Quarterly Technical Report, Fourth Quarter 1986
7. Ontario Hydro  
CANDU Operating Experience  
Report No. NGD-9 (1986)
8. Ontario Hydro  
A Submission to the Ontario Nuclear Safety Review  
August 1987
9. Ontario Hydro, Nuclear Generation Division (NGD)  
Nuclear-Electric Generation Objectives  
Report No. NGD-6 R-1  
March 1982
10. Ontario Hydro, Nuclear Generation Division (NGD)  
Performance Review 1986  
Report No. NGD-11-86
11. Atomic Energy of Canada Limited (AECL)  
A Submission to the Ontario Nuclear Safety Review  
August 1987

12. **Atomic Energy Control Board (AECB)**  
**Staff Operational Procedures**  
**Part 4: Operational Nuclear Generating Stations**  
**January 1986**

## Annex III.1-3

**Summary of Experience of Personnel Used in the  
Assessment of Safe Operations**

1. Biron, K.V., P.Eng., President, Biron Engineering Ltd. Over 20 yr of experience in nuclear engineering, nuclear physics, and advanced control systems. Twice licensed by AECB as a shift supervisor on CANDU reactors. International nuclear experience with IAEA and Karachi Power Station.
2. Feher, M.P., P.Eng., Human Factors North, Inc. Industrial engineer with experience in control room design and layout for Maine Yankee Atomic Power Co. as well as experience on CANDU system control concepts.
3. McGeachy, J.B., P.Eng., Professor of Mechanical Engineering, Queen's University. Extensive experience in the design and management of nuclear generating facilities in the United States. Involved in the assessment of training and evaluation operators for CANDU reactors.
4. Meneley, D.A., Professor of Nuclear Engineering, University of New Brunswick. Extensive background in the design of CANDU nuclear reactor control systems. Directly involved with the training curricula and texts for training operating staffs for CANDU nuclear generating stations.
5. Richmond, J.W., P.Eng., President, James Richmond and Associates. Over 35 yr of experience in the design, operation, and management of chemical plants in Canada and the United States. Performed as manager of Safety, Security, and Environment Protection for one of Canada's leading chemical producers.
6. Senders, J.W., Chairman, Human Factors North, Inc. Professor emeritus of Department of Industrial Engineering, University of Toronto, and Research Professor of Engineering and Psychology, University of Maine. Over 30 yr of experience in the fields of engineering and psychology in a variety of advanced system controls, including the nuclear industry in Canada and internationally.
7. Smiley, A., President, Human Factors North, Inc.; President, Humachine Inc. Extensive experience in control systems design and control room layout for the nuclear and other industries.



## Annex III.1-4

**Safety Characteristics of a High-performance Organisation**

1. Safety philosophy and communication
  - formal statement
  - wide exposure and acceptance
  - actions and results are consistent with philosophy
  - corporate safety policy is understood and repeatable by all employees
  
2. Organisation for safe operations
  - supportive co-operation by maintenance, operating, technical, and construction functions
  - internal co-ordinating committees are active
  - system for assimilating successful techniques used by external organisations
  
3. Training and testing
  - standards for entry-level employees
  - training facilities match plant equipment
  - standards for qualified positions and promotion
  - standards for contractors and suppliers
  - active participation of employee organisations
  - evaluation of system by external authority
  
4. Measurement of safety in operations
  - measurable indicators
  - standard measurements that allow comparison with other similar operations
  - trend analysis and evaluation of failures
  - key safety performance indicators are well known throughout the organisation
  - each individual knows his/her personal safety record and contribution to safe operations

**5. Practices and procedures**

- clearly documented practices and procedures are followed
- regular, disciplined internal audits
- regular external audits
- operating staff do self-audits
- timely follow-up to audits
- formal response to audit and record of action

**6. Incident investigation**

- a philosophy that encourages reporting of incidents
- a timely, disciplined system of reporting and investigating incidents
- a system to arrange incidents in order of priority and produce corrective steps

**7. Simulations of events**

- realistic simulations of incidents that allow check-out of all procedures and roles
- simulations that involve all internal organisations and external community organisations

**OPERATIONAL SAFETY OF NUCLEAR INSTALLATIONS**

**PICKERING NUCLEAR GENERATING STATION**

**1 - 19 JUNE 1987**

**REPORT TO THE  
GOVERNMENT OF CANADA**

## PREAMBLE

To complete its work at the Pickering Nuclear Generating Station, the IAEA Operational Safety Review Team (OSART) formulated its conclusions and views for the consideration of the responsible Canadian authorities.

The OSART experts' views on the Pickering NGS are based on the documentation made available by the Canadian organizations concerned, on oral communication with plant personnel, and on observation by OSART members of relevant plant activities.

This OSART report is being distributed initially only to the Canadian organizations concerned. Further distribution by the IAEA will be made only with the express permission of the Government of Canada.

Any use of or reference to the views expressed that may be made by the competent Canadian organizations is solely their responsibility.

## FOREWORD

by the  
Director General

The IAEA Operational Safety Review Team (OSART) programme assists Member States by advising them on ways of enhancing the safe operation of particular nuclear facilities such as nuclear power plants. Although good design, manufacture and construction are prerequisites, safety ultimately depends on the ability of operating personnel and the attitude and conscientiousness with which they carry out their responsibilities. OSART missions focus on these aspects when assessing a facility's operational practices in comparison with those used successfully in other countries and when exchanging, at the working level, ideas for improving safety.

An OSART mission is undertaken only at the request of a Member State and is not a regulatory type of inspection to determine compliance with national requirements. However, an OSART review can complement national efforts by providing an independent, international assessment that may identify areas for potential improvement which may have been overlooked.

An OSART mission affords an opportunity for OSART members and operating personnel to exchange knowledge and experience, to update the knowledge of regulatory personnel of the host country assigned to follow the OSART review, and to train personnel through observation of the experts involved in the OSART review process. This can contribute to the attainment of an international standard of excellence for operational safety, not through regulatory requirements, but through an exchange of information on, and voluntary acceptance of, successful, efficient safety practices.

The IAEA Safety Series document, including the Nuclear Safety Standards (NUSS) for nuclear power plants and the Basic Safety Standards for Radiation Protection, and the expertise of the OSART members themselves,

form the point of departure for an OSART review. However, the OSART review is performance oriented in that it accepts different approaches to safety in so far as they reflect good practice and contribute to an operating organization's safety objectives. Some OSART suggestions for long term improvement may be based on good practices at other facilities identified during previous OSART missions.

The scope of an OSART review is tailored to the specific needs of the particular facility. A full scope review would cover a number of operational areas: management, organization and administration; personnel training and qualification; conduct of operations; maintenance; technical support; radiation protection; plant chemistry; and emergency planning and preparedness. Depending on individual needs, the OSART review can concentrate on a few areas of special interest. For example, for plants under construction and approaching commissioning, an OSART can focus on plant and organizational preparedness for operation.

The OSART team presents its findings and recommendations on potential improvements in plant operational safety to the operating organization which reviews and analyses them in order to determine what further actions may be appropriate. Although findings and recommendations in the different review areas may carry different weights, no attempt is made to assign priorities. Moreover, although OSARTs assess a plant's performance in individual review areas, no assessment of overall plant safety is attempted.

In formulating its views, the OSART team discusses its findings with the operating organization and consider further comments made by team members. The team's working papers, or 'Technical Notes', are made available during the closing discussion at the site for use by the operating organization when considering the need for further action. An official Summary Report, such as the one attached, highlighting the more significant matters is prepared later and sent to the Member States.

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## INTRODUCTION

At the request of the Government of Canada, an IAEA Operational Safety Review Team (OSART) of international experts visited the Pickering Nuclear Generating Station (PNGS) of Ontario Hydro from 1 to 19 June 1987, to review the operating practices at this eight unit plant and to provide assistance and advice on how plant safety might further be enhanced.

Before visiting the plant, the Team (see Annex I) studied relevant information made available by PNGS to become familiar with the plant's main features, important programmes and procedures and the operating record of the past years. At PNGS, the team of experts, using techniques derived from their collective total of more than 200 years of nuclear experience, reviewed the relevant documentation, observed work being carried out in the various areas of review, examined the applicable procedures and instructions and interviewed plant personnel. Throughout the whole period of review, there was a thorough exchange of views on the plant's operational record, on the managerial approach, on training programmes and training facilities, on personnel performance in the control room and in the field, on the plant's upgrading programme and on the preparedness to cope with nuclear emergencies.

### General comment

The Pickering OSART mission was the nineteenth since the inception of the IAEA programme, and it was not only the first to Canada but also the first review of a large CANDU-type reactor. This mission was the tenth since the Chernobyl accident in April 1986, which showed once more the importance of proven design, reliable equipment and well-trained staff in ensuring operational safety. The process of identifying lessons to be learned from this and other events is continuing. In the past, Ontario Hydro has been following up relevant new developments, in close contact with the regulatory authority. It should carry on in the future with particular emphasis on human factors in operational safety and possible improvements in emergency response planning.



## Plant description

PNGS has been designed and constructed in two series of four units, known as Pickering A and Pickering B. The design is by Ontario Hydro and Atomic Energy of Canada. The eight reactor units have a gross thermal power of 1744 MW each, with the Pickering A units having a gross electrical output of 542 MW and a self-consumption of 27 MW each and the Pickering B units a gross electrical output of 540 MW and a self-consumption of 32 MW each. Each unit is capable of operating independently of the others while using certain common station services. The reactors are heavy water moderated and cooled. The fuel is natural uranium, and the refuelling is carried out on load. The 390 (Pickering A) and 380 (Pickering B) horizontal pressure tubes receive 12 bundles each containing 28 pencils. Pressurized heavy water is recirculated by the heat transport systems, which are equipped with 16 pumps (four in standby), 12 boilers (steam generators), four inlet headers, four outlet headers and four pump suction headers and connecting piping. Particular attention is paid to minimizing the loss of heavy water and to recovering any heavy water liquid or vapour that has escaped.

Construction of the first Pickering unit started in 1965, and its commercial operation commenced in 1971. The in-service date of Unit 8 was 1986. Neglecting the extended retubing outage of Units 1 and 2, the reactor units have been in commercial operation an average of almost ten years. Their operating record with respect to unplanned outages and availability is above average in comparison with plants of similar size and age. The number of unusual events reported to the regulatory authority was at the lower end of the expected range and none of the events had any significant radiological consequences in the station or its surroundings. Individual and collective radiation doses are similar to those in other successful nuclear power plants and much lower than the world average. Discharges of radioactive effluent are small and well below internationally recommended limits.

PNGS has safety features to cope with the design basis accidents, including shutdown systems, emergency coolant injection and a containment system. The Pickering A shutdown systems consists of shutoff rods and a

dumping mechanism for the heavy water moderator. Pickering B has two independent shutdown systems: shutoff rods and gadolinium poison injection. The emergency coolant injection of PNGS-A supplies high pressure natural water into the core followed by low pressure recovery and reinjection. At PNGS-B, each unit is supplied with high pressure injection from the station's common system. A separate low pressure recovery system is also available. The containment system is unique in that the eight reactor buildings are connected to a common single vacuum building by a pressure relief duct.

Three of the eight units were shut down at the time of the review, including the first and second units of PNGS-A for retubing. This station condition permitted the evaluation of the effects of a major outage with extensive restoration and modification work on the management of the other operating units.

#### **Operational safety features**

The OSART found the Pickering Nuclear Generating Station to be well maintained and operated by a competent and knowledgeable staff. The overall impression was that the station is performing above average in respect of safe and reliable operation. The safety status of the station was satisfactory and no shortcomings with equipment, personnel or operating practice were found that would threaten the continuation of safe and reliable electricity generation. However, several areas were identified where suggestions and recommendations for enhanced safety were made. An unusual large number of commendable practices that should be drawn to the attention of other nuclear utilities were also found at PNGS.

The OSART, noting that the organizational structure at PNGS differed from that in most other nuclear power stations, found that equivalent results are being obtained in respect of plant, personnel and environmental safety. This refers particularly to Ontario Hydro's approach to maintenance and radiation protection.

Efforts and results in industrial safety at PNGS were commendable. Most other areas associated with the station management, organization and administration were well taken care of, or had only minor improvements proposed. Regarding the training and qualification of personnel, all elements of an advanced programme together with the necessary training staff and equipment were found to be in place. Some recommendations, particularly in the refresher training of operations personnel were made.

From the review of the conduct of operations, several proposals were made aimed at making the current shift composition (particularly the shift supervisor's work) more manageable and efficient, and controlling temporary documentation on the plant/system status more closely. The planned development of symptom-based emergency operating procedures was endorsed. Regarding maintenance, the organization appeared to be in need of further optimization. Predictive maintenance is not fully used to supplement preventive maintenance. A comprehensive maintenance history programme has still to be developed to tie all databases together. There were no findings related to the work control system, in-service inspection, facilities or equipment. Several commendable practices were identified.

The operating experience feedback and surveillance testing areas offer considerable potential for improvement. The methodology for analysing unusual events needs further development for root cause determination, and the human element requires more attention. Surveillance activities should be extended and made more efficient for detecting latent deficiencies in equipment, personnel aids and personnel performance.

More advanced radiation protection concepts, limits and units are awaiting incorporation in station manuals and procedures, although station practices do not give rise to any major concern regarding dose statistics or effluent discharges. Some equipment and instrumentation, particularly for accident conditions, should be added. The chemists also need some new instruments: a long term replacement programme should be developed.

The station's emergency response capability has reached a satisfactory level but needs further development in such aspects as environmental surveillance, laboratory facilities, dose prediction capability, instrumentation and exercises held jointly with other organizations involved in public safety.

In conclusion, the OSART was satisfied to find all the equipment, personnel, and resources necessary at PNGS to ensure continued safe operation of the station. The proposals made for improvement do not qualify that statement but are necessary to avoid deterioration of this status and to permit active participation in the further advancement of operational safety. It is understood that the station will work out an action plan with an appropriate schedule for its response to the OSART Report.

1. **MANAGEMENT, ORGANIZATION AND MANAGEMENT**

Competent and knowledgeable staff are available at Ontario Hydro's Pickering Nuclear Generating Station (PNGS). The station is well managed by personnel who have a long association with the station and with Ontario Hydro. It is apparent that Ontario Hydro is committed to providing appropriate resources to support a safe, reliable and efficient nuclear operation.

**Station organization**

The PNGS organization is headed by a Station Manager who is supported by a staff of approximately 1500 people, organized into eight sections. The three major sections, headed by managers, are the Production, the Technical and the Commissioning Sections. Other sections supporting the Station Manager are Training and Conventional Safety, Quality Assurance (QA), Planning, Human Resources and Business.

The station organization is well defined, and a clear understanding of the responsibilities of organizational elements is well documented in the Pickering Station Reference Plans (P-SRPs) and Pickering Section Procedures (P-SPs). Additionally, individual job descriptions exist for the station's professional and supervisory staff. Communications within the organization, both vertical and horizontal, are good.

The station objectives are well defined and are derived from Ontario Hydro's Nuclear Generating Division "Key Effectiveness Areas", which are: Worker Safety, Public Safety, Environmental Protection, Product Quality (Reliability) and Product Cost. Each key effectiveness area is divided into performance categories, for which "Standards" and "Targets" are developed. These objectives are clearly supported at all levels of the organization and are fulfilled by well defined management programmes.

## **Quality assurance**

In addition to the station QA organization, Ontario Hydro's Technical and Training Services Division is responsible for external audits of all nuclear generating stations, including PNGS. These audits are performed by the Reactor Safety Audit Section of the Radioactivity Management and Environmental Protection Department (RMEP). The station QA organization performs QA monitoring (internal audits) and co-ordinates the RMEP (external) audits, including follow-up commitments and corrective actions.

Ontario Hydro's QA programme of auditing appears to be thorough, well co-ordinated and effective. Particularly noteworthy is the inclusion of "experts" (specialists in areas such as design, or experienced personnel from other stations) as members of audit teams, and the practice of "in-depth" audits which review entire systems (such as the heat transport system) horizontally.

## **Station/regulatory authority interface**

In general, the station's focal point for interface with the AECB is a Technical Superintendent in the Technical Section. With very few exceptions, incoming and outgoing correspondence related to AECB is reviewed by this person. In this manner, responsibilities for action are made clear and consistency is achieved. All outgoing reports and correspondence to the AECB must be signed by the Station Manager. Such centralization of the responsibility for regulatory interface seems to work well at PNGS and is viewed as necessary owing to the large, diverse organization associated with the station. Additionally, it is evident that the AECB project officers communicate regularly with the station staff; a solid, professional relationship exists.

## **Industrial safety**

Industrial safety is more than a commitment at PNGS; it is a visible part of doing business at the station. Overall the industrial safety programme was found to be exemplary. Since 1985, when the temporary total

disability rate exceeded the management objectives for that year, much effort was spent on upgrading the programme. The most noteworthy products of the effort to upgrade the programme are: the PAUSE programme, the programme of regular safety meetings, a comprehensive and diverse safety training programme, and the Conventional Safety Manual.

PNGS has an established system of fire prevention, fire detection and fire fighting. Each shift has a Fire Crew as a constituent of shift staffing. The members of this crew are trained in an annual intensive four day session at a fire school located at the Bruce Station. Hose stations and fire extinguishers are located appropriately throughout the plant and are kept ready with current tests documented. Sprinkler systems are installed in certain areas where appropriate.

The station fire crews are backed up by the Pickering Fire Department, which is located only minutes from the plant. Consideration should be given to implementing a programme of fire fighting drills involving the local fire department to establish support and ensure the compatibility of equipment and procedures.

#### Document control

The PNGS document system is organized into a five tiered hierarchy. The requirements for the preparation, review and authorization of these documents is clearly stated in a governing document. In practice, the document hierarchy is not used as intended. Many of the Pickering Station Instructions (P-SIs) address procedural rather than policy matters. Also, similar, possibly redundant, procedures were found for various plant sections. Certain outdated procedures were also located in the system. The existing document system should be reviewed to ensure proper distinction between policy and procedure level documents, to retire outdated documents and to consolidate those documents that are redundant or overlapping. It is noted that this problem has been recognized by plant management and a programme of "Document Rationalization" has been assigned to the QA Section.

## **Physical security**

The legal requirement for physical security at PNGS is as a result of the Atomic Energy Control Act amendment on Physical Security Regulations dated October 25, 1985. As a result, the station has been upgrading its security system and is still in the process of doing so. Apparently, the extent of the required physical security measures is under continuing discussion between the AECB and the nuclear generating industry.

PNGS has established close contacts with the Durham police force as the first armed response force. The principal mission of the station guards is to detect unauthorized intrusion, to notify the local police and to implement station procedures to close the station buildings and "box-up" the control rooms. It appears that the physical security improvements instituted at the station are adequate to provide a high degree of assurance that intrusion into the protected area would be detected so that appropriate response measures could be implemented.



## 2. TRAINING AND QUALIFICATION

The PNGS training programmes for station personnel compare well with corresponding programmes in other countries with advanced nuclear power programmes. Personnel have access to good training facilities in the Eastern Nuclear Training Centre (ENTC) which has two full-scope simulators, process loops and mechanical and instrumentation workshops. In general, the training programmes are of good quality. The training organizations are staffed with qualified personnel, and are equipped with the tools needed for effective implementation and control of training activities.

PNGS Training Section and ENTC personnel have work experience in technical areas for which they have training responsibilities. In addition, instructors have received instructional skills training. Training materials were found to be of high quality. Plant modifications that are relevant to training are taken account of in the systematic updating of the simulator.

### Organization

Written policies, instructions and procedures at the division, station and section/school level are available to control the training programmes. These reference plans, station instructions, section procedures, manuals and station forms are not always structured in the intended hierarchical and logical order. A thorough review of the relevant documents should be made to rationalize, update and correct these and other shortcomings. The classification system and requirements for preparation and control of station documents should be followed.

Job responsibilities are clearly defined and understood by the training staff. Performance appraisals are conducted on training personnel and feedback is obtained from trainees on the effectiveness of training programmes. Reports on training accomplishment and forthcoming scheduled training are regularly issued.

Station systems training for operators and mechanical and control maintenance staff is done systematically. The subject is presented at three discrete levels, i.e. level 4 (basic), level 3 (intermediate) and level 2 (advanced), and separate sets of training manuals have been prepared for each level and work group. This multilevel approach is a commendable practice and permits the station system training to be focused on the specific knowledge needed for the proper discharge of job responsibilities.

#### **Simulator training**

Formal arrangements should be made to enable simulator instructors to participate periodically in shift operations to maintain familiarity with job requirements, station system changes and recent practices and to gain operating experience. Such assignment would allow the instructor to obtain first hand feedback of training needs. It will also encourage close contact between trainers and trainees.

The PNGS-A simulator is not fully capable of modelling the response to boiler tube failure. Consequently, diagnostics training does not cover the full range of malfunctions which affect critical safety functions. This shortcoming, which has been known for some years, should be corrected without further delay.

#### **Operations personnel**

The three-year initial training programmes for Shift Supervisors (SSs) and Unit First Operators (UFOs) were found to be well designed and implemented in a systematic and structured manner. The refresher training, however, is not up to the same standards. Malfunctions to be practised during refresher training should be selected according to more clearly defined criteria such as related operating experience. Simulator exercise guides should also be developed providing learning objectives, scenario description and exercises to be carried out. Pre-exercise classroom briefings and post-exercise critiques should also be conducted. An evaluation system should be established to assess all participants, including the shift supervisor, as well as the overall team performance. The number of trainees per class should be limited more strictly to allow adequate time for operators to train on the simulator panels.

AECB requires examination and initial authorization of UFOs and SSs; however, contrary to the practice in most countries, there are no regulatory requirements on the requalification/retraining of UFOs/SS. It is recommended that such requirements be established.

#### **Maintenance personnel**

Generally, the mechanical and control maintenance programmes are of good quality, well designed and systematically delivered. Trainees with limited industrial experience can develop the abilities required to maintain CANDU plants through these programmes. Some maintenance staff, however, were not able to complete their initial ENTC training and/or systems training before they were assigned to work in the station. Those persons, however, were only assigned to tasks they were capable of performing, and their work was supervised by experienced staff. For the near term, a plan should be developed for these staff to finish their initial qualification training, including station systems training.

#### **Radiation control and chemical personnel**

Radiation control and chemical technicians currently take station systems courses together with the operators as part of their initial training. Since these personnel have different job responsibilities, the systems training programme should be different for each. The plan to design a separate course for each group is encouraged.

#### **General employees**

Station employees must periodically requalify in some disciplines, including: radiation protection, work protection, confined space entry, first aid, cardiopulmonary resuscitation, fire training and driver training. In addition, safety meetings are held once every five weeks to review significant event reports, reported accidents, work protection procedures and equipment. The industrial (conventional) safety part is particularly noteworthy.

### 3. CONDUCT OF OPERATIONS

The Pickering NGS (PNGS) Production Section ranks as one of the largest compared with those at other nuclear sites. The large workforce in this section (approximately 1000 persons) presents a challenge to the management which has the responsibility of ensuring that the work gets carried out safely. In addition to the normal tasks associated with operations, most plant maintenance is completed on shift as part of the normal shift work.

The operating history, capacity factors and safety record of the station are impressive. The station's management has directed attention to both achieving high safety standards and maintaining efficient operation. The overall impression of the Production Section was one of professionalism and experience in the operation of nuclear power plants. All individual members of the Production Section interviewed displayed enthusiasm for their work. They appeared fully to understand their duties and to carry them out in a commendable manner.

During the period of this review, three of the PNGS-B units were operating at nominal full power and one was shut down for overhaul. This plant status provided the opportunity to evaluate the effect of a major outage on the management of the operating units.

#### Organization and administration

The organizational arrangements for PNGS were found to differ from those in many utilities with respect to the staffing and responsibilities of the shift teams. All plant operations and most of the mechanical and control maintenance, for both functioning and shut down units, is carried out by the shift crews. The organization is based on the shift units' ability to conduct operations, maintenance, testing, modifications and training.

The co-ordination and supervision of all work is carried out under the direction of the shift supervisor (SS). SSs appear to cope with the large workload effectively. Their ability to satisfy the technical and safety aspects of the job is recognized. However, there is some concern that the administrative burden could impinge on the efficient performance of the shift crew. Consideration should be given to providing assistance to enable the SS to devote sufficient time to shift management.

Although the role of the SS is of utmost importance to the safe and efficient operation of the plant, the average tenure of SSs at PNGS-B is two years. This appears to be the consequence of the current station policy that experience as an SS is a prerequisite to being appointed as a Technical Superintendent. The position of SS should not be viewed as a 'training position', however, but as a career in itself. Alternative ways of obtaining the experience resulting from SS work should therefore be considered.

The Production Section work programme identifies specific objectives clearly and sets goals which are challenging but achievable. Appraisals cover individual achievement and shift performance in such areas as employee safety, employee relations, reliability and productivity. The importance of a 'balanced achievement' on all subject matters is stressed. Another important objective is to verify that the work load is being shared equally between shift crews.

#### **Plant operation**

The operating team on each shift consists of a shift operating supervisor (SOS), a Unit First Operator (UFO) who is responsible for the overall operation of each individual unit, a second operator and one or two assistants carrying out the field work.

To ensure the adequate transfer of information on equipment status during shift turnover, use is made of a 'long term status book'. The book is updated daily and contains a resumé of all important plant changes and documentation changes, e.g. jumper records (JRs), alarm settings etc. This is considered a commendable practice.

The written statement of shift turnover exchanged between outgoing and incoming SSs is not signed and dated. The turnover of the Second Operator is documented in the form of shift logs, but the verbal exchange of information takes place in an area remote from the room in which the log is written and kept. This means that the information transfer at shift turnover has some shortcomings which should be remedied to ensure the otherwise high professional standards of shift turnover at PNGS.

Each time an abnormal event is deemed to require reporting the SS writes a Significant Event Report (SER). One of these reports revealed shortcomings in the assessment of shift workload. Although swift action was taken to avoid a recurrence, due consideration should be given to the workload when a large number of operations are planned at the same time.

The review of operational documents showed that a large number of operating memos (used to transmit short term operating instructions) and jumpers are in force for each unit. Appropriate resources (Technical Section) should be directed to the task of substantially reducing these temporary measures.

#### Plant status controls

Plant status in the main control room is efficiently surveyed by operators using various checklists applicable to the normal operation of the unit. Provision also exists to monitor the status of equipment relating to the availability of the heat sink on a shutdown reactor. All operations, routine testing and plant isolations are conducted in accordance with written procedures.

A large number of stickers are attached to equipment on the control room panels, without clear indication of their temporary nature and without regular review of their continuing applicability. This was also noted in various plant areas, where flowsheets and other descriptive information are attached to electrical panels and other plant components. These items should be regularly reviewed for continuing applicability and removed when no longer applicable.

During the review evidence was provided of personnel inadvertently operating plant components on the wrong reactor unit. Station management should initiate a complete review of the identification of plant items, taking into account the results of events analysed and field operator experience, and carry out appropriate corrective actions.

Procedures for work authorization and tagging were found to be very detailed. At PNGS much effort is made and considerable resources are devoted to component/system isolation and worker protection. Detailed checksheets are employed to ensure that all steps relating to the isolation procedure are followed. Independent checks are also carried out to verify that isolations are adequate for work to be performed in a safe manner. These independent checks are considered to be a commendable practice.

#### **Emergency operating procedures**

The existing event based emergency operating procedures were reviewed and found to be clear and straightforward owing to the extensive use of logic diagrams. Ontario Hydro is currently studying the use of symptom based emergency procedures in addition to event based procedures. A pilot programme is planned to start soon on Pickering B. This work is endorsed.

If the main control room of Pickering B became uninhabitable, the shutdown monitoring of the four units would be carried out from the unit emergency control centre (UECC). Co-ordination is performed by the shift supervisor or shift operating supervisor from the secondary control centre. It is recommended that full scope drills be performed to identify potential difficulties in the transfer of the controls from the main control room to the UECCs.

#### 4. MAINTENANCE

Maintenance of PNGS with eight nuclear units and all supporting systems and equipment is recognized as an enormous task requiring a large expenditure in resources. The overall maintenance programme was observed to effectively support the safe and reliable operation of PNGS. Maintenance personnel appeared to be highly skilled and motivated to meet the challenge of keeping the plant in good working order. Maintenance facilities and equipment were found to be adequate.

##### Organization and administration

Approximately 90% of plant maintenance at PNGS is carried out by shift crews. Although the maintenance staff reports to the Shift Supervisors in the Operations Subsection of the Production Section, functional responsibility for the administrative aspects of maintenance rests with the Mechanical and Control Maintenance Units in the Maintenance Subsection of the Production Section. This overlapping of responsibility within the Production Section has led to some lack of continuity within the maintenance groups.

It was noted that there is a general lack of engineering support in the maintenance area which results in the failure to analyse specific problem areas as they arise. This is particularly apparent in systems not related to safety. In the areas of both control and mechanical maintenance, it is recommended that engineering manpower be provided to assist in problem investigation and strategy development within the maintenance organization.

##### Material condition/housekeeping

Poor material condition and housekeeping were observed in some plant areas. This may be due in part to the assignment of responsibilities to the shift organization which it is not capable of carrying out all tasks under



the present staffing structure. It was also noted that the maintenance supervisors are not always held accountable for the overall material condition and housekeeping of plant areas. It is therefore recommended that the present organizational and staffing structure for maintenance responsibility at PNGS be reviewed. The assignment of a single individual to be responsible for all maintenance activities and resources at the plant should be considered, including on-shift maintenance activities. The Maintenance Supervisors should be held accountable for the material condition and housekeeping of the plant.

Some plant piping systems and components such as boiler feed pumps and secondary plant heat exchangers were found to be under repair. These components were sometimes unattended, and were not always covered or otherwise secured to prevent the entry of foreign matter into the component or the piping system. No effective method of ensuring the covering of large openings in plant piping and components was in use. A method should be devised and implemented to prevent the entry of foreign materials into piping systems and components during maintenance periods.

#### **Preventive maintenance**

Preventive maintenance at PNGS consists of a comprehensive programme of equipment lubrication, inspection and checking. The purpose of this programme is to evaluate the condition of plant equipment for continued performance and to repair or replace components as appropriate before failure occurs.

The current programme, however, is manpower intensive and depends to some extent on equipment tear-down inspections to determine condition. More predictive maintenance techniques to determine the status of equipment performance while on-line are recommended.

#### **Corrective maintenance**

Corrective maintenance which includes equipment troubleshooting and repairs following breakdowns is conducted at PNGS primarily by the rotating

shift maintenance crews. Deficiency reports (DRs) are used as the basis for this work. Although a large fraction of the plant maintenance activities involves equipment repairs, a review of outstanding DRs and a look at the general material condition and housekeeping in the plant show that there is a large backlog of corrective maintenance work. Although a great amount of resources is applied in this area, observations showed that some improvements may be possible to increase work effectiveness. Delays sometimes occur when tools or equipment are not available when needed, and in starting each shift. Also, it was noted that highly trained and qualified journeymen mechanics, welders and control technicians are used on such lower skill jobs as fire watches, tool runners and parts locaters.

The use of additional operating personnel should be considered to authorize work during peak periods such as the beginning of a shift. Also, consideration should be given to using lower skilled personnel such as helpers to provide assistance to qualified personnel in order to increase productivity, and hence to improve the plant's material condition.

#### **Work control system**

Comprehensive work schedules are issued by the Planning Section and are used as the basis for meetings held by the Shift Supervisors on each crew to determine the status of work in progress and to discuss emergent work items not on the work schedules.

Sometimes the work schedule is ineffective in controlling shift activities, and shift maintenance control may not be generally co-ordinated from shift to shift. Current methods of controlling work at PNGS should be reviewed and approval of the schedules by maintenance supervisors should be considered. These supervisors could then be held accountable for work performed and for the plant's material condition and housekeeping.

#### **Stores**

There is no segregation of safety and process components in stores, and no unique facility to house safety components. Also, stainless and

carbon steel materials are stored together in the warehouse. In addition, no special storage facilities are available for storing and protecting sensitive components. A listing of quality parts and equipment used in safety related applications should be made. These parts and equipment should then be stored in separate bin locations. For sensitive components, arrangements should be made for special storage conditions, such as under temperature and humidity control.

#### **Maintenance history**

PNGS has not established a comprehensive maintenance history programme which ties all databases together. This creates gaps in the performance review process. Current maintenance history records should be consolidated and a documented programme should be developed that can be used to provide maintenance history when required for use in job planning, equipment performance trending, etc.

## 5. TECHNICAL SUPPORT

Technical support available to PNGS was found to be adequate for safe plant operation. The staff interviewed were knowledgeable and motivated to perform their duties. It was established during this review that there was a positive attitude toward safety matters.

### Organization and administration

A central role in the technical support function is assigned to the Technical Section. Its tasks support, but are separate from, those of the Production Section which executes all hands-on activities. This arrangement ensures that the Production Section's general responsibility for safe plant operation is not compromised.

A significant part of the Technical Section staff consists of System Engineers, each having responsibility for selected systems. These engineers have developed individual working practices. Consideration should therefore be given to harmonizing these currently differing approaches. Another issue of potential concern is that in certain areas the workload seems to be more than the present personnel can cope with.

### Surveillance programme

The basic elements of an adequate surveillance programme were found to be in place. All components important to safety undergo periodic testing. Responsibilities related to each test are clear. Test procedures or equivalent instructions exist for each test and provide detailed guidance on test execution. Scheduling and monitoring routines ensure that all tests are done on time. Test execution is well organized. Deficiencies found in tests are corrected and evaluated in a systematic manner.

It was noted that the surveillance programme is not based on explicit detailed requirements attached to the operating licence, as in most other nuclear power plants. A unique approach is used to determine the test needs. This concept involves setting unavailability targets of systems and calculating required test frequencies by reliability analysis techniques. The method can be regarded as a commendable practice.

Despite the sound basic structure of the surveillance programme, several improvements are recommended. Its objectives and administration should be described in a single plant procedure to consolidate the fragmented descriptions that now exist.

Some systems should receive additional attention. Within the current surveillance programme particular attention is paid only to the Special Safety Systems. However, all systems needed for ensuring reliable decay heat removal should be given equivalent treatment, and a common surveillance testing programme should be established and implemented. To have a uniform basis for the test programme, unavailability targets should be defined for all these systems. Minor hardware changes may be needed to facilitate proper testing and diagnostics of all systems providing the decay heat removal function.

The current main objective of the surveillance programme is restricted to only verifying the prescribed reliability of systems. It is not necessarily used to detect degradation of plant parts or to monitor component performance in the long term. The scope of some tests should be extended to take account of these latter aspects. In many other tests sufficient parameters are available but they are not recorded because the applicable procedure requires only that visual checks be made, and quantitative values are not entered in the data sheets. It is further recommended that systematic efforts be made also to record meaningful diagnostic data. The test reports should then be routed to the responsible system engineers. Guidance should be given for analysing trends for important parameters.

Long term monitoring of the performance of individual components would require an organized file of the test records. The current filing system does not serve this purpose and should be improved.

Some of the test procedures reviewed do not provide target values or acceptance criteria for parameters to be recorded. Consequently, the acceptance of the results depends on the test personnel's subjective judgement which may not be consistent from test to test. In other procedures the acceptable results are given without any tolerances. The test procedures should be reviewed against the input used in licensing calculations and against the commissioning test results, and the necessary acceptance criteria and associated tolerances should be added.

Testing of the reactor shutdown system and of the charcoal filters in the exhaust air system were identified as commendable practices because these tests are carried out more thoroughly than at most other plants.

#### **Reactor engineering/fuel handling**

Three topics were in the focus of the review: control of power distribution, control of reactivity and refuelling. The impression of reactor engineering was positive. Control of the reactor power distribution *ensures that the reactor is operated within its design envelope*. The core reactivity is controlled in a safe manner. Planning and execution of reactor fuelling is done with great care and clearly defined and documented work methods are used.

All off-site steps of the fuel cycle were found to be subject to a solid quality assurance programme. The fuel behaviour in the core is well understood and operating experience has led to design improvements.

At PNGS, there are on average several leaking fuel bundles per unit each year. This fuel failure rate may be influenced by the unusually high thermal loads of the fuel, especially during fuelling. Thus no definite conclusion could be drawn about whether the best results seen in other plants could be achieved with strengthened quality control or improved design features.

The on-site storage of fresh fuel and the equipment used for fuel handling were not of the high standard found at other plants. Damage to fresh fuel seen occasionally indicates that the transport packages may experience rough handling. The methods of handling and storing fresh fuel on site should be reviewed for consistency with the good controls used throughout the off-site stages of the fuel cycle.

#### Computer capabilities

Process computers, because of their use in plant control, are more important at PNGS than in most other nuclear power plants. The computers are also used to collect and process data on system parameters and to initiate alarms in the event of limits being exceeded. Hardware and software of the computers are controlled with extreme care, which contributes to its excellent operating record.

It was noted that there was a steadily increasing backlog of software changes awaiting implementation. The control of computer status, although well organized, becomes complicated when major programme revisions undergo testing for years. It is recommended that this backlog of work be speedily reduced.

An unusual problem in PNGS is the ageing of process computers. In many nuclear power plants old computers have been replaced. At PNGS this may be difficult because of the role of the computer and the need for extensive testing associated with hardware changes. However, replacement of the computer may be inevitable in the future and consideration should be given to appropriate means of carrying this out.

## 6. OPERATING EXPERIENCE FEEDBACK

The review focused on the three main stages of processing operating experience: (1) identifying the safety issues; (2) analysing these issues; and (3) taking any corrective actions necessary to close the feedback loop for improving operational plant safety.

The overall impression is positive: operating experience is processed effectively at PNGS, which is considered to be the result of various contributing factors indicative of good plant management.

Reporting and processing operating experience are clearly assigned to two separate groups: the Production Section deals with the 'on-line' activities of operation, surveillance and maintenance, and the Technical Section deals with the 'off-line' activities related to operating experience feedback.

Clear reporting and ranking criteria guide the collection of data and the analysis of safety issues.

Clear assignments of personnel to each system and plant function and their professionalism ensure thorough processing of operating experience.

Quantitative assessment of plant performance, complementing the establishment of well defined objectives, provides the basis for improvements in operational safety, with little infringement by non-technical considerations.

### Performance indicators

A set of performance indicators provides the station management with an advanced tool for assessing operational safety against targets and for initiating corrective actions if these targets are not met. This systematic



approach is a commendable practice. Ontario Hydro should consider broadening its existing programme by defining additional performance indicators. This would provide a more complete picture of both the current plant performance and the effectiveness of efforts towards further improvement.

#### Significant event reports

On examination, the analyses of the significant events reported to the regulatory authority between 1983 and 1986 were found to vary in quality. The most probable cause of these shortcomings is the lack of detailed guidance on how to carry out the analyses. Development of such analytical methods should be given priority taking into account similar work elsewhere. In addition, Ontario Hydro's head office should assume a more active role in carrying out and co-ordinating the event analyses. Particular emphasis should be placed on due consideration of human factors. Such broadened approach is expected to generally contribute to quality and consistency of the results and to facilitate the identification of underlying causes and appropriate remedies.

About 40% of the significant events were caused by human failures. This contribution shows no specific trend over the past four years. If equipment failures attributed to installation deficiencies were included, the human failure contribution would rise to more than 57%. All corrective actions were taken on a case-by-case basis. Although detailed statistics on human failures were widely disseminated to plant personnel, there is no evidence of an in depth analysis of this generic issue.

The actual contribution of human failures to the unavailability of special safety systems, safety support systems and safety related systems should be assessed. To a certain extent, this could help determine the most relevant corrective actions to meet the reliability targets.

#### Surveillance testing

Review of the significant events reported to the regulatory authority

shows that 36% are deviations detected through surveillance testing. Although 64% are spontaneous incidents, most of these could have been averted by more effective surveillance. This implies that the objectives of the surveillance activity at Pickering NGS are not being achieved. These objectives, in accordance with the Station Reference Plan (P-SRP 3.10, paragraph 1.2), are to detect all latent deficiencies in good time before they lead to spontaneous failure.

It is understood that perfection cannot be achieved but surveillance in other nuclear power plants is usually more effective. The PNGS surveillance testing programme covers the right areas but the scope of the programme, the testing procedures, the detection thresholds and the evaluation of results can still be improved (see also the discussion on surveillance testing in Section on Technical Support).

More emphasis should be put on in depth analysis of these deviations, since they precede spontaneous failures. The establishment of appropriate detection thresholds is important to detect any deviations in good time. Personnel qualification and personnel aids need to receive as much attention as equipment in the surveillance programme. Systematic root cause analysis of incidents would be a good starting point to find out how current surveillance could be improved to reveal latent deficiencies more effectively.

#### **Corrective action policy**

On reviewing the present status of the outstanding corrective actions, it is noted that in addition to corrective actions resulting from the analysis of internal and external operating experience, seven times more corrective actions are imposed by regulatory requirements. These actions are mostly design modifications, differing widely in respect of safety relevance and costs involved. Possible 'software' improvements, for instance in surveillance activity, personnel performance and procedure development, seem not to be utilized to the extent possible. Consideration should therefore be given to developing a well balanced modification

programme including both hardware and software, with priorities derived from the contribution of the individual measure to general safety. It is understood that such a programme would be developed by PNGS and authorized by AECB for implementation.

## 7. RADIATION PROTECTION

The Canadian approach to radiological protection differs from that in most other countries. For this reason, there is no typical health physics group at PNGS to provide radiological coverage for personnel; instead workers are trained and qualified to handle their own personal radiation protection, and the primary responsibility for appropriate conduct rests with the individual.

The goals and results of such a philosophy appear acceptable on the basis of low occupational exposure, small amounts of radioactive effluent discharges and few radiological incidents.

The average individual dose at PNGS in 1986 was less than 5 mSv, i.e. one tenth of the yearly limit recommended by the International Commission on Radiological Protection (ICRP) for occupational exposure. The collective dose per unit of practice (man-Sv/GW-year) in 1986 is similar to that achieved at nuclear power plants in various European countries and much lower than the world average. The trends in both individual and collective doses over the last 12 years show considerable improvement. In this respect, Pickering B is superior without any doubt, while Pickering A accounted in 1986 for 90% of the total station dose. Since this large fraction was mostly due to the refurbishing work at Units 1 and 2, a further decrease in the collective dose is expected.

Discharges of radioactive effluents are below 1% of the derived emission limits (DEL) authorized by the AECB and are less than one-fifth of the limit recommended in ICRP Publication No. 26.

Analysis of radiological incidents shows two over-exposures in the last five years and a number of minor incidents whose consequences did not exceed exposure limits. This record is within the range to be expected in view of the numbers of units and personnel.

This satisfactory performance derives not only from good radiation protection practices but also from good design and good maintenance.

#### Advanced concepts

At PNGS, radiation protection procedures are based on old concepts and should be revised to correspond with current ICRP/IAEA recommendations. The administrative dose control, which is still based on limits for each critical organ, should be replaced by a new dose limit (0.5 Sv per year) for every tissue except the eye lens to avoid stochastic effects and a limit (50 mSv per year) on uniform irradiation of the whole body using weighting factors to determine the effective equivalent whole body dose. Similarly, the outdated concepts of the maximum permissible body burden and the maximum permissible concentration should be replaced by the annual limits of intake (ALI) and the derived working levels (DWL) to control incorporation hazards.

In view of new insights into the radiation risks to pregnant women between the eighth and fifteenth weeks, the applicable dose limits in the relevant radiation protection procedures should be revised. It is recommended that access to radiation areas be prohibited during these periods of pregnancy.

Actual effluent discharges from PNGS are so low that they give rise to no concern. The authorized limits, however, are based on older ICRP/IAEA recommendations (5 mSv per year) which are still acceptable provided that the dose calculation is based on conservative assumptions and the critical group is very small. In the light of more recent recommendations (1 mSv per year), consideration should be given to introducing a lower dose limit for the 'combined exposures from all actual and future practices that will affect the same critical group'.

#### External radiation control

Radiation protection at PNGS is based on two elements: (1) management responsibility for establishing adequate procedures, making instrumentation, equipment and facilities available as required, and providing for personnel

training and qualification; and (2) individual responsibility for following applicable procedures and minimizing radiation exposure for personal benefit. Jobs performed within the controlled area require a radiological work plan (RWP) which is usually authorized by a 'green man'. This is a fully qualified worker who can supervise, directly or indirectly, the radiation protection of others after appropriate training and station experience. The RWP is reviewed by the Radiation Control Unit after completion of the work. The Health Physics Unit on site, which reports to Ontario Hydro's Health and Safety Division and is not part of the PNGS organization, performs an advance review if work has to be carried out in radiation fields of over 30 mSv/h (3 rem/h) and in air contaminations of over 1000 times the maximum permissible concentration (MPC). The scope of the involvement of the HPU, however, should be redefined by an extension to lower dose rates and contamination levels and possible inclusion of estimated individual and collective doses. During outages 'exposure control assistants' are assigned to control the work and dose of non-qualified workers.

The system and the applicable procedures for external radiation control were found acceptable. Trends in the annual radiation doses over the last 12 years show an effective feedback process for dose reduction (by a factor of two for the total annual radiation dose). No overexposures (whole body and extremity doses) have occurred in the last two years.

#### Internal radiation control

Internal exposure is minimized by good practices, adequate equipment such as ventilated suits and portable monitors, and appropriate clothing and body controls of personnel leaving contamination areas. A special personal monitor for controlling the intake of carbon-14 aerosol was developed in response to the hazard originating during the retubing of Units 1 and 2. Consideration should also be given to nose sampling techniques as a backup.

Local air extractors were lacking in work areas where heavy water spills can occur and other minor shortcomings such as poor housekeeping practices, a backlog of equipment calibration and inadequate use of

protective suits were seen. The general impression was good, however, with all required instrumentation and equipment in place and qualified personnel available to ensure effective internal radiation control.

#### Instrumentation

The original instrumentation for gamma monitoring and tritium sampling within the reactor buildings did not meet the design intention. In addition, no appropriate post-accident instrumentation was installed. As a consequence, the information on the radiological conditions inside the reactor buildings under both normal and accident conditions is insufficient. To remedy these shortcomings, a reliable gamma monitoring system, a new tritium sampling system, and wide range gamma meters up to  $10^4$  Sv/h ( $10^6$  R/h) for accident conditions in the reactor and vacuum buildings should be installed. In addition, appropriate recording facilities capable of recording transient conditions in the buildings should also be provided.

#### Radioactive effluent control

The discharges of radioactive effluents to the environs were found to be much lower than the prescribed derived emission levels. Owing to extensive use of resins, the volume of liquid radioactive waste is relatively small. After collection and storage in tanks, the effluents are sampled and discharged in batches after authorization of discharge. Monitoring of the gaseous effluents for noble gases, iodine, aerosols and tritium is satisfactory.

The service water system uses lake water for cooling several heat exchangers, including those of the moderator and shutdown cooling system. Leaking heat exchangers would cause uncontrolled radioactive releases because there is no continuous monitoring of the discharge line. Monitoring for leaks by other parameters may not be sufficient to detect incipient failures in the heat exchanger tubing. Service water should therefore be adequately monitored for early detection of heat exchanger leaks.

## Waste management

Although no waste processing is carried out at PNGS, only waste classification for temporary or permanent storage and shipment off-site, the waste facilities lack some features relevant to safety. The ventilation system should be modified to prevent possible spread of contamination, and an aerosol monitor should be installed in the waste storage area. Special efforts should be made to avoid the accumulation of plastic bags of waste over weekends when no personnel are available for packing them into metal drums. In general, fire detection and fire fighting systems in the waste storage area should be upgraded.



## 8. CHEMISTRY .

The chemistry control at Pickering B covers more than 30 types of fluids. Chemistry work for all fluids is of good quality. Especially notable is the fact that the boiler (steam generator) tubes have not suffered any failure, thanks to both excellent chemistry control and the selection of appropriate materials, namely Monel 400.

Adequate chemical analyses are carried out. The chemical technicians are well trained and experienced in following their well prepared manuals. Chemical specifications are regularly updated and measurement frequencies altered as necessary.

### Chemical laboratories

Most analytical equipment in the station laboratories is rather old. Consideration should be given to replacing the chemical analytical equipment on a planned basis. Replacement of the equipment will facilitate maintaining a high quality of chemical analysis and will provide an additional motivation to the chemical technicians.

The air velocity through the fumehood windows in the chemical laboratories does not reach the value adopted for PNGS (30 m/s minimum in accordance with applicable standards). Fumehoods equipped with air velocity indicators would help the chemical technicians to maintain good environmental conditions in the laboratory.

The radioactivity counting room should be kept isolated and clean to protect samples and detectors from contamination.

### Post-accident sampling

PNGS has no post-accident sampling system for the heat transport systems (HTS). Since the current HTS heavy water sampling station is

located in the reactor building, no sampling would be possible under severe accident conditions. The installation of an appropriate post-accident sampling system should therefore be investigated. The design objectives of such systems at other nuclear power plants should be taken into account, i.e. the capability of extracting liquid samples containing up to 37 TBq/kg (1000 Ci/kg) iodine-131 and gaseous samples containing up to 4 TBq/L (100 Ci/L) noble gases. Procedures should also be developed and personnel trained accordingly to process such highly radioactive fluid.

#### Coolant activity

The iodine-131 concentration at Pickering B of 200-700 kBq/kg (5-20  $\mu$ Ci/kg) in the HTS is relatively high in comparison with other heavy water reactor plants (40-100 Bq/kg (0.0001-0.003  $\mu$ Ci/kg)) because of a fuel failure rate of 0.1%. The oxygen concentration control in the HTS should be examined, in the light of recent publications, to reduce fuel failures. When fuel failures are avoided and the content of fission products in the HTS decreases, the cobalt-60 concentration can be measured more accurately. This measurement may give new information about possible reduction of radiation dose, since cobalt-60 is the main cause of the occupational radiation exposure.

#### Instrumentation and control

At least ten of the in-line monitors for boiler feedwater systems do not give correct values because of poor design and lack of spare parts at the supplier. Chemical technicians are obliged to monitor the parameters by manual measurements, which is less effective for good chemistry control. The in-line monitors should be replaced as soon as possible.

Development of on-line computer software should be completed to produce graphics of all measured values, which facilitate the identification of data trends on a daily basis.

## 9. EMERGENCY PLANNING AND PREPAREDNESS

The starting point for emergency response planning is the assumption that an accident causing highly radioactive releases to the environs can occur, even if its occurrence is extremely unlikely. No limitations should be applied to any accident scenarios other than those due to inherent plant safety features. Thus at PNGS a large amount of noble gases, iodine, aerosols, tritium and carbon must be assumed to be able to escape from the reactor to the environs. The emergency planning and preparedness system at PNGS must take account of this for all countermeasures.

In general the countermeasures planned by PNGS, Ontario Hydro and the provincial and municipal authorities meet these requirements. The emergency arrangements, together with practices and exercises, currently applied at PNGS provide a level of emergency preparedness that should enable an effective response to an emergency at the site. In the areas of radiation measurement and dose calculation, however, a broadening of the current approach is considered necessary.

### Organization

There is an effective corporate emergency planning and preparedness organization at the Ontario Hydro head office and at PNGS from the highest management level to the operating staff members. The involvement of the highest management level in all important aspects of emergency planning ensures the necessary support for the effective development and implementation of the emergency response capability. This is a commendable practice.

The Radiation Control Unit and the Operations Subsection Superintendent at the site are responsible for the site emergency plan and implementing procedures. They are experienced and carry out their work with care.

The emergency response organization at PNGS provides an effective framework for both the initial and the long term emergency response. It relies mainly on the Shift Supervisors and the experienced shift workers. This on-site organization has to implement immediate emergency response activities. In a later phase they are relieved and supported by the on-site management group and the Ontario Hydro emergency operations group at Toronto. This system corresponds to international practice.

The four PNGS environmental survey teams recruited from the unaffected plant have the only capability for environmental measurements in the early phase. Additional support should be obtained through co-operation agreements with external bodies, e.g. other nuclear generating stations or research centres.

The continuation of required laboratory services should be ensured should the laboratories in the on-site service building and in the Health Physics Service Centre be impaired by contamination or high background radiation levels. Backup facilities outside the primary zone in conjunction with an emergency control centre would offer a good solution under such conditions.

#### **Emergency plan and implementing procedures**

The corporate emergency plans of Ontario Hydro and PNGS are well documented and maintained. Detailed action charts are provided that are a useful aid to all persons in charge. Problems may arise from the categorization of emergencies which the shift supervisor is required to make for the notification of the public authorities. He should, therefore, be given clear and easily applicable criteria and be trained in their application. The dose criterion for a Category 3 event should be dropped in view of the difficulties with dose estimates and the adequacy of the other criteria. The dose prediction capability of the shift supervisor should be supplemented/replaced by more elaborate computer-aided systems at the earliest possible phase of an emergency. These systems should cover all relevant exposure pathways and incorporate the results of environmental measurements as they become available.

The Province of Ontario and the municipalities of Durham and Toronto have a joint responsibility to protect the public in the event of an emergency at PNGS. The Federal Nuclear Emergency Control Centre provides assistance at the request of the province and co-ordinates any emergency response between provinces or with other countries. These responsibilities and interfaces are clearly established in the emergency plans, although some of these are still in a draft version and have not yet been authorized.

It was observed that the protective action levels (PALs) established in the provincial emergency plan are lower than those recommended by ICRP and IAEA. This may be prudent regarding radiation risks only, but the resulting conventional risks, through the disruption of normal life for large numbers of people, may be higher.

#### **Facilities, equipment and resources**

The provincial, municipal and corporate emergency control centres in Toronto, Whitby and at the PNGS site are adequately equipped and have sufficient space. They can be made operational within an acceptable time.

In addition to the planned dose rate surveys, provision should be made to monitor the air contamination inside the site emergency control rooms. There is only a minimum of meteorological instrumentation available on-site, corresponding to simple dispersion models. A site specific validated dispersion model should be used incorporating washout and fallout effects. To provide the required input, the meteorological instrumentation should be supplemented accordingly.

Ontario Hydro's plans for improving the environmental monitoring equipment are endorsed, especially for the survey vehicle and the permanent survey stations, to enable faster surveys and more representative assessment of possible soil, air, food and water contamination in the event of a severe accident.

It was noted that there is no capability at present to monitor radioactive releases in the direction of the lake in the event of a severe

accident. Devices to measure any radioactive release in this direction should be installed.

#### **Training, drills and exercises**

A training programme exists for developing and maintaining the necessary knowledge and skills for the plant personnel to respond to station emergencies. Common exercises were performed in 1986 and are well documented. The results were evaluated, and deficiencies identified and corrected within an acceptable period. Integrated on-site/off-site exercises are planned by Ontario Hydro to be carried out on a two-yearly basis for all the nuclear generating stations of the utility in turn. This is endorsed provided that the interval between exercises at PNGS does not exceed six years. Drills involving individual emergency response units such as the municipal fire brigade in Pickering should be conducted more frequently. Participation of the fire brigade in the yearly exercises at PNGS is recommended.

#### ACKNOWLEDGEMENTS

The Canadian Government, the Atomic Energy Control Board, the Province of Ontario, and Ontario Hydro through its Toronto Head Office and its Pickering Nuclear Generating Station provided valuable support to the Pickering OSART. The traditionally close co-operation of Canada with the IAEA in various nuclear safety activities, particularly the Nuclear Safety Standards (NUSS) Programme, the Incident Reporting System (IRS) and the Performance Indicator Project, and the contributions to the OSART Programme through the provision of experts to earlier missions and assistance in developing guidelines had already established a good basis. Well prepared information was made available in advance to familiarize the OSART members with their assignments. The spirit of openness, co-operation and hospitality, which extended beyond working hours, was outstanding and established ties between the reviewers and their counterparts which will endure in the future. The efforts of the liaison officer and the secretarial support were outstanding throughout the whole period. The Pickering OSART wishes to express its gratitude to all concerned for the prior efforts and for the excellent working conditions the OSART enjoyed at Pickering.

**ANNEX I: PICKERING TEAM COMPOSITION**

**CAREY, James - UK**  
Central Electricity Generating Board  
Health and Safety Department  
20 years of nuclear experience  
Review area: Operations I

**DENTON, Robert - USA**  
Baltimore Gas & Electric Co.  
Corporate Planning Department  
22 years of nuclear experience  
Review area: Management, Organization and Administration

**FRANZEN, Ferdinand L. - IAEA**  
Division of Nuclear Safety  
31 years of nuclear experience  
Team Leader

**HAYDIN, Michael - IAEA**  
Division of Nuclear Safety  
20 years of nuclear experience  
Assistant Team Leader

**HILLE, Ralf - Germany, Fed. Rep.**  
Nuclear Research Centre Jülich  
Division of Safety and Radiation Protection  
13 years of nuclear experience  
Review area: Emergency Planning and Preparedness

**HOENSCH, Volker - Germany, Fed. Rep.**  
Joint Nuclear Power Plant Neckar  
Quality Assurance Department  
17 years of nuclear experience  
Review area: Maintenance

**KITABATA, Takuya - Japan**  
Power Reactor and Nuclear Fuel Development Corporation  
FUGEN Nuclear Power Station  
9 years of nuclear experience  
Review area: Chemistry

**LAAKSONEN, Jukka - IAEA**  
Division of Nuclear Safety  
15 years of nuclear experience  
Review area: Technical Support I

**PALABRICA, Ricardo - IAEA**  
Division of Nuclear Safety  
19 years of nuclear experience  
Review area: Training and Qualification



**ANNEX I: PICKERING TEAM COMPOSITION (Continued)**

**ROUXEL, Jacques - France**  
Electricité de France  
Thermal Generation Division  
24 years of nuclear experience  
Review area: Operations II

**THOMAS, Bernard - IAEA**  
Division of Nuclear Safety  
19 years of nuclear experience  
Review area: Technical Support II

**TOUZET, Rodolfo E. - Argentina**  
National Atomic Energy Commission  
Department of Operating Nuclear Power Plants  
27 years of nuclear experience  
Review area: Radiation Protection

**Scientific Visitors (Observers)**

**LEE, Jong-Chan - Korea, Rep. of**  
Korea Advanced Energy Research Institute  
Nuclear Safety Centre  
18 years of nuclear experience  
Training area: Management, Organization and Administration

**POPA, Petru - Romania**  
State Committee for Nuclear Energy  
State Inspectorate for the Control of Nuclear Activities and Quality Assurance  
25 years of nuclear experience  
Training area: Operation

**ANNEX II: SCHEDULE OF ACTIVITIES**

- |    |   |                     |
|----|---|---------------------|
| 1. | Official request of the Resident Representative of Canada to the IAEA inviting an OSART | 2 February 1987     |
| 2. | IAEA confirmation of the OSART mission  | 26 February 1987    |
| 3. | Recruitment of external experts   | 2 March-12 May 1987 |
| 4. | Preparatory meeting with Ontario Hydro on mission arrangements                          | 21 April 1987       |
| 5. | Operational Safety Review of the Pickering Nuclear Generating Station                   | 1-19 June 1987      |
| 6. | Submission of summary report  | August 1987         |

## Appendix III.3

### Ontario Nuclear Safety Review Assessment of Ontario Hydro's Response to the IAEA/OSART Recommendations

by  
W.J. Keough

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## **A. Introduction**

1. The Operational Safety Review Team (OSART) of the International Atomic Energy Agency (IAEA) visited the Pickering Nuclear Generating Station (NGS) in June 1987, at the request of the Government of Canada, and submitted a report to the Government of Canada in August 1987 on the findings of the review.
2. The OSART report has been reviewed by the Nuclear Generation Division (NGD) of Ontario Hydro, and a co-ordinated follow-up plan has been organised to respond to the recommendations of the report.
3. Although there is no formal requirement for a documented response to the OSART recommendations, it is the intention of Ontario Hydro to submit a follow-up report, through proper government agencies, to the IAEA.
4. As part of the Ontario Nuclear Safety Review (ONSR) activities, we have reviewed the OSART report and the follow-up plan initiated by Ontario Hydro's NGD. The body of this report summarises the activities planned by the operating organisation at the Pickering station and comments on the key areas of response to the OSART recommendations.

## **B. Ontario Hydro Follow-up Plan**

### **(a) Organisation**

5. The responsibility for organising and co-ordinating follow-up work lies with the Pickering Station Manager. A supervising engineer in the Technical Section has been appointed to co-ordinate the work and to arrange for the necessary support activities of head office divisions.
6. OSART recommendations have been identified and recorded. Priorities have been assigned to each recommendation, work teams have been assigned,

and scheduled completion dates and costs have been estimated. The work plan summaries are shown in Attachments I and II.

7. Several of the recommendations in the OSART report make reference to International Commission on Radiological Protection (ICRP) standards that have been recommended for adoption by member states but have not been officially accepted by the state regulatory agencies. In the instances in which this situation applies to Ontario Hydro, Ontario Hydro is working closely with the Atomic Energy Control Board (AECB) to formulate a response and a plan.

**(b) Categories of recommendations**

8. The review team identified 79 areas of operations in which improvements or alternative procedures should be considered. In this total, there were eight recommendations that could be combined with other action plans, so the total number of work categories is 71.

9. Recommendations have been defined and separated into the work categories shown in Table III.3-1.

**(c) Priorities**

10. Several of the recommendations in the OSART report dealt with areas that were already being addressed by Ontario Hydro and actions that have been completed. For the remaining work areas, Ontario Hydro has established priority rankings shown in Table III.3-2.

11. The vast majority of jobs are scheduled for completion in 1988. A few, which require extensive approval of procedure changes, will be complete in the first quarter of 1989.

Table III.3-1

Categories of Follow-up Plan

<u>Category</u>	<u>No. of jobs</u>
Action--direct result of OSART	37
Recommendation already known and action committed	21
Recommendation accepted--no additional action	3
Recommendation not accepted	<u>10</u>
Total recommendations	71

Table III.3-2

Priority Ranks of Work Categories

<u>Priority rank</u>	<u>No. of jobs</u>
Must do	15
High	26
Medium	<u>11</u>
Total priority jobs	52

**(d) Cost of follow-up action**

12. Ontario Hydro has identified an immediate need for \$200 000 to develop work plans and firm estimates for improvements that are planned. The expenditure of capital funds for new equipment and facilities will be handled in the normal course of capital budget development. Although no accurate estimate of the cost of implementing the OSART recommendations is available, it would appear that the cost will be in excess of \$2 million.

**C. Comments on Follow-up Plan**

13. We do not attempt to comment on each item of follow-up action planned by Ontario Hydro, but have selected those areas that are of most direct interest to the ONSR. The following areas were identified for attention by the ONSR team as well as by OSART, and the comments on action plans reflect the ONSR representative's assessment of the Ontario Hydro response.

**(a) Organisation and schedule**

14. Ontario Hydro has responded quickly and professionally to the recommendations of the OSART. Work plans have been established, and adequate resources are available to develop revisions to operating practices that have been accepted by Ontario Hydro. Schedules appear to be appropriate and attainable, and priorities have been thoughtfully selected.

**(b) Surveillance procedures and operating experience feedback**

15. The Radioactivity Management and Environmental Protection Department (RMEP) has been assigned the task of reviewing and responding to the recommendations concerning human performance analysis, Significant Event Report (SER) analysis, surveillance procedures, and result analysis. It is very appropriate to have these job functions combined in a single review, and RMEP has been keenly interested in this area of operation in recent months. We

would expect that a significant improvement in human factor analysis and performance should result from the OSART recommendations. Ontario Hydro management should give particular attention to the action plan that is ultimately put forth for this very important area of operational safety.

**(c) Radiation protection**

16. The ICRP/IAEA current recommendations on radiation protection procedures are strongly supported by OSART. Ontario Hydro has been urged to implement the new procedures and dose limits. However, it is not within the terms of the operating licence for Ontario Hydro to change the procedures or dose limits, and any change that is implemented must be directed by AECB. It is understood that AECB is reviewing the present radiation control procedures and assessing the need to incorporate new ICRP standards. A definitive timetable should be set for this decision.

17. A recommendation to improve the containment radiological monitoring equipment has received immediate attention. The design section of Ontario Hydro has prepared a post-LOCA (loss of coolant accident) report, and the recommendations from this report are being reviewed.

18. OSART recommended the installation of continuous radiological monitors on service water effluents. Ontario Hydro does not agree that additional protection is required. They are confident that the upstream radiological monitoring of heavy water and the batch monitoring of service water are completely adequate to identify any potential leakage of radiological material to the environment from the service water system.

**(d) Training and qualification**

19. OSART recommended a substantial upgrading of the refresher training programme for licensed operators and also proposed a requalification test procedure. Ontario Hydro has responded to the suggestions for refresher



training and is discussing a programme with AECB. The programme does not include a test procedure for formal requalification of an operator's licence.

20. The OSART reported that the Pickering A NGS simulator did not have the capability to simulate a boiler tube failure situation. This has been corrected, and future training will include boiler tube failure simulations.

#### **(e) Conduct of operations**

21. The Shift Supervisor position received careful review by the OSART, and there were several recommendations concerning work-load, tenure, and transfer communications. Ontario Hydro is in general agreement with the recommendations and has organised a detailed review of the Shift Supervisor and other operating positions. The subject of tenure had been addressed earlier, and the Shift Supervisor is currently remaining in the position for a longer career period.

22. Operating Memos, jumpers, and stickers need to be cleaned up. Ontario Hydro has a review under development, but does not have specific target levels of acceptability established. This area needs a very determined action plan, and Ontario Hydro should set demanding targets for reducing the present backlogs.

23. More simulations of operations from the Unit Emergency Control Centre (UECC) were recommended. Ontario Hydro agrees and has scheduled more training for emergency condition transfer of control to the UECC.

#### **(f) Maintenance**

24. OSART tended to focus on the maintenance philosophy and organisation and identified several areas that required strengthening. They did not feel that predictive maintenance techniques had been adequately developed. OSART felt that corrective maintenance effectiveness should be improved in order to reduce the number of deficiency reports. They also recommended improvements in

equipment history records, better job transfer between shifts, more predictive analysis, and a reduction in the backlog of work orders.

25. The brief ONSR examination of maintenance organisation and procedures supports the OSART findings.

26. The apparent philosophy of maintenance in NGD is that the maintenance department should play a reactive role: when something fails, it fixes it. They do not have adequate technical resources to predict failure. This responsibility rests in other station and head office technical functions. The technical support for maintenance at the plant level resides in the systems network of the Technical Section. We believe that the overall maintenance function needs more direct technical support to work effectively.

27. Ontario Hydro NGD does not appear to accept the suggestions for restructuring the maintenance function, nor does it appear to consider the maintenance problems with any degree of urgency. It is important to take immediate positive actions on the OSART recommendations. The maintenance functions in NGD should be structured and staffed to be a pro-active organisation that is capable of identifying incipient failures in equipment and that has the capability to develop advanced techniques for equipment reliability evaluation.

#### **D. Summary**

28. Ontario Hydro has responded to the OSART recommendations with a well-organised follow-up plan. They have committed adequate resources to address the recommendations and develop appropriate follow-up actions.

29. There are three key areas in which we would recommend that Ontario Hydro management should carefully and critically assess the actions that are adopted to respond to the OSART recommendations. These three areas are:

- surveillance procedures and operating experience feedback;
- conduct of operations--reduction of Operating Memos, jumpers, and stickers; and
- maintenance--philosophy and organisation.

O S A R T F O L L O W - U P R E P O R T

MANAGEMENT

1. DOCUMENT CONTROL .....

TRAINING

- 2. CONSISTENCY OF TRAINING RELATED INFO-DOCUMENT CONTROLS
- 3. TRAINING RECORDS
- 4. ENTC IMPROVEMENTS
- 5. REFRESHER
- 6. AECB REGULATION FOR REQUALIFICATION
- 7. COMPLETION OF MAINTENANCE TRAINING

OPERATORS

- 8. SS TWO-YEAR TENURE
- 9. FORMAL REVIEW OF ADMINISTRATION PROCEDURES
- 10. FORMALIZE TURNOVERS
- 11. REGULATE CONTROL ROOM ACCESS
- 12. MULTI UNIT RESTART WORK OVERLOAD
- 13. FLOWSHEET MODIFICATION CONTROL
- 14. SIGN OFF OF OPERATING MANUAL
- 15. LARGE NUMBER OF OPERATING MEMOS
- 16. JUMPER RECORDS
- 17. STICKERS ON PANELS
- 18. FIELD APPENDED INFO
- 19. OPERATION ON WRONG UNIT
- 20. EMERGENCY OPERATING PROCEDURES AND EARTHQUAKE SENSORS
- 21. CONTROL ROOM AIRBORNE RADIATION MONITOR
- 22. UECC IMPROVEMENTS IN PROCEDURES

MAINTENANCE

- 23. CAPABILITY OF SHIFT FOR HOUSEKEEPING, ETC
- 24. LACK OF ENGINEERING SUPPORT
- 25. POOR MATERIAL CONDITION
- 26. MANPOWER INTENSIVE PREVENTIVE MAINTENANCE
- 27. WORK AUTHORIZATION DELAYS
- 28. MAINTENANCE BACKLOG AND WORK PLANNING
- 29. SEGREGATE SAFETY AND PROCESS EQUIPMENT IN STORES
- 30. MAINTENANCE HISTORY AND PROGRAM DESCRIPTION
- 31. FOREIGN BODY INGRESS TO EQUIPMENT

	N	A	K	D	R	C
1.			X			
2.			X			X
3.			X		X	
4.			X	X	X	
5.				X	X	
6.	X					N/A
7.			X		X	
8.						
9.				X	X	
10.				X	X	
11.				X	X	
12.		X				N/A
13.			X		X	
14.				X	X	
15.			X		X	
16.			X		X	
17.				X	X	
18.				X	X	
19.				X	X	
20.				X	X	
21.				X	X	
22.			X	X	X	
23.						
24.		X				N/A
25.			X			N/A
26.			X			N/A
27.				24		N/A
28.	X					N/A
29.				X	X	N/A
30.	X					N/A
31.				X	X	

TECHNICAL SUPPORT

- 32. HARMONIZE SYSTEM ENGINEERS (S) .....
- 33. ADD TECHNICAL MANPOWER .....
- 34. ADMINISTRATION OF SURVEILLANCE PROGRAM .....
- 35. DECAY HEAT REMOVAL TESTS (S) .....
- 36. IMPROVE DESIGN FOR DECAY HEAT TESTING (S) .....
- 37. TEST ACCEPTANCE CRITERIA .....
- 38. RECORDING AND REVIEW OF TEST DATA .....
- 39. RECORD FILING AND RETENTION .....
- 40. NEW FUEL HANDLING PROCEDURES .....
- 41. MORE COMPUTER MANPOWER (S) .....
- 42. REPLACEMENT OF IBM 1800 (S) .....

OPERATOR EX FEEDBACK

- 43. NEED FOR CENTRALIZE ANALYSIS OF EVENTS .....
- 44. SER QUALITY AND ROOT CAUSES .....
- 45. GENERIC SAFETY ISSUES FROM PERFORMANCE INDICATORS .....
- 46. 41% HUMAN FAILURES .....
- 47. FAULT DETECTION BY SURVEILLANCE .....
- 48. METHODOLOGY FOR SER CORRECTIVE ACTIONS .....
- 49. TOO MANY AECB ACTION MODIFICATIONS .....

RADIATION PROTECTION

- 50. CURRENT ICRP VALUES - PUBLIC .....
- 51. CURRENT ICRP CONCEPTS - WORKER .....
- 52. PREGNANT WORKER LIMITS .....
- 53. CRITICAL ORGAN DOSE LIMIT CONCEPT .....
- 54. HEALTH PHYSICS TO REVEIW RWP FOR LOWER DOSE RATES .....
- 55. IMPROVE REACTOR BUILDING RADIATION MONITORING .....
- 56. HEAT EXCHANGER ON LAKE WATER COOLING MONITORING .....
- 57. SOLID WASTE STORAGE IMPROVEMENTS .....

CHEMISTRY

- 58. REPLACE OLD LAB EQUIPMENT (S) .....
- 59. POTENTIAL FOR EQUIPMENT CONTAMINATION .....
- 60. FUMEHOOD VENTILATION INDICATORS .....
- 61. INSTALLATION OF POST ACCIDENT SAMPLING .....
- 62. IMPROVE SAMPLING FLUSHNG TIME .....
- 63. FEEDWATER IN-LINE MONITORS DEFECTIVE .....
- 64. HIGH I-131 CONCENTRATION IN HEAT TRANSPORT .....
- 65. NEED TO INPUT ALL DATA TO CPI .....
- 66. DAILY GRAPHICS TREND OF CHEMICAL ANALYSIS (S) .....

		S E E	34		
			X		X
			X		
				X	
		S E E	35		
			X		
				X	
			X		X
			X		
		S E E	33		N/A
		X			N/A
				X	
			X		
		S E E	44		
		S E E	44		
		S E E	44		
		S E E	44		
X					N/A
X					N/A
X					N/A
X					N/A
			X		
				X	
X				X	N/A
			X		
			X		
				X	
				X	
X				X	N/A
			X		
			X		
				X	



OSART FOLLOW-UP ACTION LISTING

FINDING	ACTION PROPOSED	WORK GROUP		PRIORITY (CONSEQx PROBxKRA)	COST k\$	TARGET DATE	IF ACTION PREV IDENTIFIED REF DOC IS	RESOURCES REQUIRED (AVAILABLE ?)
		INT	EXT					
DOCUMENT CONTROL - HIERARCHY NOT USED AS INTENDED. SIS ADDRESS PROCEDURE NOT POLICY. PROCEDURES REDUNDANT, OUTDATED.	IMPLEMENT "DOCUMENT RATIONALIZATION PROGRAM" AND APPOINT COORDINATOR. WORK ACTIVELY PROCEEDING.	LOBBEZOO		HIGH	200	90 01	QA "1987 WORK PROGRAM"	COORDINATOR YES
INCONSISTENCY IN DOCUMENTATION OF TRAINING INFORMATION IN P-SI, P-SRP, P-PSP, ETC, DUPLICATION EXISTS.	REVIEW AND CLASSIFICATION OF RELATED DOCUMENTS.	TRNG		HIGH	15	COMPLETE 88 01	RMEP AUDIT QQA-85-1	YES
COMPUTERIZED TRAINING RECORD SYSTEM IS INADEQUATE AND NOT USER FRIENDLY.	EXTEND SYSTEM TO INCLUDE ALL NECESSARY RECORDS AND IMPROVE USABILITY. MANUAL SYSTEM - NO ACTION PLANNED.					N/A	MEETING 86 12 15 R. SUE (CNTD) FILE P-09500	NOT AVAILABLE
IMPROVE ENTC INSTRUCTORS TO PRACTICE UNIT OPERATION; IMPROVE SIMULATOR; COMPLETE CHECKOUT GUIDES AND TRAINING MANUALS.	INSTRUCTOR TO RETURN TO SHIFT; COMPLETE GUIDES AND MANUALS. SIMULATOR IMPROVEMENT COMPLETED.	TRNG	ENTC	HIGH	10	89 06		YES
OPERATOR REFRESHER TRAINING IS NOT PLANNED AND CONDUCTED IN A SYSTEMATIC MANNER.	DOCUMENT TRAINING REQUIREMENTS AND THE SYSTEM OF EVALUATION. DRAFT SUBMITTED TO AECB (88 01 15).	PROD-B	ENTC	MUST	60	88 12		YES
NO REGULATORY GUIDES FOR AUTHORIZATION OF UNIT FIRST OPERATOR AND SHIFT SUPERVISOR.			AECB	N/A	N/A	N/A	N/A	AECB

OSART FOLLOW-UP ACTION LISTING

FINDING	ACTION PROPOSED	WORK GROUP		PRIORITY (CONSEQx PROBxKRA)	COST k\$	TARGET DATE	IF ACTION PREV IDENTIFIED REF DOC IS	RESOURCES REQUIRED (AVAILABLE ?)
		INT	EXT					
INCOMPLETE MAINTAINER TRAINING.	COMPLETE NECESSARY TRAINING.	TRNG	ENTC	HIGH		88 12	RMEP AUDIT COA-85-1	YES
TWO YEAR TENURE FOR SHIFT SUPERVISOR IS INADEQUATE.	POLICY TO BE WRITTEN AND ISSUED.	PROD-B				88 12		
NO REVIEW OF ADMIN PROCEDURES.	TO BE REVIEWED TO ASSESS NEED FOR A PROGRAM.	LOBBEZOO				88 06		
FORMALIZATION OF: 1. SHIFT SUPERVISOR (SS).	1. SS TURNOVERS TO BE FORMALIZED.	PROD-B		MEDIUM		COMPLETE 87 09 01		
FORMALIZATION OF: 2. FIELD TURNOVERS.	2. FIELD TURNOVER REQUIREMENTS TO BE REVIEWED.	PROD-OP		MEDIUM		88 06		
REGULATE CONTROL ROOM ACCESS.	TO BE REVIEWED.	PROD-OP		MEDIUM		88 09 01		
WORK OVERLOAD FROM MULTI-UNIT RESTART IN SHORT TIME.	AGREED. NO ACTION NEEDED.							
EFFORT NEEDED TO REDUCE FLOWSHEET BACKLOG.	1. ACTION UNDERWAY TO INTRODUCE CAD SYSTEM ON TRACK.	TECH-IC	ISO	MUST	250	88 12	PREVIOUS AUDITS	NO
HANDWRITTEN FIELD MARK-UP OF FLOWSHEETS NOT CONTROLLED.	2. STANDARD TO BE REVIEWED.	QA		MUST		88 02 01		



OSART FOLLOW-UP ACTION LISTING

FINDING	ACTION PROPOSED	WORK GROUP		PRIORITY (CONSLQx PROBxKRA)	COST k\$	TARGET DATE	IF ACTION PREV IDENTIFIED REF DOC IS	RESOURCES REQUIRED (AVAILABLE ?)
		INT	EXT					
DELAYS FOUND IN "READ AND SIGN" PROCEDURE FOR O.M. REVISIONS.	REVIEW.	PROD-B				88 12		
EFFORT NEEDED TO REDUCE NUMBER OF OPERATING MEMOS AND JUMPER RECORDS.	MONITORING SYSTEM FOR JUMPERS IN PLACE AND NUMBER REDUCING.	TECH-IC PROD-B		MUST		IN PROGRESS	PREVIOUS AUDITS; AECB ACTION ITEM	YES
STICKERS ON PANELS.	INSTRUCTIONS BEING PREPARED.	PROD-OP		HIGH		88 04 01		
OPERATING INSTRUCTIONS FOUND MOUNTED IN FIELD.	CALL-UP ACTION IN PLACE TO ENSURE CONTROL. INSTRUCTIONS TO BE PREPARED.	PROD-OP				88 04 01		
INADVERTENT OPERATIONS ON WRONG UNIT.	REVIEW NOW IN PROGRESS.	PROD-B		HIGH		88 06		
AIM PROCEDURE FOR POST EARTHQUAKE OPERATION NEEDS TO BE REVIEWED. A CONTROL ROOM SEISMIC DETECTOR MAY BE REQUIRED.	REVIEW COMPLETED AND RESPONSE IN PREPARATION.	TECH-IC	PICK ENG	HIGH	100	88 03 01		
REVIEW THE REQUIREMENT FOR AIRBORNE CONTAMINATION MONITORING IN THE MAIN CONTROL ROOM.	REVIEW PRESENT INSTRUCTIONS WITH RESPECT TO USE OF PORTABLE INSTRUMENTS IN THE CONTROL ROOM. AFTER THIS REVIEW A DECISION WILL BE MADE WITH RESPECT TO THE REQUIREMENT TO HAVE PERMANENT EQUIPMENT. NOT STARTED.	TECH-IC	PICK ENG	MUST  (IF INSTRUMENT PURCHASE REQ'D)	100	88 12 01		ONE MAN-MONTH PICK ENG AND 2 MAN-WEEKS TECH

OSART FOLLOW-UP ACTION LISTING

FINDING	ACTION PROPOSED	WORK GROUP		PRIORITY (CONSEQx PROBxKRA)	COST k\$	TARGET DATE	IF ACTION PREV IDENTIFIED REF DOC IS	RESOURCES REQUIRED (AVAILABLE ?)
		INT	EXT					
1. SECONDARY CONTROL CENTRE NOT SOUND PROOF.	REVIEW AND DETERMINE ACTION PLAN.	PROD-B		HIGH		88 12	PREVIOUSLY UNDER REVIEW	ONE MAN-YEAR TECH
2. DOCUMENTATION NOT PROVIDED.	DONE.	PROD-OP				COMPLETE 88 01 01		
3. COMMUNICATION SYSTEM PA IS NOT SEISMICALLY QUALIFIED.	REQUEST I&C TO REVIEW.	TECH-IC				88 12		
4. PERFORM FULL SCOPE DRILLS.	REVIEW DRILLS.	PROD-OP				88 08 01		
DIVISION OF RESPONSIBILITY FOR MAINTENANCE/HOUSEKEEPING WITHIN PRODUCTION.	ONGOING REVIEW. NO SPECIFIC ACTION REQUIRED.							
ENGINEERING SUPPORT.	PILOT PROGRAM BEING INITIATED BY PRODUCTION SECTION/TECHNICAL SECTION WITH OBJECTIVE TO REVIEW EXISTING PM/BM MAINTENANCE ON SPECIFIC COMPONENTS TO OPTIMIZE MAINTENANCE ACTIVITIES. TEAM FORMED AND WORK IN PROGRESS.	PROD-M TECH	CNS	MUST	100	88 08		MAINTENANCE SUPPORT/ TECHNICAL SECTION/ TTSD STA
POOR MATERIAL CONDITIONS. AECB WILL TIE THIS INTO 4.1.1.	ONGOING MANAGEMENT COMMITMENT TO IMPROVE CONDITIONS TO THE EXTENT POSSIBLE WITH AVAILABLE RESOURCES AND FUNDS.	PROD-M TECH		HIGH		ONGOING		

OSART FOLLOW-UP ACTION LISTING

FINDING	ACTION PROPOSED	WORK GROUP		PRIORITY (CONSEQx PROBxKRA)	COST k\$	TARGET DATE	IF ACTION PREV IDENTIFIED REF DOC IS	RESOURCES REQUIRED (AVAILABLE ?)
		INT	EXT					
PM PROGRAM.	SEE 4.1.2.							
WORK CONTROL PROCESS.	NONE.							
PLANNING AND WORK SCHEDULING. OSART FU FINDING 4.5.1.	REVIEW COMPLETE. MEMO W.E. LAING TO E. DEWAR, FILE P-00531, DATED 87 12 17. MONITOR IN 1989.	PLAN		N/A	N/A	89 WMS	N/A	N/A
LACK OF SEGREGATION BETWEEN SAFETY AND PROCESS COMPONENTS.	SEGREGATION AT PICKERING MGS IS BASED ON MATERIAL REQUIREMENTS NOT ULTIMATE USE. ACCESS TO STORES IS TIGHTLY CONTROLLED REDUCING THE PROBABILITY OF ANY ERRORS. NO ACTION REQUIRED.							
MAINTENANCE HISTORY.	INTRODUCTION OF WORK MANAGEMENT SYSTEM.	PROD-M TECH	ISO	MUST		88 10		
FOREIGN MATERIAL. INGRESS TO OPENED EQUIPMENT.	MEMO TO SMS FOR INPUT AND AUDIT TO BE CONDUCTED.	PROD-M		HIGH		89 02		
HARMONIZE APPROACH OF SYSTEM ENGINEERS.	SEE 5.2.1.							

OSART FOLLOW-UP ACTION LISTING

FINDING	ACTION PROPOSED	WORK GROUP		PRIORITY (CONSEQ PROBxKRA)	COST k\$	TARGET DATE	IF ACTION PREV IDENTIFIED REF DOC IS	RESOURCES REQUIRED (AVAILABLE ?)
		INT	EXT					
CONSIDER NEED FOR ADDITIONAL TECHNICAL MANPOWER TO ADDRESS INCREASING BACKLOG OF WORK.	A REVIEW OF STAFFING LEVEL HAS ALREADY BEEN COMPLETED INDICATING ADJUSTMENTS IN SOME AREAS. A COMPUTERIZED MODEL HAS BEEN DEVELOPED TO SUPPORT STAFFING LEVELS. STAFF LEVELS ARE PRESENTLY BUDGET CONSTRAINED. ACTION COMPLETE.	TECH-IC		HIGH		COMPLETE 88 01 15		
CONSOLIDATE STATION SURVEILLANCE PROCEDURES.	GENERATE AN SRP OUTLINING STATION SURVEILLANCE REQUIREMENTS AND UPDATE EXISTING SURVEILLANCE PROCEDURES. WORK TOGETHER WITH CNS TO DEVELOP A SURVEILLANCE DRP. IN PROGRESS.	TECH-IC	CNS	MEDIUM	10 LAB ONLY	88 07	AECB QA AUDIT 86-QA-12-1	TECH 70 h CNS 70 h
ENHANCE TEST PROGRAMS ON HEATSINK RELATED SYSTEMS (AIM VOLUME 1 SECTION 4 PLUS RECIRCULATION COOLING AND INSTRUMENT AIR) TO A DEGREE CONSISTENT WITH SPECIAL SAFETY SYSTEMS. (S)	THIS NEEDS TO BE REVIEWED GENERICALLY. COMPLETION OF ACTIONS 5.2.5 WILL ENSURE SAFETY REQUIREMENTS ARE ADDRESSED. THIS THEN BECOMES A RELIABILITY/ECONOMIC ISSUE REQUIRING COST BENEFIT REVIEW. MANPOWER NOW DEDICATED TO BEGIN WORK IN APRIL.	TECH-PFC	CNS	MEDIUM	5	88 11 01		TECH 2000 h CNS 100 h
REVIEW DECAY HEAT SUPPORT SYSTEMS TO ENSURE ADEQUATE TESTABILITY. (S)	TO FOLLOW 5.2.2 IF REQUIRED.	TECH-PFC		MEDIUM	10	DEFERRED		TECH 70 h

OSART FOLLOW-UP ACTION LISTING

FINDING	ACTION PROPOSED	WORK GROUP		PRIORITY (CONSEQx PROBxKRA)	COST k\$	TARGET DATE	IF ACTION PREV IDENTIFIED REF DOC IS	RESOURCES REQUIRED (AVAILABLE ?)
		INT	EXT					
REVIEW ALL TEST PROCEDURES VS SADL AND COMMISSIONING RESULTS TO ENSURE ALL SAFETY ASSUMPTIONS ARE VALIDATED.	A SYSTEM BY SYSTEM REVIEW OF ASSUMPTIONS HIGHLIGHTED IN THE SADL DOCUMENTS WILL BE CONDUCTED AND ASSURANCE GIVEN THAT ADEQUATE TESTING IS IN PLACE. SEE 5.2.2.	TECH-PFC		MUST	50	88 03		TECH 1000 h
ENSURE TESTS HAVE ACCEPTANCE CRITERIA AND TOLERANCES.	PROVIDE TOLERANCES <u>ON ALL TESTS</u> AND CLEAR ACCEPTANCE CRITERIA TSP-39 TO BE REVISED TO REFLECT REQUIREMENT. PCA REJECTED BY AECB ON SS TEST TOLERANCES. WORK ONGOING.	TECH-IC	NSSD RMEP	HIGH	50	89 01 01	NUMEROUS SERs	TECH 2000 h NSSD 300 h RMEP 1000 h
SYSTEM ENGINEERS TO PERFORM PARAMETER TREND ANALYSIS.	THIS IS BEING DONE WHERE THERE IS A PERCEIVED BENEFIT, OR IN RESPONSE TO A PROBLEM. SELECTION OF IMPAIRMENT LEVELS HAS BEEN DONE SUCH THAT ACTION WILL TAKE PLACE TO ENSURE SAFETY REQUIREMENTS ARE MET. NO ACTION PROPOSED OTHER THAN 5.2.5.							
SYSTEM ENGINEERS DO NOT CURRENTLY REVIEW ALL TEST RESULTS. THIS SHOULD BE CORRECTED.	TESTS NOW ROUTED TO ENGINEERS FOR SURVEILLANCE.	TECH-IC				COMPLETED 87 10 06		

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OSART FOLLOW-UP ACTION LISTING

FINDING	ACTION PROPOSED	WORK GROUP		PRIORITY (CONSEQx PROBxKRA)	COST k\$	TARGET DATE	IF ACTION PREV IDENTIFIED REF DOC IS	RESOURCES REQUIRED (AVAILABLE ?)
		INT	EXT					
DEVELOP A STANDARD MINIMUM REQUIREMENT FOR TESTING.	TSP-39 TO BE UPDATED TO SPECIFY GENERIC TESTING REQUIREMENTS FOR SPECIFIC EQUIPMENT PARAMETERS.	TECH-IC		HIGH	2	89 01 01		TECH 35 h
THERE IS PRESENTLY NO EASILY TRACEABLE EQUIPMENT HISTORY CAPABILITY. THIS SHOULD BE IMPROVED.	THE INTRODUCTION OF THE COMPUTERIZED WORK MANAGEMENT SYSTEM WILL ADDRESS THIS CONCERN.						ALREADY IN PROGRESS	
TEST RECORDS WERE FOUND TO BE RETAINED FOR 18 MONTHS IN SOME CASES. THIS REPRESENTED ONLY ONE RECORD FOR AN ANNUAL TEST.	REVIEW RETENTION REQUIREMENTS FOR TEST RECORDS. REVISE SRP-8.5 AS APPROPRIATE. DR 582298	TECH-IC QA		MUST	2	88 03 01		TECH/BUS/QA 21 h
MANY TEST RECORDS COULD NOT BE FOUND IN RECORDS. IMPROVEMENT IN TEST RECORD RETRIEVABILITY IS REQUIRED.	BUSINESS SECTION TO ESTABLISH A PROGRAM TO ENSURE TEST RECORDS CAN BE RETRIEVED IN A RELIABLE MANNER. DR 582299	TECH-IC		MUST	10	88 07 01		BUSINESS 200 h
FUEL HANDLING/STORAGE FACILITIES AND PROCEDURES ARE NOT CONSISTENT WITH STANDARDS ELSEWHERE. CONSIDERATION SHOULD BE GIVEN FOR IMPROVEMENT.	REVIEW STORAGE/HANDLING PROVISIONS TO ENSURE CONSISTENCY WITH THE QA PROCESS APPLIED TO THE PRE SITE DELIVERY PROCESS.	TECH-FH PROD		MEDIUM	10	89 01		200 h
REVIEW COMPUTER GROUP STAFFING. (S)	INCORPORATED INTO ACTION 5.1.2.						SEE 5.1.2	

OSART FOLLOW-UP ACTION LISTING

FINDING	ACTION PROPOSED	WORK GROUP		PRIORITY	COST k\$	TARGET DATE	IF ACTION PREV IDENTIFIED REF DOC IS	RESOURCES REQUIRED (AVAILABLE ?)
		INT	EXT	(CONSEQx PROBxKRA)				
REVIEW PHILOSOPHY OF SUPPORT FOR IBM 1800. CONSIDER LONG-TERM REPLACEMENT.	CONDUCT FEASIBILITY STUDY IN 1989.							
SET UP A CENTRAL ANALYSIS GROUP FOR IN-DEPTH ANALYSIS OF SER FOR HUMAN AND TECHNICAL ASPECTS.	REQUEST RMEP TO REVIEW FINDING. LETTER WRITTEN.	TECH-PFC	RMEP	HIGH		NOT KNOWN		
SER ANALYSIS METHODOLOGY NEEDED TO ENSURE IDENTIFICATION OF ROOT CAUSES AND HUMAN FACTORS.	REQUEST RMEP TO DEVELOP GENERIC SER REVIEW METHODOLOGY.	TECH-PFC		HIGH	60 PER ANNUM	BB 12		
SER ANALYSIS METHODOLOGY NEEDED TO ENSURE IDENTIFICATION OF ROOT CAUSES AND HUMAN FACTORS.	PICKERING NGS TO IMPLEMENT HPES. COURSE BEING WRITTEN.	TRNG						
	SEE 6.2.1						N/A	
EVALUATE EFFECT OF MODIFICATIONS ON PLANT SAFETY.	ALREADY DONE.						N/A	
REVISE DERIVED EMISSION LIMITS TO PRODUCE A CORRESPONDING CHANGE IN PUBLIC DOSE LIMIT FROM 500 mrem/a TO 100 mrem/a.	NO ACTION.							

OSART FOLLOW-UP ACTION LISTING

FINDING	ACTION PROPOSED	WORK GROUP		PRIORITY (CONSEQx PROBxKRA)	COST k\$	TARGET DATE	IF ACTION PREV IDENTIFIED REF DOC IS	RESOURCES REQUIRED (AVAILABLE ?)
		INT	EXT					
CHANGE TO ICRP 26/30 METHODOLOGIES.	NO ACTION, UNTIL NEW AECB REGULATIONS ISSUED.							
TOTAL RESTRICTION ON ACCESS TO RADIATION AREA ONCE PREGNANCY HAS BEEN CONFIRMED.	NO ACTION. CURRENT MGD POLICY, IE, DRP-15.1 IS MORE CONSERVATIVE THAN REGULATORY REQUIREMENTS.							
ADMIN DOSE CONTROL BASED ON CRITICAL ORGAN.	NO ACTION, UNTIL NEW AECB REGULATIONS ISSUED.							
EXPAND SCOPE OF HEALTH PHYSICS UNIT IN RADIOACTIVE WORK PLANNING.	ISSUE SRP-7.9, "RADIOACTIVE WORK PLANNING AND EXECUTION".	HP		HIGH		88 04 01	AECB RADIATION CONTROL APPRAISALS	HP UNIT YES
IMPROVE CONTAINMENT RADIOLOGICAL MONITORING (PRE AND POST LOCA).	POST LOCA REVIEW REPORT ISSUED BY DESIGN IN DECEMBER. UNDER REVIEW.	TECH-1C		HIGH		90		
INSTALL APPROPRIATE IN-LINE MONITOR FOR CONTINUOUS MONITORING OF SERVICE WATER RADIOLOGICAL RELEASES.	NO ACTION REQUIRED. D <sub>2</sub> O MONITORING FOR ECONOMY SATISFIES ACUTE AND CHRONIC EMISSIONS REQUIREMENTS.					N/A	AECB AUDIT	



OSART FOLLOW-UP ACTION LISTING

FINDING	ACTION PROPOSED	WORK GROUP		PRIORITY (CONSEQx PROBxKRA)	COST k\$	TARGET DATE	IF ACTION PREV IDENTIFIED REF DOC IS	RESOURCES REQUIRED (AVAILABLE ?)
		INT	EXT					
IMPROVE FACILITIES IN SOLID WASTE STORAGE AREA: 1. INSTALL AIR MONITOR (CONTINUOUS) 2. MODIFY VENTILATION	1. HP TO EVALUATE AIR MONITORING NEEDS. 2. FUNDING AND RESOURCES NOT AVAILABLE. TO BE SOUGHT IN 1988.	HP		MEDIUM - HIGH	200	88 06 89 12		HP UNIT
IMPROVE FACILITIES IN SOLID WASTE 3. UPGRADE FIRE DETECTION/ PREVENTION/FIREFIGHTING	REVIEW SITUATION AND ISSUE EWR'S TO DESIGN ONCE FUNDING IS AVAILABLE.	PROD-M						
SOLID WASTE STORAGE. 4. PREVENT ACCUMULATION OF LARGE NUMBERS OF PLASTIC BAGS.	SITUATION BEING MONITORED TO DETERMINE NEED FOR ACTION.	PROD-M				88 03		
CONSIDERATION SHOULD BE GIVEN TO REPLACE CHEMICAL ANALYTICAL INSTRUMENTS ON A PLANNED BASIS.	CHEM LAB INSTRUMENTS ARE BEING REPLACED ON AN AS-REQUIRED BASIS. IMPLEMENTATION OF A PLANNED REPLACEMENT PROGRAM WILL BE REVIEWED.	TECH-PFC CHEM LAB		HIGH B4	50 PER ANNUM	ONGOING		
COUNTERMEASURES TO PROTECT SAMPLES AND DETECTOR FROM CONTAMINATION.	INSTITUTE PROCEDURAL CHANGES IN LABORATORY. CLP-1-0150-1 WRITTEN.	TECH-PFC		MUST C3	5	COMPLETE 88 01		
VENTILATION FLOW RATE THROUGH FUME HOODS INADEQUATE.	UPGRADE VENTILATION SYSTEM. EWR 88-176 ISSUED TO DESIGN TO REVIEW REQUIREMENTS.	TECH-MCD		MUST C1	130	89 12	MEMO PULLEN TO RYAN APRIL 1986	FUNDING BEING PURSUED

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OSART FOLLOW-UP ACTION LISTING

FINDING	ACTION PROPOSED	WORK GROUP		PRIORITY (CONSEQx PROBxKRA)	COST K\$	TARGET DATE	IF ACTION PREV IDENTIFIED REF DOC IS	RESOURCES REQUIRED (AVAILABLE ?)
		INT	EXT					
POST ACCIDENT SAMPLING SYSTEMS HAVE NOT BEEN INSTALLED.	REVIEW NEED FOR SYSTEMS AND DESIGN OPTIONS. REPORT IS READY FOR ISSUE TO AECB.	TECH-PFC		MUST C2	10	88 12		
PRESAMPLING LINE FLUSHING TIME WAS INADEQUATE WHEN FIELD WORK WITNESSED.	REVIEW WITH CHEMICAL TECHNICIANS THE NEED TO FOLLOW PROPER PROCEDURES. COMPLETED.	TECH-PFC		HIGH B4		COMPLETE 87 09		
FEEDWATER IN-LINE MONITORS ARE INOPERATIVE ON PICKERING NGS-B. MANUAL SAMPLING DONE INSTEAD.	PICKERING NGS-A DONE. COMPLETE INSTALLATION OF NEW IN-LINE INSTRUMENTATION ON PICKERING NGS-B IN PROGRESS.	TECH-PFC		HIGH B4	ISRL	88 06	PC 8680	
HEAT TRANSPORT SYSTEM IODINE I-131 CONCENTRATION HIGHER THAN IN LWR MAY RESULT FROM HIGH H <sub>2</sub> /O <sub>2</sub> RATIO.	CONFIRM ADEQUACY OF CURRENTLY SPECIFIED H <sub>2</sub> /O <sub>2</sub> RATIO. DONE. OK AS IS.	TECH-PFC		MUST C2	5	COMPLETE 88 01		
NOT ALL DATA MEASURED IS INPUT TO THE COMPUTER TO ALLOW ASSESSMENT OF TRENDS.	CONSIDER WAYS OF PROVIDING ADDITIONAL MANPOWER TO INPUT DATA. NOW BUDGETTED FOR 1989.	TECH-PFC		MEDIUM D4	10 PER ANNUM	89 06	MEMO FROM D.A. TUCKER TO S.E. WOLFE	
ADDITIONAL SUPPORT SHOULD BE OBTAINED VIA WRITTEN AGREEMENTS WITH EXTERNAL SOURCES TO PROVIDE OFF-SITE MONITORING AND MEASUREMENTS DURING THE EARLY PHASE OF ACCIDENTS, IE, BACKUP LABS.	REVIEW CURRENT AGREEMENT. SOLIDIFY AGREEMENT WITH SSD AND ENSURE THEY HAVE A BACKUP PROGRAM, IE, RESEARCH, SHOULD THEIR LAB BECOME CONTAMINATED. DARLINGTON NGS LAB TO BE UTILIZED WHEN STAFFED 24 H/DAY. WORK IN PROGRESS.	HP	SSD	MUST		88 06		

OSART FOLLOW-UP ACTION LISTING

FINDING	ACTION PROPOSED	WORK GROUP		PRIORITY (CONSEQx PROBxKRA)	COST k\$	TARGET DATE	IF ACTION PREV IDENTIFIED REF DOC IS	RESOURCES REQUIRED (AVAILABLE ?)
		INT	EXT					
CLARIFY/SIMPLIFY CATEGORIZATION OF EMERGENCY WHICH SS MUST DECIDE ON.	COMPLETED.	PROD-OP	RMEP	HIGH		COMPLETE 87 12 01		
UPGRADE DOSE PREDICTION (PUBLIC) FOR SS TO COMPUTER-AIDED SYSTEM.	CURRENTLY UNDER REVIEW - LIKELY WILL STREAMLINE PRESENT NORMAL CALCULATION METHOD AND NOT ADD COMPUTER SYSTEM IN PROGRESS. ONLY TRAINING TO DO.	PROD-OP		HIGH		88 03 01		
PALS ARE LOWER THAN RECOMMENDED BY ICRP/LAEA.	FOLLOW UP WITH RMEP.	PROD-OP	PROV	MEDIUM		88 04 01		
PROVIDE DOSE RATE METERS AND AIR CONTAMINATION METERS INSIDE EMERGENCY CENTRES.	REVIEW DONE. EQUIPMENT IS ORDERED.	PROD-OP		HIGH	8	88 03		
THE METEOROLOGICAL EQUIPMENT NEEDS UPDATING.	RMEP REPORT HAS SPECIFIED REQUIREMENTS. THIS IS BEING REVIEWED AT SITE.	TECH-IC		HIGH		89 01		
TLDs FOR PLUME ACTIVITY MEASUREMENTS SHOULD COVER LAKE SIDE OF SITE.	REVIEW IN PROGRESS BY RMEP.	PROD-OP	RMEP	MEDIUM		88 04 01		
UPGRADE PLUME TRACKING VEHICLE EQUIPMENT.	REVIEW TO BE DONE. SOME EQUIPMENT WILL BE TESTED AND PURCHASED.	PROD-OP	RMEP	HIGH	15	88 07 15		

OSART FOLLOW-UP ACTION LISTING

FINDING	ACTION PROPOSED	WORK GROUP		PRIORITY	COST k\$	TARGET DATE	IF ACTION PREV IDENTIFIED REF DOC IS	RESOURCES REQUIRED (AVAILABLE ?)
		INT	EXT	(CONSEQx PROBxKRA)				
IN-STATION ESCAPE/ASSEMBLY AREA ROUTES TO BE MARKED.	SIGNS BEING MOUNTED.	PROD-OP		HIGH		88 03	AECB AUDIT	
MUNICIPAL FIRE BRIGADES SHOULD PRACTICE WITH STATION DURING ANNUAL DRILLS/EXERCISES.	WILL BE DONE. FIRE BRIGADE TO BE INVOLVED IN OCT 1988 DRILL.	PROD-OP		MUST		88 10 01		
SEMIANNUAL COMMUNICATION OF EMERGENCY PLAN CHANGES.	NOW TIED INTO EMERGENCY CREW REFRESHER TRAINING DRILLS.	PROD-OP		MEDIUM		COMPLETE 87 12 01		

## Appendix IV

# The Canadian Approach to Nuclear Safety

by  
J.A.L. Robertson and D.G. Hurst\*

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\* Formerly President of the Atomic Energy Control Board.

### **Note by Commissioner**

There have been few comprehensive statements of the Canadian approach to nuclear safety, in spite of the leading role this country played in formulating the principles and methods on which safety systems should rest. Accordingly, I asked two prominent authorities to summarise the Canadian approach as it has governed the methods of the Atomic Energy Control Board and Atomic Energy of Canada Limited. This appendix is their considered statement.

**F.K.H.**

## **A. Introduction**

1. The exemplary safety record for the generation of nuclear electricity in Canada has been attributed to both the inherent design features of the Canada Deuterium Uranium (CANDU) reactors and the professional competence and safety consciousness of the utilities that operate them. An important factor that is often overlooked is the particular approach to reactor safety adopted in Canada and the fact that it was essentially in place before the first CANDU reactor was built.

2. This approach has much in common with those followed in other countries, partly because some of the features now widely accepted originated in Canada. Other original features, and the way they all are integrated into a coherent whole, render the approach uniquely Canadian.

## **B. Organisational Structure and Responsibilities**

3. In nuclear energy, as in any other activity, there is no absolute safety: equipment fails, humans make errors, and acts of God occur. One function of government is to enact laws and regulations providing reasonable safety to workers and the public and protecting the environment, despite these failures.

4. This basic policy received early public expression in the enactment in 1946 of the Atomic Energy Control Act, which declared atomic energy (more logically termed "nuclear energy") to be a federal responsibility and established the Atomic Energy Control Board (AECB) as the agency responsible for controlling atomic energy in Canada.

5. The Atomic Energy Control Act is a short document providing very broad, enabling legislation that gives extensive discretionary power to the AECB: it empowers the AECB to make regulations that, when approved by the Governor in Council, have the force of law. The regulations, of which the latest edition was issued in 1974, contain many specific requirements for safety, licensing,

radiation dose limits, etc. Other relevant legislation is the Nuclear Liability Act of 1976 concerning responsibility and compensation for nuclear damage in the event of a serious accident.

6. Because energy is in general a provincial matter, decisions on the need for nuclear energy, the amount to be provided, and siting of the power plants are provincial responsibilities, subject to overall regulation by the AECB.

7. That the agency responsible for regulating nuclear energy should be separate from those developing, promoting, and exploiting the technology is an important aspect of policy. In 1954, at the start of the CANDU programme, an amendment to the Atomic Energy Control Act made the AECB independent of other organisations with an interest in nuclear energy, in both the public and private sectors.

8. The AECB currently reports to Parliament through the Minister of Energy, Mines and Resources (but not through the Department of Energy, Mines and Resources, which is responsible for energy policy development). The AECB is subject to annual review by parliamentary committee.

9. The Board\* consists of a full-time President and four part-time members, supported by technical and administrative staff. The President is also the Chief Executive Officer with direct responsibility for the organisation and management of the AECB staff. The Board has determined that the AECB should be a technical body that makes technical decisions on health, safety, and security matters.

10. Although the five-member Board is ultimately responsible for all AECB decisions and reviews the more significant ones, many of the policy development, assessment, and surveillance functions are conducted by staff members, assisted by advisory committees with members drawn from without the

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\* In this appendix, "AECB" refers to the organisation; "Board" refers to the five-member board.



AECB, notably the Advisory Committee on Nuclear Safety and the Advisory Committee on Radiological Protection [1].

11. Originally, in accordance with then-current secrecy requirements for nuclear energy, the AECB conducted its business in a closed manner, with negligible public interaction. With the general trend towards more open government, more directly accountable to the public, the AECB has adopted the policy of being as open as possible, consistent with its other responsibilities.

12. Two principles resulting from the new policy of openness are that:

- the public should have access to all relevant information unless there is a justifiable reason to the contrary; and
- in developing new policies and regulations, the AECB should provide the means for input from both interested parties and the general public.

13. As a result:

- licences, as well as the referenced supporting documentation, staff recommendations, and Board decisions, are public documents;
- applicants have to justify claims for confidentiality of any of the information in their supporting material;
- policy statements, regulatory documents, and regulatory guides are publicly available;
- consultative documents are issued for public comment as part of the AECB process for developing new policies and regulations;
- the AECB maintains a public reading room for all these documents; and
- during the site acceptance stage, the applicant is required to announce publicly the intention to construct the plant and to hold public information meetings at which the public can express its views and ask questions.

14. Most provinces, under their environmental legislation, require public hearings on major projects, a requirement supported by the AECB.

15. Another important aspect of the Canadian approach is that the responsibility for the safe operation of nuclear plants rests squarely with the utility that operates them: the regulatory authority primarily sets safety objectives and some performance requirements and audits their achievements.

16. In exercising its regulatory responsibility, the AECB sets safety criteria that must be satisfied by the licensee, but does not specify the means for meeting the criteria. Regulatory requirements emphasise numerical safety goals and objectives and avoid giving specific design or operational rules; the AECB thus tries to avoid making regulations too detailed or prescriptive. The licence applicant has to submit, for AECB approval, its proposal on how to satisfy the criteria.

17. An essential function complementary to setting the requirements is the assessment of the application. Whereas this proceeds almost continuously throughout the lifetime of any project, it is focussed on three specific stages of the licensing process: site acceptance, construction approval, and the operating licence. The review of the design and safety analysis is performed by AECB staff, acting as advisers to the five-member Board. The AECB, in assessing the proposal, can demand from the applicant further evidence until it is convinced that the criteria will be satisfied. This requires that the AECB has available to it the necessary expertise and experience in nuclear technology, including experience in operating nuclear reactors.

18. The AECB avoids the specification of design methods and procedures, thus ensuring that it provides an independent review of the proposal and does not have a proprietorial interest in it, as might happen if the AECB had specified how things should be done. Also, this does not destroy incentives for the applicant to develop innovative approaches to improving safety--approaches that could be unintentionally excluded by set rules for how safety is to be achieved.

19. Similarly, in regulating the operation of nuclear plants, the AECB requires safety surveillance functions to be performed by the licensee, whereas it acts as an auditor in monitoring the surveillance, again providing an independent check that would otherwise not be present.

20. Appointment of operators and certain other key staff of the operating organisation requires approval by the AECB. In the approval process, the AECB reviews the training and experience of the nominees and further tests their qualifications by subjecting them to a set of five written examinations.

21. At each nuclear power station, the AECB has resident professional staff members who serve both as inspectors and as licensing advisers.

### C. Safety Requirements

22. The safety objective is most simply expressed as that of ensuring that the nuclear generation of electricity should be at least as safe as comparable industries and activities, in particular the production of electricity by practical alternatives, for both workers and the public. From this stem other objectives: to prevent pollution and other harm to the human environment, and to protect the interests of future generations.

23. The hazard to workers and the public that is specific to the nuclear generation of electricity is ionising radiation emitted by radioactive materials released from the reactor or from the discharged fuel. Control of risks arising from normal industrial hazards is achieved by adopting the relevant regulations and delegating enforcement to the appropriate regulating agency. For instance, the risks associated with conventional pressure vessels are handled in this manner. The AECB, however, retains overall control, which it exercises through construction approvals and operating licences. This document deals only with the radiation risks.

**(a) Risk**

24. In regulating nuclear energy for "reasonable safety," it is necessary to define and quantify the term "risk." The dictionary definition of risk is "exposure to possible harm," which is commonly quantified in the practice of risk assessment by the equation:

$$\text{Risk} = \text{Probability of Event} \times \text{Harmful Consequences of Event [2]}$$

In risk assessment, "consequences" are often fatalities or injuries, but there can be others, such as releases of toxic substances or their effects on humans. For nuclear energy, the "consequences" are normally doses of radiation, from which estimates of human harm and environmental damage can be made. Even when numerical values are not available, this equation can be useful in showing that reducing either a probability or a consequence will reduce the expectation of harm.

25. According to the risk equation, two events would have the same risk if their products of probability and consequences were the same: a low-probability event with very serious consequences could have the same risk as a much more probable event with lesser consequences. However, the simple concept of a single numerical value to describe the risk is complicated by what is known as "risk aversion" in our society. It appears that the public has a greater horror of low-probability disasters, e.g., aircraft crashes, than of less serious, more likely events with a similar, or even much greater, numerical risk, e.g., car accidents. Acceptance of risk aversion is inherent in setting the permissible risk from nuclear energy.

26. It is assumed that even though there are uncertainties in any estimate of risk, it is better to base decisions on what estimates are possible than to ignore them. The uncertainties can, to some extent, be allowed for by making conservative assumptions in performing the estimates.

**(b) Radiation risk**

27. In assessing the risk from radiation--the single hazard that most distinguishes the nuclear industry from other industries--the AECB closely follows recommendations developed by the International Commission on Radiological Protection. This independent body reviews and assesses internationally available information to make recommendations on practices and dose limits that are almost universally accepted.

28. Thus, no exposure to ionising radiation is justified unless there is a compensating benefit, and exposures should be as low as reasonably achievable, economic and social factors being considered (known as the ALARA principle) [3]. Implicit in this principle is the acceptance of cost/benefit analysis as one procedure for setting acceptable risk, but as yet this has not often proved feasible in practice. It is also assumed, as a prudent measure, that the harm is linearly proportional to the exposure, with no threshold (the linear hypothesis).

29. A logical consequence of the linear hypothesis is that any radiation dose, however small, involves some risk, even if the dose is small compared with the normal variation in natural background radiation to which we are all exposed. A possibility, under discussion but not yet agreed upon, is that there exists a level of risk, and hence of radiation dose, so small in comparison with other risks to which we are inevitably exposed that events causing a lesser risk need not be considered in risk assessment (the de minimis principle).

30. For regulation of nuclear hazards, the AECB's general safety objectives are translated into a requirement that reference dose limits for both individuals and populations not be exceeded. For routine operation of nuclear plants, including such failures as might be expected to occur from year to year, these regulatory limits are set so that the risks to the most exposed members of the public, and to the surrounding population, are acceptable. In recognition of the ALARA principle, operating targets for maximum releases of radioactive materials from plants are set at 1% of the regulatory limits.

31. Even in the event of an accident such as might be expected no more often than once per 3000 reactor-years, the limits for individuals were chosen as those judged tolerable for a "once-in-a-lifetime" emergency dose. The limiting population dose was chosen to have a very small relative effect, currently about a 0.1% increase in the lifetime incidence of cancer in a population of a million people.

#### **D. Means of Ensuring Safety**

32. The means of achieving these safety objectives can be divided into two broad categories:

- general principles that govern the designers and operators of a nuclear power plant and are used by the regulatory body in assessing the safety (procedural means); and
- the design of the plant and the manner in which it is operated (technical and administrative means).

##### **(a) Procedural means**

33. Although enunciating and enforcing safety principles and practices is a statutory responsibility of the AECB, not all of these principles and practices originated within that body: some originated with employees of the organisations responsible for developing, designing, and operating CANDU reactors. More recently, the AECB's Advisory Committee on Nuclear Safety has reviewed the general safety requirements for CANDU plants and has prepared a recommended list for consideration by the Board (AECB 1983).

34. Canada has participated in the activities of the International Atomic Energy Agency since its inception in 1957 and contributed significantly to the preparation of its series of codes and guides on reactor safety. The safety principles used in the development of these documents have been summarised in a concise list (IAEA 1980). The large measure of agreement between the

Canadian and international lists attests to the fact that many of the nuclear safety principles now internationally agreed upon were already in effect in Canada.

35. In what follows, some of the more important principles will be explained in non-technical language.

36. Doses can be kept within prescribed limits by ensuring that the likelihood of a serious release of fission products is negligibly small. Preventing the rupture of the sheaths containing the fuel will generally satisfy this requirement, but this criterion is, strictly, neither necessary nor sufficient.

37. The four basic causes of a release of radioactive material from the fuel are:

- loss of control of the reactor, so that its power increases out of control;
- loss of coolant to the fuel, so that its temperature increases out of control;
- loss of a heat sink to remove the heat, so that again the fuel's temperature increases out of control; and
- mechanical damage to the barriers surrounding the fuel.

Safety measures are provided to prevent these from occurring.

38. For the purposes of stating safety requirements and assessing the degree of safety, the systems in a nuclear plant are assigned to three groups:

- process systems (those used in the normal operation of the station);
- safety systems (those required to prevent or mitigate an accident and not needed for normal operation, i.e., shut-down and emergency cooling systems); and

- containment systems (those provided to contain any radioactive material that might be released from the process systems).

Failure of any process system is defined as a "single failure," and of any process system combined with the unavailability of any safety or containment system as a "dual failure."

39. All safety systems of CANDU reactors have to be capable of fulfilling their functions for all operating conditions: in Canada, licences do not permit operation when these systems are not fully effective.

40. Particular care is taken to eliminate the potential for "cross-linked" faults and faults that would be common to more than one system.

41. Originally, maximum permissible doses (and hence risks) were stipulated for single and dual failures. The applicant had to demonstrate to the satisfaction of the AECB for each category that the reliability of the systems was such that the frequency of failures that could have consequences within a specified range would be less than a stipulated value. Since then, the simplistic two-category approach has been modified to allow for a more comprehensive review when considered appropriate.

42. In addition, the plant must be shown to be capable of withstanding events such as earthquakes, tornadoes, floods, and aircraft impacts with magnitudes appropriate to the specific site.

43. Implicit in the procedures for assessing and controlling the risk is an acceptance that events expected to occur no more frequently than once in 10 million reactor-years are so unlikely as not to be considered in licensing, however serious the estimated consequences. Nevertheless, these events are taken into account in the emergency planning.

44. An important strategy that has many applications in nuclear safety is "defence-in-depth." The essence of this is the anticipation of failure of an



item important to safety by the provision of a back-up to counter the effects of a failure. The strategy can provide multiple layers of defence against a given failure and can be applied at different levels.

45. In its broadest application, defence-in-depth:

- prevents an accident (accident prevention);
- limits its progress (accident mitigation); and
- limits its effects (accident accommodation).

46. In practice, defence-in-depth means designing and constructing the plant to such high quality that safe operation can be expected from the process systems alone, and that many minor failures could be tolerated without causing any harm. In the event of a more serious failure, i.e., a dual failure, the shut-down and special cooling systems should be capable of arresting a potential accident before damage to the reactor results. Even in the event that these systems fail to prevent damage to the reactor, the containment should prevent escape of radioactive material to the environment.

47. The AECB sets limits on the expected frequency of serious failures of process systems and on the measured unavailability of safety and containment systems, so that a serious accident should not occur more often than once per 3000 reactor-years.

48. Defence-in-depth allows unforeseen faults to be identified and corrected without causing harm. Thus, provided that one learns from experience, the safety of nuclear plants can be continuously improved. To be fully effective, the defence-in-depth principle must be combined with a sound feedback mechanism to ensure that the lessons learned are applied to correct faults.

49. Other important principles in the Canadian approach to nuclear safety are that:

- process and safety systems should be physically and functionally separate with no common components, and each safety system should be separate from the others;
- process and safety systems should be independent of each other and as diverse as possible in concept, manufacture, and location within the reactor;
- redundancy of systems and components should be employed to ensure reliability;
- the operability of safety systems should be capable of being tested while they are in service, to confirm that reliability targets are being achieved;
- wherever possible, systems and components should be designed to "fail-safe," i.e., a failure should result in an inherently safe condition rather than the opposite; and
- passive safety devices that require no external action to make them effective are preferred to active ones.

**(b) Technical and administrative means**

50. Application of these principles can be found in all stages in the life of a nuclear power plant: design, manufacture, construction, commissioning, operation, and, eventually, decommissioning.

51. Throughout all stages, the AECB requires that an appropriate, formal programme for quality assurance (QA) be in existence. The Canadian approach puts the primary responsibility for establishing the QA programme on the licensee, with the AECB responsible for reviewing, approving, and auditing it.

52. Accident prevention is provided by:

- designing according to good engineering practices, using nationally and internationally accepted codes and standards;
- conducting design audits, both within the designers' organisation and by the regulatory agency;

- constructing the station and its equipment from high-quality materials by approved methods; and
- applying accepted QA procedures to procurement, design, construction, installation, commissioning, and operation.

53. Accident mitigation is provided largely by those characteristics of the CANDU design that contribute to preventing and mitigating accidents:

- near-optimum core configuration, which means that most physical disruptions would lead to the reactor shutting itself down;
- low excess reactivity, so that the risk of large reactivity increases, through the ejection of control devices, is small;
- separation of moderator from coolant, allowing control and shut-down devices to be located outside the pressure envelope in a benign environment;
- the moderator, which consists of a large volume of cool water that can serve as a heat sink in the event of a loss of coolant accident combined with a loss of emergency core cooling;
- the lack of combustible material at high temperature in the reactor, such as caused many fatalities and most of the release of radioactive material in the Chernobyl (USSR) reactor accident; and
- the lack of risk of a criticality accident for natural uranium fuel in the light-water storage bays, because natural uranium cannot be made critical in any configuration in light water.

Negative characteristics (a positive void coefficient, potential for flow instabilities in horizontal pressure tubes, and the possibility of the fuel coming into contact with the pressure boundary) are relatively few and are off-set by designed protection.

54. The primary purpose of the emergency cooling systems is accident mitigation, but they can also contribute to the third component, accident accommodation.

55. Accident accommodation is primarily provided by multi-barrier containment, another example of defence-in-depth at a different level. Before radioactive material within the reactor core could reach the public, it would have to escape from:

- the corrosion-resistant refractory fuel;
- the weld-sealed metal sheath;
- the leak-tight primary coolant system;
- the air-tight containment building; and
- the 1-km-radius exclusion zone surrounding the station.

56. There are several examples of the principles of redundancy and diversity, including:

- the physical separation of certain process and safety systems into two independent groups, either of which is capable of shutting down the reactor, providing for removal of the decay heat, and monitoring the reactor, from two distinct locations, in an emergency;
- the provision of two independent and diverse shut-down systems; and
- the provision of stand-by power and water supplies, as well as duplicated valves, pumps, and other components.

57. Instruments and circuits critical to the control and shut-down of the reactor are triplicated, providing not only redundancy but also the opportunity for in-service testing. By incorporating a two-out-of-three logic circuit between a set of three instruments and the system that they control, one component can fail without jeopardising safety, either by not shutting down the reactor or by causing an unnecessary shut-down. Also, the instruments can be tested one at a time without activating the system, a highly desirable characteristic.

58. The fail-safe principle is employed in:

- the control computers, where, if one develops a fault, it switches control to the other, and where, if they are both faulty, the reactor is shut down; and
- the shut-off rods, which drop into the reactor core automatically if there is a power failure.

59. Multi-unit stations, such as those of Ontario Hydro, offer advantages in safety as well as in economy of construction and operation, because they have a larger experienced staff and more equipment available in an emergency. They also permit economic exploitation of containment buildings linked to a vacuum building, which is a passive safety device that can be continuously tested.

60. The training of reactor operators is conducted over several years in such a manner that they thoroughly understand the operation of the reactor and its associated equipment. Because the CANDU safety systems are designed so that even in the event of a serious accident operator intervention is not essential for about 15 min, the operators would have both time and competence to assess the situation and contribute to the mitigation and accommodation of the accident. Reactor simulators are widely used both in the initial training of operators and in their refresher course.

61. During normal operation of the reactor, its operators contribute to safety through rigorous programmes of monitoring, inspection, testing, and maintenance, according to well-defined procedures. The automatic control of CANDU reactors by computer relieves the operator of the need to make any quick decisions under stressful conditions, allowing full use of their diagnostic abilities.

62. Learning from one's mistakes, which, because of the defence-in-depth principle, do not develop into catastrophes, is achieved primarily by the completion and processing of Significant Event Reports (SERs) for all failures and faults, whether of equipment or of humans. These are reviewed internally,

with the more serious ones being scrutinised at a senior level, by the Nuclear Integrity Review Committee in the case of Ontario Hydro. In this connection, it is important that the management style encourages open and honest reporting of faults while discouraging their occurrence.

63. For each of the radionuclides that might be released from the station, a derived emission limit (DEL) is calculated such that as long as it is not exceeded, the permissible radiation exposure to members of the public, as stipulated in the licence, would not be exceeded. In recognition of the ALARA principle, targets for releases are set at 1% of these permissible values.

64. The AECB requires, as a condition of licensing, the preparation of an emergency response plan. The appropriate provincial government has the responsibility for response outside the plant, whereas the licensee is responsible for on-site response, initial action, and continuing support to the province.

65. To ensure that resources devoted to improving safety are used in the most efficient manner, Ontario Hydro has initiated a probabilistic risk assessment incorporating best estimates rather than conservative assumptions.

## **E. Acknowledgements**

66. In preparing this review, the authors have drawn on several existing publications, especially "The Canadian Approach to Nuclear Power Safety" (AECB Report INFO-0104) by R.J. Atchison, F.C. Boyd, and Z. Domaratzki (1983).

## **Notes:**

- [1] During the first two decades of power reactor deployment, the AECB was assisted in safety assessment by a committee of experts, the Reactor Safety Advisory Committee. Several members were drawn from the federal government agency charged with developing nuclear energy, AECL, because that was where most of the expertise existed then. AECB staff

members provided support for committee activities. At each stage--siting, design, construction, and operation--the applicant for a licence or for licence renewal had to appear before the committee and respond to a searching technical examination of the proposal. Several meetings might be held at each stage. As the number of reactors increased and the AECB staff grew in size, the staff members undertook more of the reviewing. For the past decade, they have dealt with the applicants on their own, but they are still assisted by advisory committees in policy development and other general aspects.

- [2] The quantity called "risk" here is sometimes called "detriment" in the literature on nuclear risk assessment.
- [3] The AECB has expressed its objectives in this regard as:
- all detrimental effects that can be detected within days or weeks, which are caused only by very large doses of radiation, should be avoided and the risk of deferred effects, e.g., cancers, should be minimised in accordance with the ALARA principle; and
  - the probability of equipment malfunctions should be limited to small values, decreasing as the severity increases, so that the likelihood of catastrophic accidents is virtually zero.

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- IAEA. 1980. International Atomic Energy Agency Safety Series No. 50-SG-D11 Annex I; and IAEA Bull. 22(1):51.

## Appendix V

**La sûreté nucléaire en France  
après Tchernobyl  
(Nuclear Safety in France after Chernobyl)**

par  
Wladimir Paskievici

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## Executive Summary

1. The Ontario Nuclear Safety Review (ONSR) has commissioned a study on the organisation of nuclear reactor safety in France, on the French nuclear reactor safety philosophy, and on the French response to the accident at the Chernobyl reactor in the USSR.

2. The report on this study is divided into four main chapters. The first chapter describes the main characteristics of the civil nuclear programme in France. Initiated by General de Gaulle, this programme aims at obtaining a third of all energy resources from nuclear power by 1990. At the end of 1986, the installed nuclear capacity of Électricité de France (EDF), the public utility equivalent to Ontario Hydro, reached about 43 000 MW, i.e., 71% of its total capacity, and produced 237.4 TWh nuclear, i.e., 75.8% of its total electric energy. The availability of its mature series was higher than 80% (83.2%), the unplanned outages being limited to 4.3%.

3. The second chapter deals with the organisation of nuclear safety. The regulatory function is carried out by the Department of Industry through its "Service central de sûreté des installations nucléaires" (SCSIN), equivalent to the Atomic Energy Control Board (AECB). Safety analyses for the licensing process are prepared by EDF, reviewed by an independent body, the Institut de Protection et de Sûreté Nucléaire (IPSN) of the Commissariat à l'Énergie Atomique (CEA), equivalent to Atomic Energy of Canada Limited (AECL), and approved by SCSIN.

4. The Minister of the Department of Industry is advised by a body having no equivalent in Canada, the Conseil Supérieur de la Sûreté Nucléaire, a group of "wise people" comprising personalities chosen for their prestige within their organisations (scientific and otherwise). SCSIN has its own advisory groups. Part of SCSIN responsibilities are regionally decentralised. SCSIN also has the responsibility of informing the public on matters related to nuclear safety. This responsibility is shared, in a very structured way, with the Department of Industry, EDF, and CEA.

5. Emergency situations are handled by the Department of Internal Affairs, which takes advice from the departments of Industry, Health, and Defence.

6. The French philosophy on nuclear safety is similar to the one that can be found in Canada and other Western countries. From the organisational point of view, the responsibilities are very well defined, the technical evaluation is independent of the licensing process (this practice differs from that in Canada), and the advising committees are independent. There are safety objectives, design criteria, and detailed requirements. The design is based on the concept "defence-in-depth," which refers to normal operation, anticipated transients, and accidents. Three barriers preventing escape of radioactive products are provided: the fuel sheath, the primary coolant envelope, and the containment. A strong quality control is exercised to ensure that these barriers meet all design specifications.

7. Problems related to the application of these principles are discussed. They include: frequency and consequence limits on postulated severe accidents; classification of accidents for design purposes; use of probabilistic methods; special procedures for mitigating consequences of non-design-base accidents; and measures to decrease the frequency and the severity of human errors.

9. The third chapter deals with lessons learned from the Chernobyl accident. According to the French authorities, the analysis of the Chernobyl accident did not reveal any unknown phenomena. Although several important deficiencies in the design of the Soviet reactor were uncovered, the main cause of the accident lies within the operational procedures of the plant. The French safety experts are convinced that no accident of similar gravity to that of Chernobyl can occur in France; just the same, however, they are taking measures to improve the overall safety in nuclear power stations by implementing the special procedures mentioned above, by reviewing all accident sequences leading to an excess of reactivity, by extending the nuclear training to maintenance personnel, by systematically using feedback from experience, and by commissioning studies for the use of robots in contaminated areas, for new decontamination measures, for fire control, etc.

10. The Chernobyl accident and subsequent handling of the events have raised legitimate concerns in the mind of the public. Major efforts will therefore be devoted to promptly and comprehensively informing the public on all matters related to nuclear energy in order to regain support for the nuclear programme.

11. In conclusion, it is indicated that France has developed, over the years, an impressive expertise in all matters related to the civilian use of nuclear energy. This experience can be helpful to Canada, and it would be highly commendable to increase exchanges of information with that country in these matters.

## A. Introduction

12. Ce rapport fait suite à une mission commanditée par le Ontario Nuclear Safety Review (ONSR) qui consistait à effectuer une étude sur l'organisation de la sûreté nucléaire en France, sur la philosophie de la sûreté dans ce pays et sur les réactions en France à l'accident de Tchernobyl.

13. Dû à des contraintes de temps très sévères, cette mission s'est limitée à deux interviews et à l'étude de plusieurs documents obtenus lors du voyage. Les deux personnes rencontrées ont été Monsieur Pierre Tanguy, Inspecteur général pour la sûreté et la Sécurité nucléaires, Électricité de France (EDF)\* et Monsieur David Lévy, Chef de la Division des tranches nucléaires de la filière à uranium enrichi et eau sous pression du Service central de sûreté des installations nucléaires (SCSIN)\*\*, rattaché auprès du Ministère de l'Industrie, des Postes et Télécommunications et du Tourisme. La documentation obtenue comporte les rapports annuels de 1986 de EDF et du SCSIN, un "grand dossier" du Ministère sur la sûreté nucléaire en France, un article sur la philosophie de la santé en France, deux articles sur l'impact de Tchernobyl, un rapport sur la formation du personnel des centrales nucléaires et plusieurs autres documents divers.

14. La liste complète de ces documents est fournie dans les références [1-6].

15. Le rapport est constitué de cinq parties : principales caractéristiques de l'effort nucléaire français, organisation de la sûreté des centrales nucléaires, approche française en matière de sûreté, impact de Tchernobyl et sujets associés.

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\*EDF est l'équivalent de Ontario Hydro.

\*\*Le SCSIN est l'équivalent de la CCEA (Commission de contrôle de l'énergie atomique).

## B. Principales caractéristiques de l'effort nucléaire français

16. La France possède une longue tradition dans le domaine de l'énergie nucléaire. François Joliot-Curie, prix Nobel de physique pour la découverte de la radioactivité artificielle, a également été le premier à démontrer la possibilité de la réaction en chaîne de l'U-235. Après la Libération, le Général De Gaulle créa, dès 1945, le commissariat à l'énergie atomique (CEA) qu'il a mis sous son autorité. Le CEA avait pour mission de promouvoir l'énergie nucléaire sous toutes ses formes. Les premiers réacteurs construits par le CEA étaient du type GGCR (uranium naturel/graphite/gaz carbonique) et avaient pour but de produire du plutonium. Ils furent installés en 1956 à Marcoule.

17. La même année, les pouvoirs publics demandèrent à EDF de construire les premières centrales commerciales. La première de ces centrales fut localisée à Chinon et devint opérationnelle en 1963. En plus de développer des réacteurs à graphite, EDF acquit une expertise dans les réacteurs PWR (uranium enrichi/eau naturelle) sur un projet en collaboration avec la Belgique (centrale de Chooz, 1967).

18. L'année 1969 marqua un tournant dans l'histoire de l'énergie nucléaire en France. Elle abandonna la filière gaz/graphite et se lança dans la filière PWR. Ceci pour des raisons à la fois économiques (coût de production anticipé moins cher) et industrielles (possibilités d'avoir accès à un marché international). Pour assurer leur indépendance vis-à-vis les États-Unis, les Français développèrent leur propre technologie en matière d'enrichissement d'uranium, construisirent leur premier sous-marin atomique et se séparèrent de Westinghouse, en achetant les parts de cette compagnie dans Framatome, compagnie entièrement française, et en négociant la fin des ententes sur les octrois de licence de cette compagnie.

19. Le choc pétrolier de 1973 marqua profondément la France. Afin de diminuer aussi rapidement que possible sa dépendance des sources énergétiques extérieures, le gouvernement français décida, en 1974, de se lancer dans un programme extrêmement ambitieux visant l'utilisation de l'énergie nucléaire pour un tiers de l'ensemble de ses besoins énergétiques, pour l'année 1990.

20. Grâce à la centralisation des pouvoirs publics en France, une stratégie de développement de l'industrie nucléaire fut relativement facile à mettre sur pied. Les principaux éléments de cette stratégie furent la création d'une industrie nationale, la concentration des moyens techniques et la coopération entre différents organismes nationaux pour atteindre les objectifs fixés. Ainsi, EDF est le seul concepteur, architecte-conseil, propriétaire et opérateur des centrales nucléaires en France. Framatome, créé en 1958, est le seul concepteur et fabricant du système de production de vapeur, tandis que Alsthom-Atlantique, filiale de la Compagnie générale d'électricité (CGE) est la principale compagnie qui conçoit et construit le reste de l'équipement lourd. Par ailleurs, EDF, le CEA et le Ministère de l'Industrie collaborent activement sur l'ensemble des questions touchant le cycle nucléaire.

21. Outre cet effort gouvernemental, le programme nucléaire français a su faire l'unanimité des principaux partis politiques qui, de droite à gauche et souvent pour des raisons différentes, ont appuyé avec peu de réserves ce programme.

22. EDF a pu ainsi commander, pendant plusieurs années de 5 à 6 tranches par année. A ce rythme et avec la concentration industrielle en place, la standardisation des équipements commandés fut relativement facile, les améliorations étant introduites par stages discontinus. On trouve ainsi en France des séries (ou tranches) de 900 MW et de 1300 MW\*, chacune comprenant plusieurs sous-séries (ou trains) de plusieurs modèles identiques.

23. La France a fait également figure de pionnier dans le domaine des surgénérateurs (prototype Phénix 250 MW, 1973, et centrale de Creys-Malville 1200 MW, 1987).

24. L'effort français a porté ses fruits. A la fin de l'année 1986, la puissance nucléaire installée en France par EDF était de 42 955 MW (Chooz, Creys-Malville et Phénix non-compris) sur une puissance totale de 60 296 MW\*\*, soit 71,2%. En 1986, EDF fournissait 313 TWh, dont 237,4 TWh d'origine nucléaire, soit 75,8%\*\*\*. Durant l'année, 7602 MW dont 6615 MW nucléaires (87,0%) furent ajoutés au réseau et 1381 MW dont 1275 MW nucléaires (92,3%) furent commandés.

25. On assiste depuis deux ans à un ralentissement de l'effort nucléaire français. Cela est dû à la fois à une production d'électricité plus élevée que prévue et au ralentissement du développement de l'économie française.

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\*Les Français ont devancé les Américains dans la mise en exploitation commerciale des PWR-1300.

\*\*Les chiffres se réfèrent à l'EDF, la capacité nationale totale est légèrement plus élevée.

\*\*\*73,3% de la production nationale.

26. La performance des centrales nucléaires est très élevée : la disponibilité en énergie a été en 1986 de 83,2% pour la série "mature" des 900 MW (33 tranches) et de 64,5% pour la série "adolescente" des 1300 MW (6 tranches en service industriel). Si l'on exclut les arrêts programmés, les indisponibilités fortuites ont été limitées à 4,3% pour la série de 900 MW.

### C. Organisation de la sûreté nucléaire\*

#### (a) Structures

27. La sûreté nucléaire recouvre la prévention des accidents et la limitation de leurs effets. On y rattache également les mesures techniques destinées à assurer le fonctionnement normal des installations, sans exposition excessive des travailleurs aux rayonnements et sans rejets excessifs d'effluents radioactifs dans l'environnement.

28. Son domaine s'inscrit dans celui, plus vaste, de la sécurité nucléaire qui vise d'une manière générale à assurer la protection des personnes et des biens contre les dangers, nuisances et gênes de toute nature, résultant de la création, du fonctionnement et de l'arrêt des installations nucléaires fixes ou mobiles, des substances radioactives naturelles ou artificielles. Cet objectif ne peut être atteint qu'à travers l'action des départements ministériels concernés, au premier rang desquels figurent ceux chargés de l'industrie, de l'intérieur, de la santé et de la défense.

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\*Cette section contient des larges extraits de [4].



29. Un comité interministériel regroupe les ministres en charge de ces départements, sous la présidence du Premier ministre. C'est à travers la coordination exercée par ce comité et par son secrétaire général, que sont définies l'ensemble des actions visant à atteindre la sécurité nucléaire.

30. Aux termes de la réglementation nucléaire, c'est l'exploitant d'une installation qui peut seul recevoir l'autorisation de créer cette installation et c'est lui qui est finalement responsable de la sûreté.

31. L'intervention réglementaire des pouvoirs publics dans le domaine de la sûreté s'exerce par trois voies principales et complémentaires :

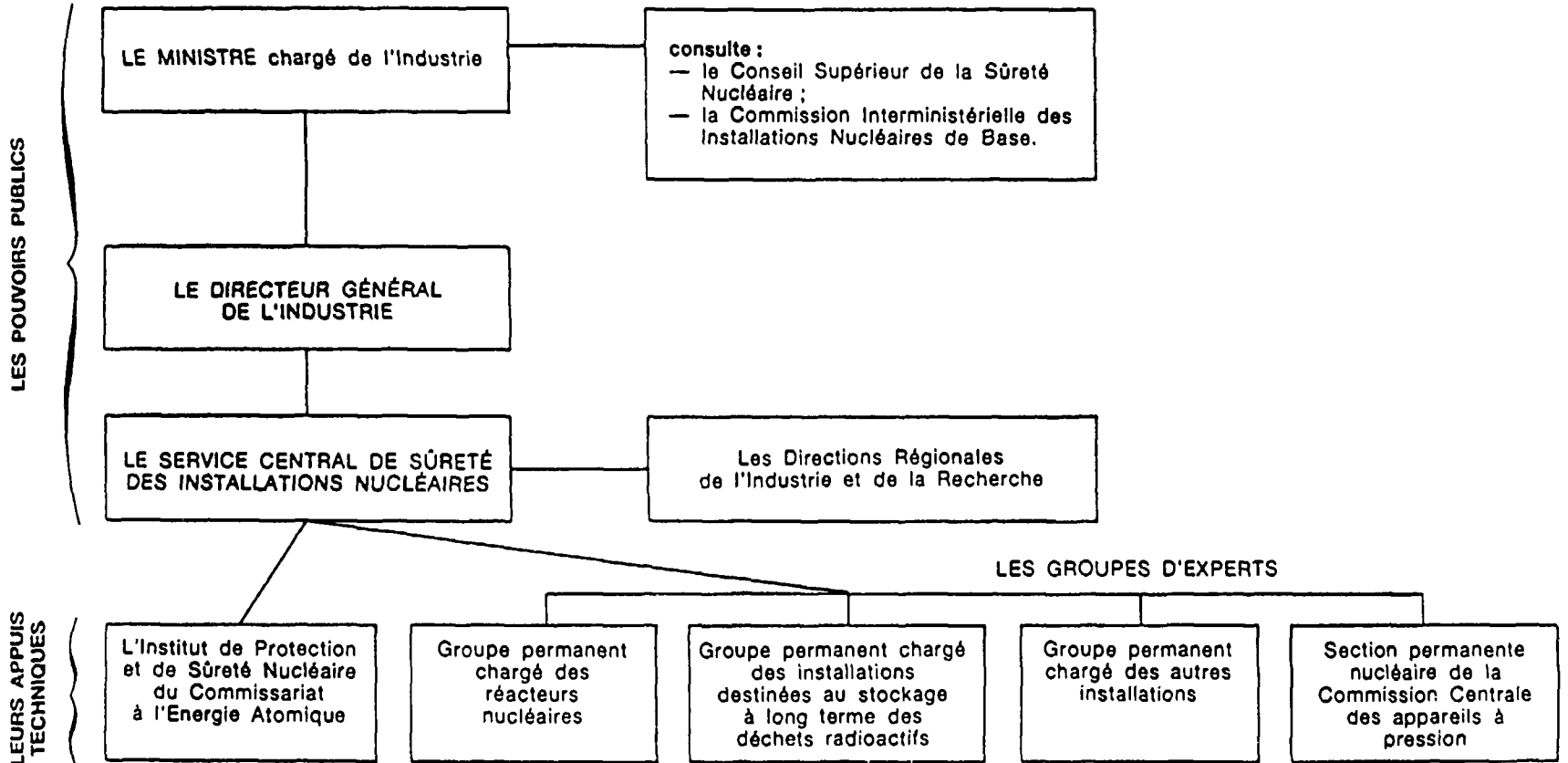
- l'élaboration et l'application de règles techniques de caractère général concernant la sûreté,
- un système d'autorisations individuelles, concernant chaque installation après examen technique approfondi des dispositions destinées à en assurer la sûreté,
- la surveillance.

32. C'est le ministre chargé de l'Industrie qui est en charge des questions de sûreté des installations nucléaires.

33. L'organisation mise en place est schématiquement représentée dans la Figure V-1 de la page suivante.

Figure V.1

## L'ORGANISATION DE LA SÛRETÉ NUCLÉAIRE EN FRANCE



34. Les principaux organismes intervenant dans cette organisation sont:

(i) Le conseil supérieur de la sûreté nucléaire

35. Il s'agit d'un conseil de "sages" de haut niveau composé de personnalités choisies en raison de leur compétence scientifique, technique, économique ou sociale, de représentants d'organisations syndicales représentatives et d'associations ayant pour objet la protection de la nature et de l'environnement, de représentants des exploitants et de hauts fonctionnaires. Il adresse au ministre chargé de l'Industrie toutes recommandations qu'il juge utiles pour accroître l'efficacité de l'action d'ensemble poursuivie dans le domaine de la sûreté nucléaire.

36. Le conseil supérieur de la sûreté nucléaire peut constituer des groupes de travail sur des sujets techniques particuliers.

37. Par ailleurs, les missions du conseil supérieur de la sûreté nucléaire viennent d'être élargies à l'évaluation de l'information sur le nucléaire.

(ii) La commission interministérielle des installations nucléaires de base

38. Cette commission est consultée par le ministre chargé de l'Industrie sur les demandes d'autorisation de création ou de modification d'installations nucléaires de base et sur l'élaboration et l'application de la réglementation relative à ces installations. Elle est composée de représentants des ministères et organismes concernés par la création d'une installation nucléaire.

(iii) Le service central de sûreté des installations nucléaires (SCSIN)

39. Le SCSIN est l'organisme par lequel les pouvoirs publics exercent leur rôle de gardiens de la sécurité publique et de l'environnement. Ce service relève du ministre de l'Industrie, des Postes et Téléphone, et du Transport, via le directeur général de l'Industrie.

40. Le SCSIN est principalement chargé :

- d'examiner les problèmes posés par le choix des sites,
- de mener les procédures d'autorisation relatives aux installations nucléaires (autorisations de création, de mise en service, de rejets, etc.),
- d'organiser et d'animer la surveillance de ces installations par les inspecteurs des installations nucléaires,
- d'élaborer et de suivre l'application de la réglementation technique générale,
- de préparer la mise en place d'une organisation en cas d'incident et d'accident sur une installation nucléaire, lui permettant d'intervenir dans le cadre des responsabilités du ministre chargé de l'Industrie et conformément aux directives du Premier ministre,
- de proposer et d'organiser l'information du public sur la sûreté nucléaire.

41. Le SCSIN suit les travaux de recherche et de développement dans le domaine de la sûreté nucléaire des organismes relevant du ministère chargé de l'Industrie, en particulier le CEA et EDF.

42. Il recueille toutes les informations utiles sur les problèmes de sûreté nucléaire et les mesures prises en France et à l'étranger afin d'être à même de préparer et de proposer, en ce domaine, les positions françaises dans les discussions avec les gouvernements ou les administrations des pays étrangers.

43. Le SCSIN comprend :

- une division chargée des installations nucléaires de base autres que les réacteurs (installations du cycle du combustible nucléaire, en particulier),
- une division chargée des tranches nucléaires de la filière à uranium enrichi et eau sous pression du palier technique de 900 MWe,
- une division chargée des tranches nucléaires de la filière à uranium enrichi et eau sous pression des paliers techniques de 1300 MWe et 1400 MWe et de la centrale à neutrons rapides de Creys-Malville.
- une division chargée des autres réacteurs.

44. Chacune de ces quatre divisions a principalement pour responsabilité de conduire les procédures d'autorisation des installations nucléaires dont elle a la charge et de surveiller la conception, la construction, puis le fonctionnement de ces installations.

- une autre division chargée des questions de la chaudronnerie nucléaire, de l'organisation et de l'animation de l'inspection, ainsi que de l'organisation en cas de crise,
- un secrétariat général chargé des affaires juridiques, financières et administratives.

45. L'effectif du SCSIN, en incluant les inspecteurs qui y sont directement rattachés et les effectifs qui, dans les directions régionales de l'industrie et de la recherche, s'occupent de sûreté nucléaire, est d'environ 170 personnes.

(iv) les directions régionales de l'industrie et de la recherche

46. Il s'agit de divisions spécialisées nucléaires ayant pour but de décentraliser la surveillance exercée par les inspecteurs des installations nucléaires vers les 15 directions régionales et de la recherche (DRIR) déjà existantes.

47. Les responsabilités ainsi confiées aux DRIR en font les interlocuteurs naturels des autorités locales pour toutes les questions de sûreté touchant à la construction et au fonctionnement des installations nucléaires, en relayant ainsi l'action du SCSIN.

(v) l'Institut de protection et de sûreté nucléaire (IPSN)

48. L'IPSN fait partie du CEA mais agit comme organisme-conseil auprès du SCSIN en effectuant les analyses de sûreté qui permettront à ce dernier d'évaluer les projets présentés par les exploitants d'installations nucléaires, notamment EDF pour les centrales nucléaires.

(vi) les groupes d'experts

49. Le SCSIN s'appuie également sur les avis et recommandations de différents groupes d'experts :

- les groupes permanents,
- la section permanente nucléaire de la commission centrale des appareils à pression.

50. Les groupes permanents sont formés d'experts et de représentants de l'administration chargés d'étudier les problèmes techniques que posent, en matière de sûreté, la création, la mise en service, le fonctionnement et l'arrêt des installations nucléaires et de leurs annexes. Un premier groupe traite des problèmes relatifs aux réacteurs nucléaires\*, un deuxième des problèmes relatifs aux installations destinées au stockage à long terme de déchets radioactifs, un troisième des problèmes relatifs aux autres installations nucléaires de base.

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\*Ce groupe se réunit une fois par semaine.

51. Les groupes permanents sont consultés par le chef du SCSIN sur la sûreté de chaque installation nucléaire relevant du domaine d'activité qui leur est confié. A ce titre, ils examinent les rapports préliminaires, provisoires et définitifs de sûreté de ces installations.

52. A l'issue des travaux relatifs à un dossier déterminé - qui fait l'objet d'une analyse par l'IPSN qui fait fonction de rapporteur -, ils émettent un avis, assorti de propositions de prescriptions techniques.

53. Les présidents, vice-présidents et experts de ces groupes permanents sont nommés, par décision du ministre chargé de l'Industrie, pour une durée de trois ans renouvelable.

54. Chaque groupe peut faire appel à toute personne dont la compétence lui paraît justifier le concours. Il peut faire procéder à l'audition, par le groupe, de représentants de l'exploitant.

55. La section permanente nucléaire de la Commission centrale des appareils à pression (CCAP) est consultée sur les modalités d'application de la réglementation des appareils à pression aux chaudières nucléaires.

56. La CCAP est une commission interministérielle placée auprès du ministre chargé de l'Industrie.

## (b) Organisation en cas de crise

57. L'organisation des pouvoirs publics en cas d'incident ou d'accident concernant la sécurité nucléaire est fixée par des directives du Premier ministre. Ces directives visent à assurer la pleine efficacité des dispositions à prendre, en cas d'incident ou d'accident par les autorités chargées de la sûreté nucléaire, de la radioprotection, de l'ordre public et de la sécurité civile.

### 1. L'organisation au niveau local et départemental

58. L'exploitant d'une installation nucléaire, outre ses responsabilités en matière de sûreté et de radioprotection, a la charge d'informer et de conseiller le commissaire de la République\* dès qu'il a connaissance d'un incident ou d'un accident intéressant son installation et de participer à la mise en oeuvre du plan particulier d'intervention (PPI) pour ce qui le concerne. Le PPI est un volet spécifique à l'installation du plan ORSECRAD (Organisation de la sécurité en cas de radiations).

59. Le commissaire de la République, en application de sa mission générale en matière de sécurité des personnes et des biens, est chargé de prendre toutes mesures nécessaires pour assurer la sécurité publique et l'ordre public. En cas d'incident ou d'accident survenant dans une installation et selon le niveau de gravité, il veille à l'action d'information des populations et des élus : s'il le juge nécessaire, il déclenche le plan particulier d'intervention.

60. Les départements ministériels concernés prennent toutes dispositions pour permettre au commissaire de la République de mener à bien sa tâche, notamment en lui fournissant, comme le fait également l'exploitant pour ce qui le concerne, les informations et avis susceptibles

\*Le commissaire de la République relève du ministre de l'Intérieur.



de lui permettre d'apprécier l'état de l'installation, l'importance de l'incident ou de l'accident, et les évolutions possibles, et de prévoir les mesures à prendre.

61. En ce qui concerne l'information, des conventions d'information pour le cas d'incident ou d'accident sont établies entre les commissaires de la République et les exploitants des installations nucléaires ou leurs représentants. Ces conventions précisent par avance les modalités suivant lesquelles se ferait l'information mutuelle des autorités et de l'exploitant.

62. Le secrétaire général du comité interministériel de la sécurité nucléaire assure l'information permanente du Président de la République et du Premier ministre.

63. Des exercices divers sont effectués pour vérifier le bon fonctionnement de cette organisation.

## 2. L'organisation au niveau central

64. Au ministère de l'Intérieur, la direction de la sécurité civile, en accord avec la direction générale de la police nationale lorsque l'ordre public est concerné, anime et coordonne les services chargés de la mise en oeuvre des mesures de prévention et de secours, destinées à assurer la sauvegarde des personnes et des biens, en cas d'accidents, de sinistres et de catastrophes.

65. Le ministre chargé de la défense et le ministre chargé de l'industrie sont, chacun dans les installations relevant de son autorité ou de sa tutelle, responsables de la sûreté des installations nucléaires définie comme "l'ensemble des dispositions techniques imposées au stade de la construction, puis de la mise en exploitation, pour en assurer le fonctionnement normal, prévenir les accidents et en limiter les effets".

66. En ce qui concerne le ministre chargé de la santé, le service central de protection contre les rayonnements ionisants est le conseiller technique des pouvoirs publics pour les mesures de radioprotection relatives à l'homme et son environnement. Il s'appuie sur le comité national d'experts médicaux pour les questions relatives aux accidents créant un risque pour la population.

67. Une directive du Premier ministre a confié au ministère chargé de l'industrie la coordination au plan national de l'information du public et des médias. En cas d'incident ou d'accident affectant une installation nucléaire relevant de sa tutelle, ce ministère peut activer en particulier une cellule d'information renforcée si nécessaire par la présence de représentants des administrations concernées, de leurs appuis techniques et de l'exploitant.

### **3. L'organisation mise en place par le Service central de sûreté des installations nucléaires**

68. En cas d'incident ou d'accident survenant dans une installation nucléaire, le SCSIN, dans le cadre des responsabilités du ministère chargé de l'Industrie, met en place l'organisation suivante :

- un PC\* direction, situé au centre de crise du ministère chargé de l'Industrie, dirigé par le chef du SCSIN ou la personne assurant son intérim;
- une équipe de crise située au centre technique de sûreté du centre d'études nucléaires de Fontenay-aux-Roses du CEA, animée par le chef du département d'analyse de sûreté de l'IPSN ou par la personne assurant son intérim;

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\*Poste de commande.

- une mission locale répartie entre le site et la préfecture concernée, composée d'inspecteurs des installations nucléaires et d'ingénieurs de la direction régionale de l'industrie et de la recherche, du SCSIN et éventuellement du département d'analyse de sûreté de l'IPSN. Sur le site, le rôle premier des représentants de cette mission est de faciliter la bonne information de l'équipe de crise ci-dessus. Le chef de cette mission, désigné par le PC direction du SCSIN, est, en principe, envoyé auprès du commissaire de la République.

69. L'équipe de crise est chargée de faire toute analyse permettant au PC direction d'assurer ses missions de suivi de l'état de l'installation, de conseil au commissaire de la République et de préparation de l'information du public. Les prévisions faites dans ce cadre, comme celles qui lui sont directement fournies par l'exploitant, doivent permettre au commissaire de la République, après avis du service central de protection contre les rayonnements ionisants, de prendre les mesures nécessaires de protection des populations.

70. Si nécessaire, et dans le même cadre, le SCSIN fournit au ministre chargé de l'Industrie les informations utiles pour lui permettre de "prendre d'office toutes mesures exécutoires de nature à faire cesser le trouble".

### **(c) Information du public**

71. Le SCSIN est chargé également "de proposer et d'organiser l'information du public sur les problèmes se rapportant à la sûreté nucléaire". L'action de ce service s'inscrit dans le cadre plus large de l'information du public sur l'énergie nucléaire, dans laquelle EDF et le CEA jouent un rôle important.

72. Le rôle spécifique du SCSIN consiste à informer le public sur la sûreté des installations nucléaires et sa mission de contrôle. Il est assisté pour ce faire par l'IPSN du CEA.

73. EDF, pour sa part, diffuse de plus des versions publiques des rapports de sûreté, très proches des rapports de sûreté des centrales, mais dans lesquelles ne figurent pas les renseignements liés aux secrets industriels, ou qui pourraient être utilisés pour perpétrer des actes de malveillance.

74. Ces rapports sont diffusés aux élus, aux administrations et aux commissions locales d'information. De plus, EDF transmet ces rapports à toute personne qui lui en fait la demande.

75. Le SCSIN produit un bulletin sur la sûreté des installations nucléaires (bulletin SN). Il s'agit d'un bulletin trimestriel, tiré à quatre mille exemplaires, et destiné à informer le public et les administrations ou organismes intéressés par le fonctionnement des installations nucléaires, de l'ensemble des actions menées par le ministère chargé de l'industrie en matière de sûreté nucléaire. Il comporte en moyenne douze pages.

76. Pour sa part, le ministère chargé de l'Industrie édite et rend publics divers documents notamment :

- une brochure d'information sur la sûreté nucléaire en France;
- le rapport des groupes de travail du conseil supérieur de la sûreté nucléaire;
- un bulletin officiel où sont reproduits les actes réglementaires importants et de portée générale, comme les règles fondamentales de sûreté;
- le rapport d'activité annuel du SCSIN qui est diffusé en plus d'un millier d'exemplaires;
- par l'intermédiaire du Journal officiel, les décrets et les arrêtés.

77. Par ailleurs, en collaboration avec le ministère chargé de la Santé, un service d'information sur le nucléaire, accessible par minitel\*, est assuré. Il diffuse notamment des données hebdomadaires sur l'état de la radioactivité dans le pays et sur la sûreté de fonctionnement des installations nucléaires.

78. Finalement, l'IPSN a créé un centre de documentation sur la sécurité nucléaire, ouvert au public, dans les locaux du Centre d'études nucléaires de Fontenay-aux-Roses.

79. Pour plus de renseignements sur l'information du public, on peut se référer aux documents [3] et [4].

#### **D. Approche française en matière de sûreté nucléaire\*\***

80. L'approche française en matière de sûreté nucléaire montre de fortes similitudes avec celles des autres pays ayant recours à l'énergie nucléaire. Lors de l'adoption de la technologie des PWR, l'approche américaine servait de modèle. Avec le temps et surtout après 1978, la France a apporté des modifications à cette approche en innovant même dans certains domaines.

81. Aujourd'hui, l'approche française peut se résumer comme suit:

##### **(a) Principes organisationnels**

82. L'organisation de la sûreté est basée sur trois principes généraux :

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\*Système de vidéo-texte interactif qui jouit d'une grande diffusion en France.

\*\*Cette section est basée sur les références [4] et [5].

## (i) définition claire des responsabilités

83. La responsabilité ultime en matière de sûreté est détenue par l'exploitant (EDF). Le rôle des autorités gouvernementales (SCSIN) est de fixer des normes de sûreté, d'émettre des autorisations de construction et d'opération après avoir vérifié les analyses de sûreté et de surveiller l'application des procédures de fonctionnement.

## (ii) séparation entre l'évaluation technique et le processus administratif

84. Si le SCSIN est responsable du processus administratif, c'est l'IPSN qui évalue les analyses de sûreté.

## (iii) utilisation de l'expertise de comités indépendants

85. Ces comités ont été décrits précédemment. Les plus importants sont le Groupe Permanent Réacteurs (GPR), au niveau technique, et le Conseil Supérieur de la Sûreté Nucléaire (CSSN), au niveau des grandes préoccupations de l'heure (leçons de l'accident de Three Mile Island, la gestion des déchets, la sûreté des usines de retraitement, etc.).

**(b) Réglementation**

86. La réglementation française se situe à trois niveaux, selon son degré de généralité et de son domaine d'application.

87. Au premier niveau, un principe général :

"Le risque d'un accident ayant des conséquences plus élevées que celles jugées acceptables pour un fonctionnement normal devra être limité de façon à ce que la probabilité d'un accident soit d'autant plus faible que les conséquences de l'accident sont plus élevées\*. L'acceptation d'un tel risque doit être basée en égard des autres risques acceptés dans les activités courantes de la société".

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\*Traduction de : "... so that the probability of an accident should be commensurate with the gravity of subsequent damage".

88. Au deuxième niveau se trouvent les critères de sûreté et les règlements d'application. C'est à ce niveau que l'on définit les objectifs en matière de sûreté concernant la conception, la construction, les essais et l'entretien de l'ensemble du circuit primaire.

89. Au troisième niveau se trouvent les exigences techniques particulières devant être satisfaites par le demandeur d'un permis d'exploitation. Ces exigences sont désignées par "Règles fondamentales de Sûreté" (RFS) et touchent l'ensemble des sujets ayant rapport à la sûreté d'une centrale nucléaire.

### (c) Philosophie de base

90. La philosophie de base est la "défense en profondeur".

91. Le premier niveau de cette défense se réfère au fonctionnement normal: la conception, la construction et l'opération d'une centrale nucléaire doivent posséder des marges de sûreté suffisantes pour garantir le bon comportement de la centrale en fonctionnement normal.

92. Le deuxième niveau de défense se réfère aux transitoires normaux (démarrages, montées en puissance, variations de puissance, incidents mineurs): les marges préalablement définies doivent permettre d'éviter le déclenchement d'actions correctives irréversibles et les systèmes de protection auxquels on a recours doivent posséder la redondance nécessaire pour pouvoir rétablir les conditions normales de fonctionnement.

93. Le troisième et dernier niveau de défense se réfère aux transitoires accidentels consécutifs à des défaillances : les analyses de sûreté doivent démontrer, avec suffisamment de conservatorisme, que les systèmes de protection peuvent garder les conséquences d'une série d'accidents postulés en deça des conséquences jugées acceptables.

94. Associé au concept de la défense en profondeur se trouve celui des trois "barrières": la gaine des éléments combustibles, l'enveloppe du circuit primaire et l'enceinte de confinement. L'intégrité de ces barrières entre les produits radioactifs dangereux et le personnel d'une centrale ou le public en général fait l'objet d'un soin particulier au triple niveau de la prévention, de la surveillance et des mesures de parade.

95. Dans ce contexte s'affirme l'importance du contrôle de la qualité. Ce contrôle s'exerce avec une rigueur particulière sur le comportement du combustible (aucune fuite de radioactivité n'est tolérée en fonctionnement normal), l'étanchéité du circuit primaire (tout écart par rapport aux spécifications doit être prouvé comme étant sans conséquences) et l'efficacité des systèmes de protection.\*

#### (d) Problèmes particuliers

96. L'application des principes de sûreté décrits plus haut soulève plusieurs questions dont on traitera brièvement ici :

##### (i) Définition des limites dans les accidents graves

97. La conception d'un réacteur PWR doit être telle que la probabilité de conséquences inacceptables reste inférieure à  $10^{-6}$  par année.

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\*Pour ces derniers, on se fie sur les essais effectués durant la mise en service.



Les conséquences inacceptables sont définies en termes de la nécessité d'une évacuation immédiate de la population environnante. Le niveau de déclenchement d'une telle évacuation est jugé cas par cas mais se situe entre 5 et 50 rems de dose d'exposition individuelle équivalente au corps entier.

98. Un corollaire de cet objectif est que pour qu'une catégorie d'accidents\* puisse être exclue dans les exigences de conception, il faut que la probabilité pour que cette catégorie d'accidents mène à des conséquences inacceptables soit inférieure à  $10^{-7}$  par année, par réacteur.

(ii) Classification des accidents pour fins de conception des systèmes de protection

99. Pour fins de conception des systèmes de protection, on utilise habituellement quatre classes de défaillances, selon leur fréquence, auxquelles on impose des limites aux conséquences radiologiques :

Classes I et II : fréquences  $< 10^{-2}$  a, dose d'exposition en fonctionnement normal

Classe III : fréquence  $10^{-2}$  à  $10^{-4}$  a, 0.5 rem au corps entier

Classe IV : fréquence  $10^{-4}$  à  $10^{-6}$  a, 15 rems au corps entier.

100. La classification des accidents - et des séquences d'accidents - selon ces catégories fait l'objet d'un examen approfondi qui tient compte des retours d'expérience.

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\*"Family of events."

(iii) Place des méthodes probabilistes

101. On n'utilise pas directement les méthodes probabilistes d'évaluation des risques dans la procédure d'autorisation. Ces méthodes sont utilisées pour s'assurer que des séquences d'accidents n'ont pas été oubliées, pour comparer deux systèmes de conception différentes, pour éliminer de l'analyse certains accidents improbables, en particulier ceux provenant de sources extérieures, et pour faire des prédictions sur le comportement des systèmes.

(iv) Procédures "hors-dimensionnement" (H)

102. Advenant la perte de systèmes complets, indépendamment de la cause initiatrice, il convient de prévoir des méthodes de parade adéquats. Il s'agit de procédures spéciales, appelées "procédures H" que les opérateurs doivent appliquer si les systèmes de protection font défaut ou ne suffisent pas à leur tâche.

103. Les procédures H s'appliquent à la perte des sources de refroidissement du circuit primaire ou du circuit secondaire, à la perte des sources d'alimentation électrique, à la perte de certains systèmes de contrôle, etc.

(v) Procédures "ultimes" (U)

104. Malgré le soin apporté à la conception, à la construction et au fonctionnement d'une centrale, malgré la qualité des analyses de sûreté et le soin apporté au contrôle de la qualité et malgré le soin apporté à la formation du personnel, des accidents non prévus peuvent arriver (exemple de Tchernobyl) ou bien des opérateurs peuvent mal diagnostiquer la nature de l'accident (Three Mile Island). Pour faire face à ces cas, les Français ont mis au point des procédures extraordinaires appelées "ultimes" ou "U" qui méritent d'être détaillées :

105. Le principal souci, pour un réacteur du type PWR est d'empêcher la fusion du cœur. Pour cette raison, un ensemble de paramètres a été établi dont la connaissance permet de faire le diagnostic précis de l'état instantané du cœur (p.ex. l'écart par rapport à la température de saturation de l'eau du circuit primaire) et de connaître la disponibilité des différentes sources de refroidissement. La procédure U<sub>1</sub> précise les actions correctives à prendre selon l'état du réacteur et la disponibilité des systèmes de protection.

106. Il est important de noter que tous les renseignements nécessaires pour effectuer un bon diagnostic apparaissent sur le panneau de sûreté de la chambre de contrôle de la centrale, que cette information est à la portée du spécialiste de la sûreté de la centrale - qui peut ainsi se faire une opinion indépendante de la situation par rapport au chef de quart -, et que les mêmes signaux apparaissent sur un panneau dans le PC de crise mentionné dans le chapitre précédent.

107. Si malgré tout, le cœur fond, il reste comme dernière barrière l'enceinte de confinement dont l'utilité a été démontrée lors de l'accident de Three Mile Island. Après avoir étudié les différents modes de défaillance de l'enceinte de confinement - explosion de vapeur, explosion d'hydrogène, fonte du rodier (base en béton) et perte des systèmes d'isolation-, plusieurs procédures U ont été définies. Ainsi par exemple, la procédure U<sub>5</sub> se réfère à une décharge contrôlée d'une suppression à travers un lit de sable ayant la capacité de réduire d'un facteur 10 la concentration des substances radioactives.\*

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\*A l'exception des gaz nobles.

(vi) Les erreurs humaines

108. Les erreurs humaines constituent l'élément le plus difficile à quantifier et à en tenir compte. Les mesures de parade sont : une solide formation, un entraînement faisant appel aux simulateurs pour recréer des situations accidentelles, une introduction progressive d'automatismes pour diminuer les fréquences d'intervention, l'établissement de procédures claires pour le fonctionnement normal et en cas d'accident (procédures A\*, H et U), la mise à la disposition des opérateurs des signaux permettant un diagnostic précis de l'état des principaux systèmes de la centrale, un contrôle serré des procédures de maintenance, un échange de renseignements sur les problèmes rencontrés dans d'autres centrales nucléaires, etc., etc.

(e) Recherche

109. La France possède d'importants laboratoires de recherche (Saclay, Fontenay-aux-Roses, Grenoble, Marcoule, etc.). Ces laboratoires effectuent de nombreuses études dans le domaine de la sûreté des réacteurs. Par ailleurs, la France participe également à des études faites en collaboration avec d'autres pays dans des domaines particuliers. L'aspect recherche n'est pas développé dans ce rapport.

E. Impact de Tchernobyl

110. L'accident de Tchernobyl a suscité de nombreuses réactions auprès des pouvoirs publics et des organismes responsables de la sûreté nucléaire, dans les médias et dans le public en général.

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\*Pour les accidents prévus lors du dimensionnement (conception).

**(a) Pouvoirs publics\***

111. Les pouvoirs publics avaient à réagir devant l'inquiétude de la population concernant les effets des retombées radioactives, la contamination éventuelle des produits importés des pays de l'Est et la sécurité des centrales nucléaires françaises soudainement mise en doute.

112. Le manque initial de renseignements sur l'accident et sur son importance, un certain flottement dans la coordination entre les organismes responsables de la sécurité nucléaire et de l'information au public, ainsi que le manque flagrant d'une concertation sur les mesures à prendre entre les pays de l'Europe de l'ouest, ont fait que les pouvoirs publics se sont sentis mal à l'aise devant la situation durant les premiers jours suivant l'accident.

**(b) Organismes responsables de la sûreté nucléaire**

**1. SCSIN [4]**

113. A la suite de l'accident, le ministre de l'Industrie a demandé au chef du SCSIN d'élaborer une note définissant les thèmes des différentes mesures ou études qu'il lui apparaissait opportun d'engager.

114. Cette note, élaborée avant les informations fournies par les représentants de l'URSS lors d'une conférence organisée par l'Agence internationale de l'énergie atomique à Vienne en août 1986, portait sur la vigilance qu'il convient de maintenir pour prévenir les accidents graves et sur l'approfondissement de la préparation à la gestion par l'exploitant de tels accidents. Les principaux sujets étudiés ont été:

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\*Cette partie n'est pas documentée; elle est basée sur des impressions personnelles.

- l'achèvement de la mise en place des dispositions correspondant aux procédures H et U;
- l'examen des accidents non pris en compte dans le dimensionnement et non couverts par les procédures H et U;
- le réexamen des plans d'urgence internes;
- l'utilisation de la robotique en situation d'accident grave sur un réacteur nucléaire;
- l'approfondissement des connaissances permettant l'amélioration de la gestion technique d'accidents graves.

115. Après les révélations de Vienne, le SCSIN a étudié le scénario proposé par les Soviétiques, l'a jugé vraisemblable et a conclu que

"ce scénario, s'il comporte des erreurs et des violations de consignes par les opérateurs, met en cause une mauvaise préparation et un mauvais encadrement des essais, ainsi que la conception même du RBMK\* qui est un réacteur difficile à piloter et intrinsèquement instable".

116. Les principales conclusions de l'analyse effectuée par le SCSIN sur l'accident de Tchernobyl portent sur les sujets suivants:

- les accidents de réactivité
- la formation du personnel
- la qualité en exploitation
- le confinement
- la gestion d'une situation de crise.

117. Les mesures prises dans ces domaines sont les suivantes:

- (i) EDF s'est engagée à réexaminer les situations susceptibles d'entraîner une variation rapide de la réactivité.

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\*Le type de réacteur de Tchernobyl.

- (ii) Le SCSIN avait déjà demandé à un groupe de travail d'examiner la formation du personnel sur les réacteurs à eau sous pression. Une des conclusions du rapport de ce groupe [8], pertinente au sujet en discussion, est que la formation devait être étendue à tous les personnels intervenant dans une centrale nucléaire, quels que soient leurs niveaux et leurs domaines d'intervention.
- (iii) Le SCSIN a demandé à EDF d'accélérer l'introduction d'un système informatique pour repérer les pièces qui ne respectent pas les spécifications techniques d'exploitation.
- (iv) Les conclusions sur le confinement n'ont fait que confirmer les orientations retenues par le SCSIN dans sa première note.
- (v) La gestion d'une situation de crise nécessite une prévision aussi précoce que possible des éventuels rejets. Pour cela, il faudra bien connaître l'état du coeur, des systèmes de procédés et du confinement de même que son évolution. L'IPSN étudie actuellement les mesures concrètes pouvant améliorer ce triple diagnostic et ce triple pronostic.

118. En résumé, pour le SCSIN,

"Les principales conclusions tirées en France de cet accident ne concernent pas en fait la sûreté des tranches, aucun principe de conception n'ayant été réellement remis en question; elles portent plutôt sur la gestion de l'accident, et en particulier sur le traitement de l'information".

## 2. EDF [6,7]

119. EDF a également effectué une étude sur les causes de l'accident de Tchernobyl et sur les leçons que l'on peut tirer de cet accident. Les résultats de cette analyse rejoignent ceux de l'analyse effectuée par le SCSIN.

120. Ainsi, EDF estime que l'analyse de l'accident de Tchernobyl n'a pas fait apparaître de phénomènes nouveaux qui remettraient en cause les évaluations de sûreté des autres types de réacteurs. L'accident résulte avant tout de défaillances graves dans l'exploitation de la centrale mais l'analyse met également sérieusement en cause la conception du type de réacteur utilisé, le RBMK. Parmi les défauts de conception, on note:

- l'inefficacité du système de contrôle vis-à-vis les contreréactions déstabilisantes,
- le manque de rapidité et d'efficacité du système d'arrêt,
- l'incapacité du système de confinement de retenir des produits radioactifs pour des accidents de coeur non prévus dans le dimensionnement.

121. Sur le plan de l'exploitation, on note qu'il y a une profonde différence d'organisation et de comportement, en particulier en matière de procédures avec la pratique occidentale. Même s'il estime qu'un "Tchernobyl" n'est pas possible en France, EDF reconnaît qu'un accident sévère ne peut pas être totalement exclu et s'efforce à le rendre très peu probable et à en limiter les conséquences.

122. En conséquence, EDF :

- procède à une révision systématique de l'analyse de sûreté, notamment sur les accidents potentiels de réactivité, des différents types de réacteurs en exploitation;
- a préparé et diffusé un dossier d'information sur Tchernobyl à tous les responsables afin de maintenir la vigilance dans la prévention des accidents;
- poursuit les actions entreprises depuis plusieurs années pour garantir une bonne sûreté en exploitation (formation de personnel, organisation de l'exploitation, procédures, interface homme-machine);



- étudie l'emploi d'engins robotisés dans des environnements très radioactifs;
- se propose d'intensifier la collaboration avec les exploitants soviétiques qui paraissent avoir obtenu à Tchernobyl des résultats remarquables dans la conduite des opérations post-accidentelles sur le site, en particulier en matière de décontamination radioactive.

(c) Les médias

123. Les médias d'information français semblent avoir donné une image correcte de la situation, sans tomber dans le sensationnalisme. Par ailleurs, ils ont aussi montré l'incohérence entre les mesures de protection prises par différentes autorités de différents pays qui se sont écartés, souvent pour des raisons d'opportunisme politique, des normes d'intervention fixées par des organismes scientifiques internationaux tels que l'Organisation mondiale de la santé (OMS).

(d) Le public

124. Le public a réagi de façon moins négative en France que dans d'autres pays occidentaux comme l'Allemagne par exemple. L'appui au nucléaire a cependant chuté et la méfiance envers les autorités a augmenté. Ces changements sont loin de mettre en cause le programme nucléaire français mais constituent un avertissement sérieux dont les autorités françaises sont bien conscientes. Pour cette raison, les responsables en matière de sûreté en France mettront dorénavant autant d'emphasis sur l'information au public que sur la sûreté elle-même.

## F. Sujets associés

125. Cette section traite des sujets discutés lors des deux interviews mentionnés dans l'introduction et qui n'ont pas été incorporés dans les sections précédentes.

Les sujets discutés avec monsieur P. Tanguy (EDF) ont porté sur :

(i) la philosophie de la sûreté

126. Répartition des responsabilités, formulation des principes de sûreté, risques résiduels (hors dimensionnement), rôle des méthodes probabilistes, importance des erreurs humaines.

(ii) l'organisation de la sûreté

127. Responsabilité des différents organismes, types d'approche (collaboration, tension ou confrontation), interface entre le promoteur (CEA), le concepteur (EDF, Framatome), les fabricants, l'exploitant (EDF) et les autorités de sûreté (SCSIN), les plans d'évacuation.

(iii) le rôle de l'exploitant

128. Analyse de sûreté, contrôle de la qualité, formation du personnel, relations avec les pouvoirs publics, place de la sûreté dans l'organisation, information auprès du public.

(iv) les effets de l'accident de Tchernobyl

129. Réactions des pouvoirs publics, de la presse et de la population, impacts sur l'EDF.

(v) les leçons tirées de l'accident

130. Réexamen des accidents de réactivité, essais en marche, erreurs humaines, mesures post-accident.

131. Les points importants de cette conversation ont été les suivants:

- (i) Les textes officiels sur la sûreté nucléaire en France relèvent du ministère de l'Industrie, via le SCSIN.

- (ii) Un résumé officieux de l'approche française se trouve dans la référence [5].
- (iii) Les principes généraux et la mise en oeuvre de la sûreté des réacteurs nucléaires à eau sous pression se trouvent dans un "livre blanc" d'EDF présentement sous presse [10]\*.
- (iv) L'approche entre les différents organismes responsables de la sûreté en est une de collaboration lorsqu'il s'agit de questions techniques.
- (v) En ce qui concerne les analyses de sûreté, EDF les prépare, l'IPS les commente et le SCSIN les approuve.
- (vi) EDF est responsable de l'assurance qualité des équipements fournis par l'industrie.
- (vii) EDF est entièrement responsable de la formation du personnel d'exploitation. Elle exerce cette responsabilité avec le concours du CEA. Le rôle du SCSIN consiste à réviser et à approuver les programmes de formation; le SCSIN ne fait pas passer des examens aux futurs chefs de quart.
- (viii) Le responsable de la sûreté à EDF relève directement du président directeur général de l'établissement.
- (ix) EDF estime comme étant extrêmement bénéfique la présence d'un ingénieur spécialiste en radioprotection sur le site même d'une centrale nucléaire. Il s'agit généralement d'un jeune ingénieur, ayant deux années d'expérience, et qui entretient des relations de confiance avec le chef de quart.

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\*Document actuellement disponible, reçu le 14 août 1987.

- (x) Les études probabilistes sont considérées comme une aide aux analyses de sûreté (afin de définir les spécifications techniques optimales) et comme indispensables, après la mise en marche de la centrale, pour mieux intégrer les données sur les défaillances des matériaux et des défaillances humaines. Ainsi, EDF a mis sur pied un système de recueil de données de fiabilité (SRDF) dont le but est la maintenance préventive.
- (xi) Concernant l'erreur humaine, EDF estime qu'elle est incontournable. Sur les quelque 300 incidents significatifs signalés par année, 30 à 40% ont à l'origine une erreur humaine. Pour diminuer ce pourcentage, EDF utilise des simulateurs pour faire passer des "mises en situation recréée" à des chefs de quart en stage d'entraînement ou de mise à jour.
- (xii) En cas d'accident grave, nécessitant le déclenchement des mesures d'urgence, les responsabilités des autorités est claire au niveau local, ou le responsable est le préfet (ou "commissaire de la république"). Celui-ci est avisé par EDF et demande les avis des autorités sanitaires responsables. Au niveau national, c'est plus compliqué car il s'agit de coordonner les renseignements et les avis de EDF, Framatome, IPSN et des ministères de la Santé, de l'Intérieur et de l'Industrie. Il existe des "plans particuliers d'intervention" et l'on fait périodiquement des exercices simulés.
- (xiii) La gestion des actions publiques en France faisant suite à l'accident de Tchernobyl à été confiée au ministère de la Santé qui s'est bien acquitté de sa tâche, du point de vue technique.
- (xiv) Après Tchernobyl, les pouvoirs publics ont réexaminé la question de la coordination entre ministères. Le gouvernement a redonné au ministère de l'Industrie le rôle de coordonnateur et le Parlement a révisé les différents plans d'organisation de secours (ORSEC) au niveau départemental, régional et national.

- (xv) En ce qui concerne les leçons tirées de Tchernobyl, EDF a renforcé les procédures de sûreté, a entamé des études sur la violation éventuelle des consignes et elle revoit la logique des tests.
- (xvi) Pour l'avenir, EDF estime important d'augmenter le niveau des automatismes et de motiver davantage son personnel à l'excellence.

132. Les discussions avec monsieur D. Lévy (SCSIN) ont porté sur les sujets suivants :

- les responsabilités du SCSIN
- les leçons de Tchernobyl
- les erreurs humaines
- sujets sous étude.

133. En plus de confirmer les points déjà traités, cette conversation a apporté les nouveaux éléments suivants :

- L'importance et l'efficacité de l'intervention de l'armée russe dans la gestion de la situation de crise après l'accident de Tchernobyl.
- L'importance du retour d'expérience provenant de la maintenance.
- Le vieillissement des équipements ne constitue pas une préoccupation spéciale du SCSIN, car les programmes de maintenance usuels sont jugés comme suffisants.
- Les sujets sous étude présentement par le SCSIN sont :
  - les incendies : risques, prévention, parades;
  - la perte totale d'alimentation électrique : procédures H<sub>3</sub>;
  - la rupture de la canalisation dans les générateurs de vapeur (surtout à cause des effets médiatiques de ce type d'accident).

## G. Conclusions

134. Malgré sa brièveté, la mission en France a permis de constater l'importance de l'énergie nucléaire pour l'économie française, la qualité de l'organisation de la sûreté nucléaire et le soin apporté par l'exploitant (EDF) à la conception et au fonctionnement des centrales nucléaires.

135. L'analyse de l'accident de Tchernobyl a confirmé la justesse de l'approche française (et occidentale) en matière de prévention des accidents. L'approche nucléaire en France après Tchernobyl sera caractérisée par :

- une attention accrue à la formation du personnel,
- la mise en place des procédures H et U,
- l'incorporation plus systématique du retour d'expérience,
- une révision de la gestion des cas de crise,
- des études techniques sur la maîtrise des incendies, sur l'utilisation des robots en milieu contaminé, sur les procédés de décontamination,
- une meilleure information au public.

136. Pour le Canada, même si le contexte politique est différent, l'exemple français peut servir d'inspiration. Il est donc recommandé aux autorités canadiennes et aux organismes impliqués dans le domaine nucléaire (L'Energie atomique du Canada, Limitée, la Commission de contrôle de l'énergie atomique et Ontario Hydro) de s'intéresser davantage à la situation en France et d'accroître les échanges en matière de sûreté nucléaire avec ce pays (le Canada n'apparaît pas dans la liste des pays avec lesquels EDF a des échanges de collaboration).

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## Appendix VI

### Review of Nuclear Emergency Measures Affecting Ontario, and Other Related Matters

by  
A.T. Prince \*

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\* Formerly President, Atomic Energy Control Board.



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## A. Introduction

1. I agreed to undertake certain assignments in support of the Ontario Nuclear Safety Review (ONSR), with the following terms of reference:

The Contractor shall examine and report upon the existing emergency response arrangements in Canada with respect to their adequacy, the way in which emergency planning is divided among jurisdictions, and the way in which jurisdictional responsibility is coordinated and integrated with specific reference to the following:

- (a) The Federal Nuclear Emergency Response Plan (FNERP);
- (b) The way in which FNERP relates to the Province of Ontario's emergency planning with respect to the possibility of a nuclear accident, which plan is administered by the Ministry of the Solicitor General;
- (c) The existing agreements between American States and Canadian Provinces dealing with the possibility of an accident at one of the American nuclear generating stations located near the Canadian border.

The Contractor shall also examine the role played by the Atomic Energy Control Board ("AECB") in regulating the Ontario Hydro nuclear generating stations, focussing upon the implications of a federal body regulating a provincial utility pursuant to the Atomic Energy Control Act.

In addition, the Contractor shall be available for consultation as required.

2. Further to the above terms of reference, I held discussions with the staff of the ONSR regarding the possible impact of a nuclear power plant accident on the waters of the Great Lakes. Justifiably, much thinking and planning have gone into protecting people, property, and land, but the question of the impact on water supplies from the Great Lakes for many millions of people has been largely overlooked. It was agreed that I should explore this additional subject.

## **B. Perspectives on Nuclear Emergencies**

3. The subject of emergency measures designed to cope with nuclear power plant accidents involves many considerations, some of which are discussed in this section.

### **(a) Prevention versus cure**

4. Where limited resources are available, their use for the prevention of an accident far outweighs their use for the improvement of emergency measures. Release of radiation in excess of acceptable levels, in-plant or to the population at large, constitutes an irreversible impact on the people exposed that is proportional to the levels of release. Damage to property, soils, and water supplies may be permanent or long-term and may be costly to rectify. The Chernobyl experience stressed the importance of the axiomatic concept--prevention.

### **(b) Experience of emergency measures organisations with nuclear accidents**

5. Most emergency measures organisations are required to respond to a wide variety of situations in which help is needed. Almost daily there are reports of transportation accidents involving road vehicles, trains, and aircraft. In most cases, these incidents are handled promptly and efficiently by local and provincial police, firefighting personnel, and municipal officials. Where toxic and dangerous goods are involved, personnel from provincial and federal environmental agencies and provincial emergency measures agencies provide back-up assistance. Transportation and health departments may also be involved.

6. Similarly, much experience has been gained in coping with major fires in oil storage depots, chemical plants, and forests. Natural disasters, such as tornadoes, coastal hurricanes, river floods, avalanches, and earthquakes, occur from time to time and provide opportunities for emergency agencies to gain experience.

7. Nuclear accidents are fortunately rare occurrences, and little field experience has been gained by emergency personnel thus far. There have been a few transportation accidents involving radioactive materials, but packaging and container specifications are carefully prescribed by licence in order to prevent release of radioactive materials. The movement of such materials (notably radioisotopes and spent fuels) by qualified carriers is also under licence, and procedures for reporting accidents are in place.

8. Although the terms of reference of the ONSR preclude investigations of uranium mining and processing, it may be said in passing that spills of uranium ore concentrates in the form of yellowcake or of fabricated uranium dioxide fuel do not present a serious radiation hazard, because they have been largely stripped of traces of radium and its progeny during chemical processing; these uranium products mainly constitute a heavy-metal ingestion hazard.

9. To date there have been four significant nuclear reactor accidents involving release of minor to major amounts of radioactivity beyond the plant containment structures. These are discussed below in terms of emergency measures needed to cope with them.

### **(c) The NRX (Ontario) accident, 1952**

10. In December 1952, the National Research Experimental (NRX) reactor at Chalk River, Ontario, suffered a major accident due to an uncontrolled excursion from low power while some experimental work was being done. Operator error was primarily responsible for the failure to intercept the rapid increase in power. Poor maintenance of the several emergency shut-down systems and the emergency core cooling system led to extensive damage to the core and to release of large amounts of radioactivity to the reactor building, the ventilation filters, and stack effluent. Emergency sirens sounded at the Chalk River plant, and the entire staff was promptly evacuated. The accident occurred on a Friday. So little radioactive contamination affected the site that the staff returned to work on Monday. No off-site nuclear emergency measures organisation existed at the time, and the necessary actions for evacuation of staff,

monitoring of on-site and off-site radioactivity, and clean-up were done by the Chalk River operational personnel.

11. The release of radioactivity from the NRX reactor to the environment was limited in extent. Some was found within the Chalk River plant boundary and was cleaned up. Tests of water from the Ottawa River showed little evidence of radioactivity from the accident at the plant--this was a matter of much concern because of the many municipalities using the Ottawa River for their water supplies. The large quantity of highly radioactive water that had flooded the reactor building was stored and ultimately released by a new pipeline to the monitored disposal area on the plant property.

12. Much was learned from the accident concerning simplified safety shut-down systems, emergency core cooling systems, and operator training. These factors were reflected in the future licensing of power reactors in Canada (see Appendices I and II). A very useful summary of the NRX accident, and its effects on safety systems for Canadian nuclear power plants and on licensing philosophy, is given by Sims (1980).

13. It should be noted that the fuel in NRX was metallic uranium, and that an estimated 12% of the rods were oxidised (burned) and completely destroyed, releasing their volatile fission products. Obviously, combustible metal is not a desirable material for fuel rods. As discussed later in this appendix, the more stable uranium dioxide fuel was subsequently developed for use in the proposed Nuclear Power Demonstration (NPD) reactor to be built at Rolphton, Ontario.

#### **(d) The Windscale (UK) accident, 1957**

14. The Windscale reactor was not a power reactor, but was used for the production of fissile material for a weapons programme. The reactor had a graphite moderator, was air-cooled, and was uranium-fuelled. It thus possessed all the constituents for a good bonfire, which subsequently took place. A characteristic of graphite known as the Wigner Effect occurs when this crystalline form of carbon is exposed to intense radiation, causing a build-up of

energy in its structure. Dissipation of this excess energy is achieved by periodically heating the core to moderately high temperatures as part of the controlled operation. The Windscale accident occurred when the reactor overheated during an energy dissipation operation, causing oxidation, melting, and burning of the uranium metal fuel, with liberation of its volatile fission products. The graphite moderator also ignited. Widespread dissemination of radioactive material occurred in the countryside.

15. The emergency measures in effect in the United Kingdom at that time were not well developed, but had been the subject of mock exercises and proved to be reasonably effective in coping with the accident (AECL 1987).

16. The accident significantly affected a large part of England, resulting in contamination of pasture land, uptake by cattle, and unsafe levels of iodine-131 in milk. Emergency measures were effective in preventing ingestion of unsafe amounts of radionuclides by the public.

17. Clearly, in the context of prevention of accidents, uranium metal - graphite - gas coolant combinations leave much to be desired. Their avoidance in the Canada/US power reactors is a strong component of inherent safety.

18. Operator error was involved in the initiation of ignition of graphite and uranium and the resulting fire in the reactor core.

**(e) Three Mile Island (USA) accident, 1979**

19. On 28 March 1979, at the Three Mile Island Nuclear Plant in Pennsylvania on the Susquehanna River (a short distance below Harrisburg, the state capital), a series of unfortunate mistakes was made by the operators. The control room instruments indicated that the steam pressuriser unit was full of water and had lost its steam pocket. The operators therefore reduced water flow into the reactor, leading to overheating of the core and its disintegration. What they failed to observe was that a valve was open in the pressuriser circuit, allowing the escape of water from the primary heat transport system.

Had this valve been closed by the operators, the accident could have been avoided, the core of the reactor would not have been destroyed, and the enormous economic loss would have been avoided. Operator error and failure to understand the system completely were thus the prime causes of the accident (Kemeny 1979; Brooks and Siddall 1980).

20. The containment structure proved capable of withstanding the pressure and temperature build-up, and only minimal releases of radioactivity occurred from time to time following the accident. Some critics of nuclear power and some local residents claim that much more radioactivity was released than the official records show, but there is no hard evidence to support this allegation.

21. With regard to emergency measures, many sources of assistance were rapidly assembled. One of the difficulties arising was a lack of good communication between and among the utility and the various emergency organisations. The lack of a single clear centre of administration and direction of the operations was also a source of some confusion. Measures have been taken by US authorities to ensure that these problems will not occur in future. Emergency measures responses to simulated accidents held at nuclear plants since the Three Mile Island accident have been demonstrated, most recently at the Zion, Illinois, plant on 24-25 June 1987 (in which observers from the ONSR took part). These realistic exercises have proved useful in establishing optimum procedures for dealing with a nuclear power plant accident.

#### **(f) The Chernobyl (USSR) accident, 1986**

22. On 26 April 1986, the most disastrous nuclear power plant accident thus far occurred. Details of this accident, its causes, and its consequences have been fully documented in many reports issued in the last year (e.g., AECB 1987; Howieson and Snell 1987).

23. Thirty-one persons died from exposure to intense radiation or from burns, all within the plant, and large population dosages in the USSR and elsewhere in Europe may lead to many premature deaths in the future. This is the first

nuclear power plant accident in which large releases of radioactivity occurred with loss of life and serious damage to property and terrain. Most of the casualties occurred among the emergency measures personnel (chiefly firefighters) and the plant operators. The firefighters acted with great courage and self-sacrifice in their efforts to extinguish the many fires at the buildings adjacent to the exploded reactor, caused by flaming fragments of debris hurled by the explosion. Radiation was at lethal levels where the firefighting was in progress.

24. It has been estimated that about 10% of the emissions of radioactivity from the plant resulted from the initial core explosion due to the uncontrolled energy growth, and 90% resulted from the intense combustion of the core material (mainly graphite). It has also been estimated that if water had been used as the moderator, rather than graphite, only about 1% of the release of radioactivity by the graphite-moderated Chernobyl reactor would have occurred. The nuclear fuel was enriched uranium dioxide, not metallic uranium. The latter would have burned vigorously, releasing even more heat.

**(g) Pertinent factors in past serious nuclear reactor accidents leading to emergency measures actions**

25. Based on the foregoing information (obtained from various sources) on the four major nuclear accidents, Table VI-1 gives a summary of the several items of construction and operation that contributed to the accidents or minimised their effects. These factors, in particular the negative impacts of human error and deliberate intervention, along with poor containment and the use of combustible fuel and moderator, will now be discussed. (Note that oxidisable metals such as zirconium are unavoidably required and continue to be used in reactor core construction.)

26. Table VI-1 and the previous outline of four reactor accidents lead to obvious conclusions concerning operations, design, and materials of construction. A disturbing fact is evidence of human failure, i.e., errors by plant operators and their actions of deliberately not following established rules. Rigorous



Table VI-1

Summary of Pertinent Factors in Past  
Nuclear Reactor Accidents

<u>Factors</u>	<u>NRX</u>	<u>Windscale</u>	<u>Three Mile Island</u>	<u>Chernobyl</u>
<b>Fuel</b>				
Uranium (MP* 1130°C)	Yes	Yes	-	-
Uranium dioxide (MP ~ 2700°C)	-	-	Yes	Yes
<b>Coolant</b>				
Water	Yes	-	Yes	Yes
Gas	-	Yes	-	-
<b>Moderator</b>				
Graphite	-	Yes	-	Yes
Water (heavy or light)	Yes	-	Yes	-
<b>Reactor function</b>				
Power	-	-	Yes	Yes
Research	Yes	-	-	-
Fissile production	-	Yes	-	-
<b>Operator action</b>				
Ignoring of rules	Yes	?	?	Yes
Error	Yes	Yes	Yes	Yes
<b>Containment</b>	Limited	Limited	Good	Poor
<b>Emergency measures</b>	Staff	Yes	Yes	Yes
<b>Consequences</b>	Small, local	Widespread release	Low release	Major release
<b>Casualties</b>	None	None	None	Many deaths

\* Melting point.

training programmes for reacting to abnormal conditions and improved administrative supervision could have avoided some of the past accidents.

27. For the operation of Canada Deuterium Uranium (CANDU) systems or research reactors, a clear-cut set of rules has been established by the AECB in consultation with the utility or other licensees. No departure from the rules for matters of maintenance or testing, or other reasons, can be made without the proposal being referred to the AECB resident Project Officer and his staff at CANDU sites, for approval or rejection (if unsafe). The Project Officer refers proposed departures from rules to the headquarters staff of AECB in Ottawa for further opinion, except for the most routine of requests. Failure to follow the foregoing procedures could result in serious legal action and/or suspension of licence and shut-down of plant.

28. Obviously, plant design to avoid escape of radioactivity must include the best of containment. The Three Mile Island plant had good containment and hence very little off-site release; the Chernobyl plant did not have secure containment, allowing a major off-site release. A further element of plant design is the presence of fail-safe control and shut-down systems that can thwart errors and unsafe intervention by an operator.

29. The use of graphite as the moderator is commonly practised in past and current power reactor design in the Soviet Union and United Kingdom. As a combustible material (crystalline carbon), graphite constitutes an obvious hazard, as shown by experiences at Windscale and Chernobyl. Graphite has not been used as the moderator in Canadian and US nuclear power plants. Water, either light or heavy, comprises the heat transport vehicle and moderator in these plants, adding to the safety of such systems.

30. The use of uranium metal or alloys has long been known to be dangerous for large-scale use (tonnage amounts) in power plant design and has been avoided. Uranium alloys are, however, used in specialised power plants for ships and space vehicles, in much smaller amounts than for nuclear power plants.

31. Following the NRX accident at Chalk River in 1952, which involved metallic fuel, it became clear to the research workers engaged in power plant design that more stable compositions for nuclear fuel would have to be found. As plans proceeded at Atomic Energy of Canada Limited (AECL) for the building of the NPD plant, the critical problem of producing a stable, refractory fuel remained unsolved. In late 1954 and early 1955, little was known about uranium dioxide as a potential fuel, other than some abortive attempts in other countries to achieve practical means for fabrication of pellets or rods. AECL approached the Mines Branch (now Canmet, in Energy, Mines and Resources Canada [EMR]) in Ottawa for urgent assistance. The problem was given for solution to a small group, the Physical and Crystal Chemistry Section. At that time, all atomic studies and development were done under tight security requirements, but complete disclosure of the fuel work was made within a year or two to encourage peaceful uses of atomic energy.

32. The basic procedure for the practical and economic preparation of maximum density (near 10.97) pellets of uranium dioxide was found in a comparatively short time in 1955 by the Mines Branch team (Bright et al. 1956). The basic discovery of the uranium dioxide process at the Mines Branch was rapidly scaled up to tonnage production by the work of the Chemical Engineering Branch at AECL, Chalk River, Eldorado Mining and Refining (now Eldorado Nuclear), Port Hope, and the Carbaloy Division of Canadian General Electric, Toronto. The Canadian uranium dioxide fuel process is now used world-wide for power reactors and contributes materially to their safety.

### **C. The Ontario Nuclear Emergency Measures Organisations**

#### **(a) The three emergency measures organisations**

33. The provinces have the prime responsibility of providing assistance and protection to their people using their own resources, together with those of the municipalities. Much experience has been gained by these two levels of government in dealing with non-nuclear accidents; however, only the mechanics

of response exist for the case of a nuclear accident, with its unfamiliar and different problems. In Ontario, much effort is going into providing information and education to the public near nuclear plants to indicate the nature of the risks and the course of action to be taken, even though the probability of an accident with off-site consequences is considered extremely remote.

34. There are three organisations for nuclear emergency planning and action in Ontario, closely integrated within themselves and in relation to each other:

- the Ontario Hydro emergency measures organisations at each nuclear generating station;
- the Ontario Hydro Emergency Operations Centre (OHEOC), Toronto; and
- Emergency Planning Ontario (EPO) - Nuclear Emergency Plan (NEP) (Several Parts), which is based for action at the Provincial Operations Centre in Toronto; the Ministry of the Solicitor General is responsible for EPO.

**(b) Visits to Ontario Hydro locations and to  
Emergency Planning Ontario (EPO)**

35. I visited several individuals and groups at Ontario Hydro concerned with safety and emergency measures, including:

- the Radioactivity Management and Environmental Protection Department (RMEP), located at The Atrium on Bay Street, Toronto;
- the Vice-President, Design and Construction, and Chief Health Officer at 700 University Avenue, Toronto; and
- the Senior Project Officer, AECSB, at Pickering Nuclear Generating Station (NGS) and his staff.

Several senior Ontario Hydro officers concerned with emergencies and laboratory operations at Pickering NGS were also visited.

36. I did not visit Darlington NGS (now under construction) or Bruce Nuclear Power Development, which I had visited some years ago. It is evident from discussions at RMEP that emergency plans for radioactivity problems at Bruce Nuclear Power Development are similar to those at Pickering as well as to the future plans for Darlington.

37. The Bruce plant has the additional risk of chemical hazards from the use of large quantities of hydrogen sulphide in its heavy-water plants. This hazardous substance must be dealt with promptly by local action. Its impact is of short duration, but it is highly toxic.

38. Documents provided to me at Pickering NGS and at RMEP in Toronto were all of high quality. Safety and emergency measures documents on nuclear power stations in the form of memoranda and instruction manuals are written by the respective staffs of the nuclear generating stations to which they apply. Assistance is available from RMEP if required, and RMEP reviews all such documents to ensure compliance with corporate policies.

39. I also visited the headquarters of EPO at its Grosvenor Street offices, Toronto. Useful discussions were held with the co-ordinator of NEP, who provided several publications on emergency planning that have proved to be of great assistance in understanding the complexities of the systems. These documents are prepared with meticulous care with a view to ensuring the clarity of the notification procedure and assigning the responsibilities of all parties involved.

#### **D. Notification Procedures at Nuclear Generating Stations**

40. The nuclear generating station manuals and instruction memoranda give much detail in concise and well-compartmented form. Complete instruction is given on how notification of emergencies is to be given to all persons and units involved, both on-site and elsewhere. The on-site staff is assigned clear-cut

duties and is instructed where and when to report and how to relate to other groups in an emergency.

41. The control room Shift Supervisor is the key person initiating emergency action when an abnormal signal arises on one of the four control panels recording and controlling the reactors. Evidence of a breach of plant security, earthquake, tornado, aircraft impact, or other externally caused factor would also lead to emergency alert. The importance of rapid and effective action by the Shift Supervisor cannot be overemphasised. Interception and correction of any malfunction in the complex reactor system by the Shift Supervisor mean that action by emergency measures units further down the line will be unnecessary. The main function of the Shift Supervisor thus should be to put other emergency measures units out of business.

42. At the two eight-reactor stations in Ontario, there are two control rooms--one for each group of four reactors. At Pickering, if the A group is on alert, the B group of operators and maintenance staff continues operating, but also follows a planned assignment to assist and protect the immediate off-site residents and support the local municipal authorities until the provincial government's NEP can be put into effect.

43. For the rapid alerting of all personnel on-site, nuclear generating stations are equipped with warning sirens and loud-speakers. The timing of siren blasts and announcements is prescribed in a procedures document and extends to public park areas adjacent to plants. The procedure for Pickering NGS is described in Pickering NGS (1986), section 5.0, pp. 9-10; this reference also gives complete instructions to all personnel on-site--both those where the accident occurred, called the Incident Shift Unit (e.g., Pickering A), and those in the Opposite Shift Unit (e.g., Pickering B).

44. Notwithstanding the urgent action required by all station personnel related directly to the alert or the accident itself, there is a further responsibility of notification to off-site organisations. Two channels of

notification have been set up in the station plan as outlined in Pickering NGS (1987).

45. A sketch of the notification procedure is given in Table VI-2, using the Pickering NGS as an example. It is obvious that the control room Shift Supervisor and Plant Manager would not in most cases be making the calls, but would instruct members of their staffs to make them. Presumably the reason for notification via the Shift Supervisor is to apprise key agencies first-hand on the status of the emergency.

46. From the list in Table VI-2 of persons and agencies notified by the Pickering NGS Shift Supervisor and the Emergency Response Manager, three fan-out notifications take place:

- item 4 by the Shift Supervisor, whereby OHEOC in Toronto is alerted;
- item 5 by the Shift Supervisor, alerting the very important EPO-NEP in Toronto; and
- item 3 by the Emergency Response Manager, alerting the AECB Senior Site Representative (Project Officer) at the nuclear generating station and the headquarters in Ottawa.

47. The OHEOC operation is described next in this appendix, and the AECB fan-out is described in a later section.

#### **E. Ontario Hydro Emergency Operations Centre (OHEOC)**


48. This organisation, located in The Atrium, Bay Street, Toronto, is alerted as quickly as possible by the control room Shift Supervisor of a nuclear generating station where an incident occurs. Management of OHEOC comes under the leadership of the Director, Nuclear Generation Division--a senior official, formerly a Plant Manager at nuclear generating stations in Ontario. As head of OHEOC during an emergency, the Director assumes the title Emergency

Table VI-2

**Pickering Nuclear Generating Station  
Radiation Emergency Notifications**

Control Room Shift  
Supervisor

Emergency Response  
Manager (Plant Manager)

- |  |   |
|--|---|
| <ol style="list-style-type: none"> <li>1. Emergency Response Manager</li> <li>2. System Supervisor, System Control Centre, Richview, Ontario</li> <li>3. Seven on-site contacts for emergency communications and follow-up action; also local ambulance and hospital</li> <li>4. Ontario Hydro Emergency Operations Centre (OHEOC), The Atrium, Toronto</li> </ol> <p>Emergency Recovery Manager<br/>(Director--Nuclear Generation Division)</p> <p>Alternatives if not incident site:</p> <ul style="list-style-type: none"> <li>- Operations Manager Bruce Nuclear Power Development</li> <li>- Station Manager Pickering NGS</li> <li>- Station Manager Darlington NGS</li> <li>- Station Manager Bruce A NGS</li> <li>- Station Manager Bruce B NGS</li> </ul> <ol style="list-style-type: none"> <li>5. Duty Officer, Ontario Provincial Police Headquarters (to notify EPO at Provincial Operations Centre)</li> <li>6. Durham (County) Regional Police Communications Officer or Pickering Fire Department (both if public involved)</li> <li>7. Metropolitan Toronto Police Communications Officer (if any possibility of Toronto public involvement)</li> </ol> |  <ol style="list-style-type: none"> <li>1. Production Manager, Pickering NGS</li> <li>2. Technical Support Manager, Pickering NGS</li> <li>3. Senior AECB Site Representative (and AECB Duty Officer, Ottawa)</li> <li>4. Assistant Response Manager, Pickering NGS</li> <li>5.* On-site Co-ordinator, Pickering NGS</li> <li>6.* Off-site Co-ordinator, Pickering NGS</li> <li>7.* Communications Co-ordinator, Pickering NGS</li> <li>8.* Staff Development Co-ordinator, Pickering NGS</li> </ol> |
|--|---|

\* The previous four co-ordinators have further notifications to make.



Recovery Manager and is aided by the Assistant Emergency Recovery Manager. The Emergency Recovery Manager is also assisted by four expert managers skilled in various aspects of nuclear technology and emergencies.

49. The head of the Emergency Preparedness Section, RMEP, is transferred to the direction of the Emergency Recovery Manager where he becomes the OHEOC Co-ordinator; staff of the Emergency Preparedness Section is also assigned to OHEOC. The physical operation of OHEOC is undertaken by the Emergency Preparedness Section, and much of the co-ordination of activities is done by it (Ontario Hydro, date unknown).

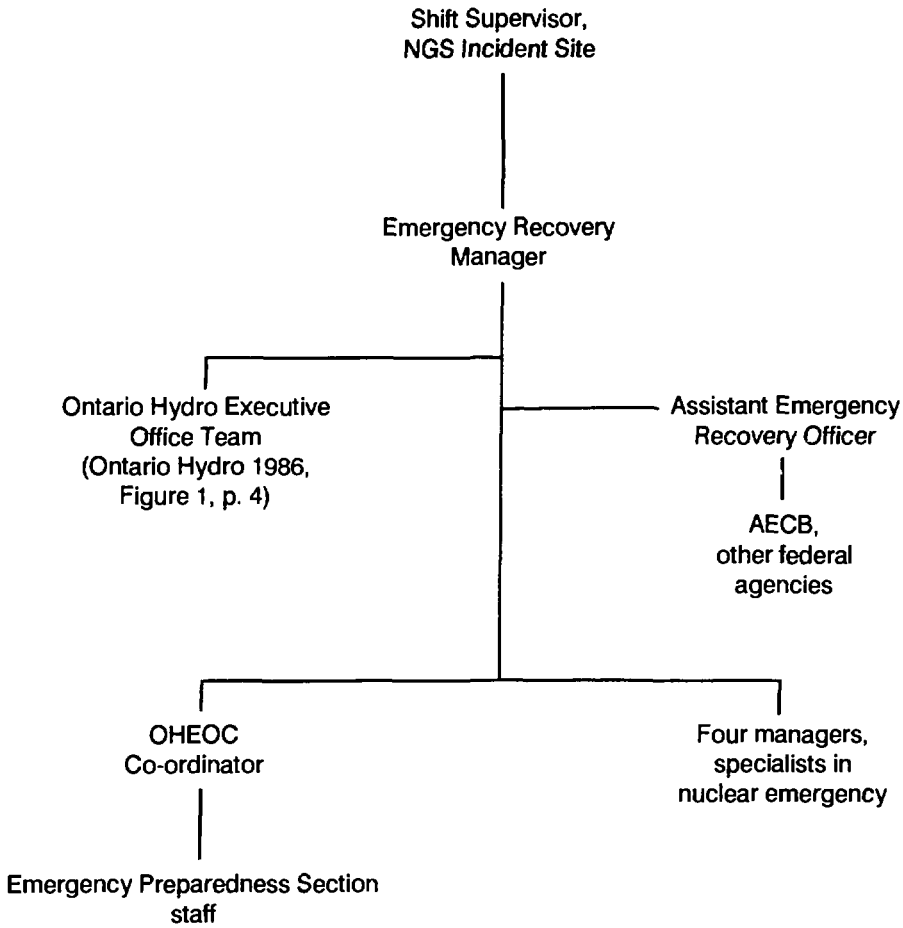
50. As soon as possible after initial appraisal of the incident, the Emergency Recovery Manager must provide essential information to EPO at the Provincial Operations Centre in Toronto and continue the flow of information, because of the responsibilities of OHEOC for front-line action with the municipalities and the public within the potential sphere of influence of the incident at a nuclear generating station. Table VI-3 shows the notifications required.

51. The function of OHEOC is to provide back-up to the emergency actions being taken at the nuclear generating station where the incident is taking place. A nuclear accident may be of limited extent and short duration or more extensive and longer in duration. With a fully qualified group of experts under the direction of a senior experienced manager working off-site, it is possible to have very reliable direction given to the nuclear generating station Shift Supervisor and Emergency Response Manager, both of whom are working under severe stress at the site. A continuous exchange of information, questions, and answers flows between OHEOC in Toronto and the subject incident site. The existence of OHEOC is an added strength to Ontario Hydro's emergency plans.

## **F. Comments on Ontario Hydro Emergency Measures**

52. Efforts by Ontario Hydro to comply with licensing and regulation requirements of the AECB and to respond to provincial government requirements

Table VI-3

Notifications by the Emergency  
Recovery Manager, OHEOC

for protecting the public have been fully and effectively implemented. Resources available to the nuclear wing of Ontario Hydro are impressive and well operated for combatting emergencies.

53. Emergency measures at nuclear generating stations and other ongoing operations are the subject of surveillance and review by RMEP of the Technical and Training Division. A continuing review process by an organisation somewhat removed from production operations gives an opportunity for impartial evaluation, a process that is vitally important where so much is at risk. One has the impression that RMEP is effectively carrying out its responsibilities.

54. Other evidence of effective emergency planning is the schedule of drills, exercises, and the testing of equipment. These operations, depending on their nature, are carried out on prescribed and compulsory schedules--monthly, quarterly, or semi-annually, as required. It is important that schedules be kept and that all personnel participate with serious commitment. Joint exercises are held with other groups, such as EPO, regional municipalities, and police forces (Ontario Hydro 1987).

55. One wonders in passing at the multiplicity of examinations, reviews, appraisals, and evaluations being made of Ontario Hydro by official agencies and unofficial critics. Examples in recent years include the Select Committee of the Ontario Legislature on Ontario Hydro Affairs, the Commission on Electric Power Planning in Ontario, Nuclear Fuel Disposal Study, the Bruce Exercise 1986, an Institute of Nuclear Power Operations study, an Operational Safety Review Team (OSART) review by the International Atomic Energy Agency (IAEA), the present ONSR, and of course the ongoing AECS activity. The co-operation and patience shown by Ontario Hydro officials are noteworthy. It is to be hoped that the time devoted to such co-operation does not detract unduly from regular duties to an extent that jeopardises the safety efforts!

56. In such a large, diversified, and competent organisation with such massive resources at its command, there is a danger that complacency to the point of indifference may arise relative to outside criticism or suggestions. Although no

evidence was found of such a view, it is an attitude that must be guarded against.

57. One area that needs to be evaluated further is the level of physical security at nuclear generating stations. There is a vast difference in the practices of security measures among Western nations. The European countries have very high levels of security, with armed guards and very restricted access. Plants in the United States have almost as much security as the European plants, including armed guards. Canadian stations have much lower levels of security than US plants, and the United Kingdom has even less than Canada. Security measures reflect the nature of behaviour in each society, and it is noteworthy to consider in the present context the difference in security measures being taken by nuclear plants on either side of Lake Ontario.

58. In summary, but with some question regarding security measures, I am satisfied that Ontario Hydro is pursuing an effective programme of prevention of a nuclear accident due to mechanical or systems failures and of limiting radioactive release to containment by effective on-site emergency measures. Plans also include prompt action by Ontario Hydro to assist the public until provincial forces arrive, in the event of significant off-site releases.

59. From discussions in Section B of this appendix, and in particular as shown by Table VI-1, it is clear that human error and ignoring of operating rules have been the chief causes of major accidents in other countries that led to the invoking of emergency action. Under AECB licensing requirements, Ontario Hydro has a major programme of training of staff for duty in the nuclear generating station control rooms. The Ontario Hydro trainees are subjected to thorough AECB examinations--mostly written, but with oral options. Failure rates have been high at times in the past; the present system, because of modifications, is much more effective in training with fewer failures. The use of simulator systems now in effect is a desirable development. Every encouragement should be given to Ontario Hydro to continue and improve its training programmes and to AECB to examine the trainees with strict objectivity.

## **G. Emergency Planning Ontario (EPO) - Nuclear Emergency Plan (NEP)**

### **(a) Review of operations**

**60.** EPO is the central co-ordinating agency for planning and initiating action and response for all accidents of any significance in Ontario. Exceptions are local accidents and fires that can be handled by municipal or regional forces. Even for local incidents, however, the forces involved have access to provincial guidelines and can call on support from EPO.

**61.** EPO in general, and NEP in particular, have very strong backing from provincial statutes and from the Premier of Ontario through a Cabinet Minister--the Solicitor General of Ontario.

**62.** Statutory authority stems from the Emergency Plans Act 1983 (Ontario), especially Section 8. General clauses applicable to nuclear incidents are Sections 2, 3, 4, 7, 9, 10, 11, and 12 (EPO 1986a:106-108). Other statutes of general application to nuclear matters are:

- Health Protection and Promotion Act 1983;
- Farm Products Grades and Sales Act 1937 (for control of possible contamination of foods by radioactivity); and
- Workers' Compensation Act 1984 (this act applies to "a person who assists in connection with an emergency that has been declared to exist by the head of council of a municipality or by the Premier of Ontario").

**63.** The group in EPO responsible for administering NEP is a very small unit (1.5 person-years) that has done an outstanding job of preparing a remarkably detailed and practical series of plans. Outside assistance from other agencies has undoubtedly been utilised, but the final product is the responsibility of EPO-NEP.

64. Based on the guidelines given in NEP Part I (Master Plan) (EPO 1986a), seven other detailed plans have been or are in the process of being produced:

Part II	Pickering Nuclear Emergency Plan
Part III	Bruce Nuclear Emergency Plan
Part IV	Rolphton Nuclear Emergency Plan (operation of this plant has been discontinued)
Part V	Chalk River Nuclear Emergency Plan
Part VI	Enrico Fermi 2 Nuclear Emergency Plan (near Detroit, Michigan)
Part VII	Darlington Nuclear Emergency Plan (currently under construction)
Part VIII	Provincial Plan for Transborder Emergencies (at present in draft form only)--this involves arrangements with all nuclear plants in the states of New York, Ohio, and Michigan and the Gentilly 2 plant in the Province of Quebec.

65. It is virtually impossible to do justice to the EPO-NEP documents by trying to summarise them in this report, and only a few further highlights can be mentioned.

66. First, it must be recognised that for off-site incidents, EPO-NEP is on the firing line and is the command post that directs action through its authority to enlist and commandeer resources from the array of provincial ministries, police forces, Ontario Hydro, the municipalities, and, by agreement, agencies of the federal government such as AECB, AECL, FNERP, and some elements of the Department of National Defence (DND) (EPO 1986a:29). Not only are the foregoing ministries and federal agencies (except FNERP and DND) under notification and on-call, but the Master Plan (EPO 1986a) spells out in considerable detail the responsibilities that each must undertake. Some 29 pages of statements of responsibilities are given in the Master Plan. Of particular interest is the section on AECB responsibilities in relation to Ontario Hydro

(EPO 1986a:99), where the EPO may ask AECB to instruct Ontario Hydro to do certain things.

67. The Master Plan gives details on notification according to three progressively more serious situations, each defined as to expected radiation at the exclusion-zone boundary of the plant. Actual dosage levels are given for each category (EPO 1986a:37).

68. Phases of operations are defined according to time frame--Phase 1, before and a few days after release of radioactivity; and Phase 2, days or months after release of radioactivity (the restoration stage). Actions required by the nuclear facility, the municipalities, and the province according to each of the two phases are given in detail (EPO 1986a:38-41).

69. Notification charts to participants have been developed in such great detail that summarising them is virtually impossible. Ten pages of notification charts have been prepared for use in the various parts of the NEP plans; box compartments are used to represent the various sections, units, or agencies as required for the particular group being notified. For example, List No. 3--Technical Group contains 15 compartments with two to six names with phone numbers given in each of the various compartments. Arrows between adjacent compartment boxes show the routing of notification fan-out. I reviewed examples of the notification fan-out system in drafts of NEP Part III, Bruce Nuclear Emergency Plan (EPO 1986b), and Part VIII, Provincial Plan for Transborder Emergencies (EPO 1986c).

70. The function and responsibilities of the Ontario Liaison Team that would be sent to report on incidents outside Ontario are given in the Provincial Plan for Transborder Emergencies. This team has the following members: Head, Technical Member, Senior Information Officer, and Information Officer, with alternates, where an extended stay is involved. The Ontario Liaison Team would be sent, as needed, to the Gentilly 2 reactor in Quebec (based at Trois-Rivières) or to the State Emergency Operations Centers in the state capitals of New York, Ohio, and Michigan. Close liaison would be maintained with Quebec

provincial and utility officials and, in the United States, with state, federal, and utility officials. Information from the United States or Quebec would be relayed to EPO-NEP for fan-out as required.\*

71. Information services are a vital part of emergency plans, and they are identified in appropriate places throughout the organisation, function, and notification documents issued by EPO-NEP. Information-group notifications were also listed in the Ontario Hydro manuals and emergency planning documents. In this appendix, the review has been largely confined to the more technical aspects of emergencies and the marshalling of such personnel for prevention and control of nuclear accidents. Although little reference has been made to information flow to the media and the public, its importance must be and is recognised. Similarly, it has not been possible in this appendix to deal adequately with the complexities of organisations and functions on the assumption that all agencies and units know what they are doing in these areas. Notification to qualified groups is considered the prime requirement in emergency planning and has therefore been used as the chief thread of logic in this appendix.

72. An excellent public information report relative to the Pickering NGS was issued in June 1987. It was prepared jointly by the Ministry of the Solicitor General of Ontario, the Durham Region, and Ontario Hydro (Ministry of the Solicitor General of Ontario/Durham Region/Ontario Hydro 1987). This is a very frank, informative, and helpful publication written for the general public. Complete instructions on what to do to minimise the effects of radiation are given. Detailed plans are given for population removal from locations near the Pickering plant, if necessary. Evacuation routes and places to use as reception centres are all spelled out in detail. The frankness and straightforward discussions in this report may disturb the public at first, but the realisation that someone is offering protection will soon be accepted.

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\* See Section J for progress on this proposal.



**(b) Comments on EPO-NEP**

73. Little more need be said regarding the excellent work being done in giving leadership in nuclear emergency planning by EPO. Explicit detail is given in practical terms in all the documents reviewed. Several are still in draft form.

74. Discussions with the head of NEP were very helpful. Assistance to NEP from FNERP was discussed, and the NEP head indicated that in an emergency there is a need for additional laboratory facilities, surveyors, samplers, and back-up monitoring equipment. Reference was made to the need for improving the response capability of FNERP and to the fact that its present level is similar to that of Ontario plans about 5 yr ago. Further discussion dealt with the difficulties of a technical and scientific group, such as FNERP, being responsible for the operation of an emergency plan. Indeed, the suggestion was made that technical functions should be separated from Emergency Management Organisation (EMO) type operations and called up as part of the overall plan.

**H. Review of Federal Agencies Involved in Nuclear Emergency Planning****(a) Atomic Energy Control Board (AECB)**

75. The AECB occupies a special position relative to the provinces in matters of all nuclear power plant operations, including emergency measures. There are two distinct responsibilities for AECB. The first responsibility is that deriving from the Atomic Energy Control Act concerning licensing, regulation, and inspection--e.g., the responsibility of ensuring that the licensee has a satisfactory emergency measures plan both on-site and off-site. The second responsibility of AECB is as a core member of FNERP. As a member of FNERP, which the Department of National Health and Welfare (NHW) directs as lead agency, AECB is assigned appropriate supporting duties.

76. In this section, some of AECB's special functions arising from the Atomic Energy Control Act are briefly outlined.

77. Much of the influence of AECB results from the presence of its resident project officers at the large, composite nuclear generating stations in Ontario as well as at the single-unit stations elsewhere in Canada. At the eight-unit stations--Pickering and Bruce--there are four to six project officers; currently there are about as many at the Darlington four-unit station. The single-unit plants usually have two AECB officers. Project officers are assigned to a nuclear generating station early in the station's construction schedule in order to grow up with the installation. Inspections are made of components for the reactor core, the heat transport systems, control and shut-off systems, instrumentation, and other items.

78. Operating rules for each plant are documented under joint consultation between AECB and the licensees and must be adhered to at all times or until such time as modifications are jointly approved. The log books for each 12-h shift in the control room are reviewed daily by AECB project officers. Where any unusual event takes place in operations or maintenance of controls or other systems, the AECB on-site staff must be immediately notified to pass such information on to Ottawa headquarters for evaluation and to consult with Ontario Hydro staff for further investigation.

79. The AECB project officers are well-qualified and experienced engineers and scientists. They are supported by a pool of specialists at AECB headquarters. In conducting certain aspects of their work, the AECB on-site and headquarters staffs follow strict quality assurance principles dealing with construction, operation, and maintenance of the plants. Co-operation and joint inspections take place with the Boiler and Pressure Vessel Branch of the provincial Ministry of Consumer and Commercial Relations. The provincial agency takes the lead in dealing with the secondary heat transport systems and the primary system from the core to the steam generator, whereas AECB takes the lead in dealing with the reactor core and calandria; however, both parties are involved in all phases of the system. The joint decisions of the two

inspection agencies (agreement must be and is reached) define the course of events, and the licensee, Ontario Hydro, must abide by them.

80. The foregoing brief outline of AECB involvement in licensing and regulation of nuclear power plants is presented in this appendix to illustrate the intimate way in which AECB operates at the generating stations and at headquarters. Ontario Hydro and other nuclear utilities in Canada in effect have a built-in consulting and regulating group. The AECB staffs are not paid by the utility, nor are bulk cash transfers made to cover the costs. Office space is provided to AECB at on-site locations by the utilities, without charge. Thus, the AECB officers are not under career development constraints or administrative pressures and are free to perform their required duties in a strictly objective way. The freedom so established is an added component of safety in the complex operation of nuclear power stations.

#### **(b) AECB involvement in emergency measures**

81. Because of its detailed responsibilities with nuclear power operations, AECB is notified of and involved in all nuclear emergency measures in Ontario. AECB can respond in varying degrees, through its resident staffs at generating stations or through additional staff from Ottawa. A draft report outlining AECB responsibilities in emergency measures is under consideration. Both on-site and near-off-site activities are involved as a means of direct support of Ontario Hydro and the Ontario NEP. These activities would be undertaken quite apart from those assigned to AECB under FNERP led by NHW.

82. A further direct involvement of AECB is an agreement to provide a representative to each of the emergency response centres in the three provinces with nuclear generating stations--Ontario, Quebec, and New Brunswick. This action would be taken in the event of off-site radiation contamination. Requests from provinces during night-time hours would be arranged by contact through the AECB 24-h Duty Officer, who would make the necessary notification fan-outs.

**(c) Notification responsibilities of AECB**

83. AECB is also responsible for notifying other agencies in the event of a significant nuclear accident at a power plant. A notification sequence fan-out for AECB is given in Table VI-4.

**(d) The AECB/Ontario Hydro interface**

84. In Section A of this appendix, the terms of reference raised the question of the implications of having a provincial utility regulated by a federal agency--the AECB--under the terms of the Atomic Energy Control Act. An implied question also arises as to whether a utility is failing to discharge its full responsibility by accepting regulation and hence relying on another agency to enforce controls.

85. There is no doubt about the powers of the Atomic Energy Control Act to authorise the current actions being taken by AECB. Section 9 states in part:

The Board may with the approval of the Governor in Council make regulations

- b) for developing, controlling, supervising and licencing the production application and use of atomic energy.

Section 17 further states in part:

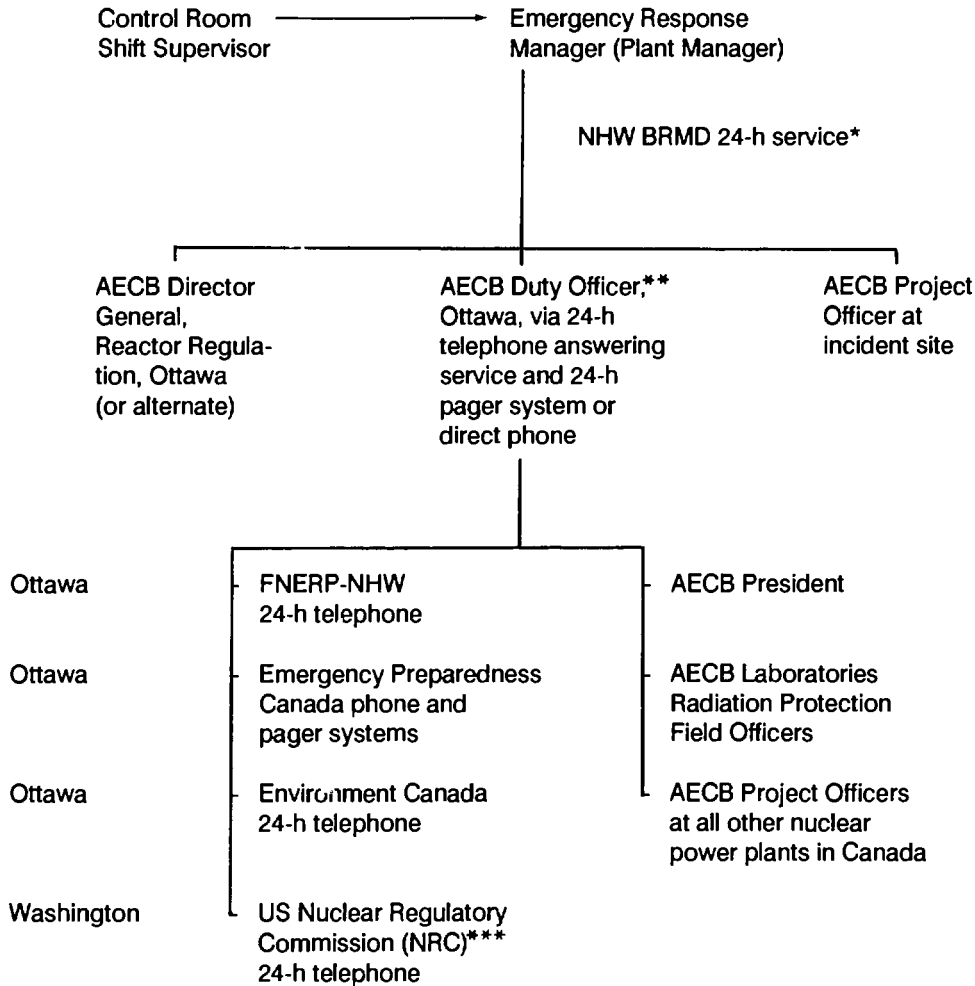
All works and undertakings whether heretofore constructed or hereafter to be constructed

- a) for the production, use and application of atomic energy, are and each of them is declared to be works or a work for the general advantage of Canada.

These two sections of the act make clear AECB's right to take the actions referred to and further imply an obligation that they be done.

Table VI-4

Notification Sequence via AECB



\* The Bureau of Radiation and Medical Devices (BRMD), NHW, Ottawa, may be notified, because they operate a network of field monitoring stations near the exclusion-zone boundary of all nuclear power plants.

\*\* All or some of the contacts listed as notifications by the Duty Officer are contacted at the discretion of the Director General, who may elect to make some of the contacts himself.

\*\*\* US NRC is contacted if an accident is expected to affect any part of the United States.

86. There are, however, legal and constitutional questions arising in situations in which a company, utility, mine, or comparable body is established as a Crown company. Some of the more recent statutes have a clause giving the right to regulate Crown agencies, e.g., in pollution control, but the Atomic Energy Control Act does not confer this right. Hence, when the act is applied to a Crown agency or company, whether it be provincial or federal, difficulties are likely to occur. For example, AECL and Eldorado Nuclear Ltd., as Crown corporations, were not licensed by AECB until recently, and only then by accepting such control. Mines that are the property of a provincially owned energy corporation do not necessarily accept control by AECB. In general, however, because of the sensitive nature of atomic energy matters, the Crown aspect of an agency rarely creates difficulty for AECB.

87. Ontario Hydro evolved from a type of co-operative of many large and small electrical utilities amalgamated as joint owners of the composite utility under the leadership of Sir Adam Beck in the early part of this century, when the Ontario Hydro Electric Commission was formed. The commission was not a Crown company, nor is it now, since the formation in 1971 of Ontario Hydro as a Special Company under the Atomic Energy Control Act.

88. These legal questions are pursued elsewhere in this appendix. In the present context it will be sufficient to deal with certain administrative and operational matters related to the Ontario Hydro/AECB interface.

89. At no time during the many discussions held with Ontario Hydro and AECB officials was there any evidence of problems concerning arrangements concerning regulations. No AECB regulation or operational rule is established and enforced without the full knowledge of the utility and an opportunity to comment before it is put into effect.

90. The licensing and regulation of a utility by an agency of the senior government provide a second well-informed opinion to aid the operators and offer added assurance to the public. AECB also constitutes a court of appeal if divergence of opinions arises between the utility and other government agencies.

It constitutes a safety net to prevent errors by operators or avoidance of established rules, which may lead to accidents.

91. Quotations from two emergency measures documents of the Ontario government illustrate some of the points made in the previous paragraph. In the introduction to the recently issued brochure, "How You Can Be Prepared" (Ministry of the Solicitor General of Ontario/Durham Region/Ontario Hydro 1987), which was distributed to the public near the Pickering NGS, the following comment is made:

Every two years before being granted a license to continue operating the Pickering Station, Ontario Hydro must apply to the Atomic Energy Control Board. Proof must be submitted that the station's in-depth safety systems are being maintained and properly operated to ensure that the station is capable of protecting nearby residents in case of an accident even without an emergency plan. Even though the Pickering station is designed and operated with safety as the top priority and a nuclear accident is unlikely, this emergency plan is in place to protect you, your family and your home.

Signed by:

Chairman  
Durham Region

Solicitor General  
Province of Ontario

Chairman  
Ontario Hydro

92. In the NEP Part I (Master Plan) (EPO 1986a:98-99), the officials of the Ministry of the Solicitor General spelled out 13 responsibilities of AECB. Item No. 12 reads as follows:

In case the Province and the AECB, in consultation, consider that the timing and/or scope of any proposed action by the facility staff should be modified to decrease possible off-site effects of such action, the AECB will issue the necessary instructions or emergency orders to the facility operator.

93. The foregoing responsibility illustrates a situation in which AECB becomes a court of appeal with intervention capability. These two quotations illustrate the importance the provincial and municipal government departments attach to the AECB operation. Similarly, the Ontario Hydro Chairman endorses the

licensing responsibility of AECB. Neither statement implies any failure of the utility to accept its responsibilities, although the Master Plan quotation illustrates the need for resolution of differences of opinion.

94. It is clear, in summary, that the existence of a well-qualified, impartial regulating agency is an acceptable and valuable asset contributing greatly to safety. No failure in accepting responsibility by either the regulator or the regulated body can be tolerated. There is no evidence of failure to accept responsibility by either Ontario Hydro or AECB. The nuclear reactor is a demanding machine and cannot accept indifference by either party: the probability of a serious accident is very low, but the consequences can be severe.

## **I. The Federal Nuclear Emergency Response Plan (FNERP): Jurisdiction**

### **(a) Jurisdictional co-ordination**

95. In the terms of reference of this study, specific information was requested about the co-ordination of jurisdictional responsibility for emergency planning and, in particular, how FNERP relates to Ontario's NEP established by the Ministry of the Solicitor General.

96. Considerable detail was given in Section H on the role of AECB, based on the Atomic Energy Control Act jurisdiction, in the prevention of accidents and on AECB's involvement in emergency measures activities under the direction of the utility or NEP. The present section deals with further federal-provincial jurisdictions. Throughout this appendix, the notification systems have been used as the indices of responsibilities and co-ordination.



### **(b) Ontario jurisdiction for emergency planning**

97. With regard to Ontario, jurisdictional guidelines have been established. The Ontario NEP Part I (Master Plan) Section 3.2.2 (EPO 1986a:18) clearly states the provincial position:

#### 3.2.2 Coordinator of Emergency Planning

This person is appointed by the Lieutenant Governor in Council under the Emergency Plans Act 1983. Under the direction of the Solicitor General the Coordinator is responsible for monitoring, co-ordinating and assisting in the formulation of all provincial emergency plans and in their implementation. The Coordinator ensures co-ordination with the emergency plans of municipalities, the Government of Canada and its agencies and other jurisdictions whose emergencies or emergency plans might affect the province.

98. As part of Ontario's NEP, a Provincial Plan for Transborder Emergencies (dated August 1986) is still in the process of finalisation (EPO 1986c). This is a very comprehensive proposal for close and distant incidents (anywhere in the world). It is mentioned that such notification at present comes from the Government of Canada and some US federal agencies. Section J of this appendix gives additional information on transborder emergencies.

### **(c) Federal jurisdiction for emergency planning**

99. The jurisdictional responsibilities of the federal government in off-site emergency planning are authorised by a letter from the then-Prime Minister, dated 23 March 1984, to the Minister, NHW. These responsibilities are outlined in FNERP (off-site) (NHW 1986), and various excerpts from this plan are as follows:

Scope, para. 3, p. 1:

This plan covers the federal response to nuclear emergencies requiring federal action and to provincial and federal requests for assistance. The plan is designed to ensure compatibility with Provincial plans and to provide an interface between federal and

provincial governments while retaining operational control at that level of government having jurisdictional authority.

Federal/provincial Co-ordination, para. 14, p. 3:

The coordination of the provision of assistance from one level of government to another will be done through existing lines of authority. The coordination and allocation of their resources will rest with the level of government providing them, but operational control will be assumed by the requesting level of government.

Under the heading of "Control of Operations," the following two items give further clarification between the provincial and federal governments:

para. 17, p. 5:

When a nuclear emergency occurs, control of off-site operations will rest with the provincial emergency authorities. Throughout the emergency, the province will undertake the liaison, as appropriate, with federal and municipal authorities and personnel of provincial authorities and agencies.

para. 18, p. 5:

Under circumstances where the nature of the incident requires federal action or where coordination between provinces and other countries is required, coordination of the operations will be the responsibility of the Federal Nuclear Emergency Control Centre. The Centre shall undertake the liaison with United States authorities concerning emergencies which might affect both Canada and the United States of America. These circumstances are detailed in Annex A.

100. With regard to the last item, relating to transboundary incidents (interprovincial or Canada/US), there may be some problems as to what role Ontario assumes and what the federal government does. Annex A (p. A-1) of FNERP (NHW 1986) says that Canada will undertake a federal response in cases where "more than one province may be affected," where "the United States may be affected by a nuclear emergency in Canada," and also where "the emergency originates outside Canada and may affect Canada." These statements are,

however, qualified in Annex A, where it states that "a federal response does not automatically presume federal control."

101. Apart from some need for clarification of jurisdictional responsibility and proposed actions in the transborder areas, co-ordination between the Province of Ontario and the federal government (FNERP) seems quite clearly defined. Even in the transborder case, there is no suggestion of serious conflict.

**(d) FNERP notification channels**

102. The sources of notification to FNERP for nuclear incidents, relative to ONSR, are limited to nuclear power plants and do not pertain elsewhere in the nuclear fuel cycle. The pattern of notification is simplified in Table VI-5.

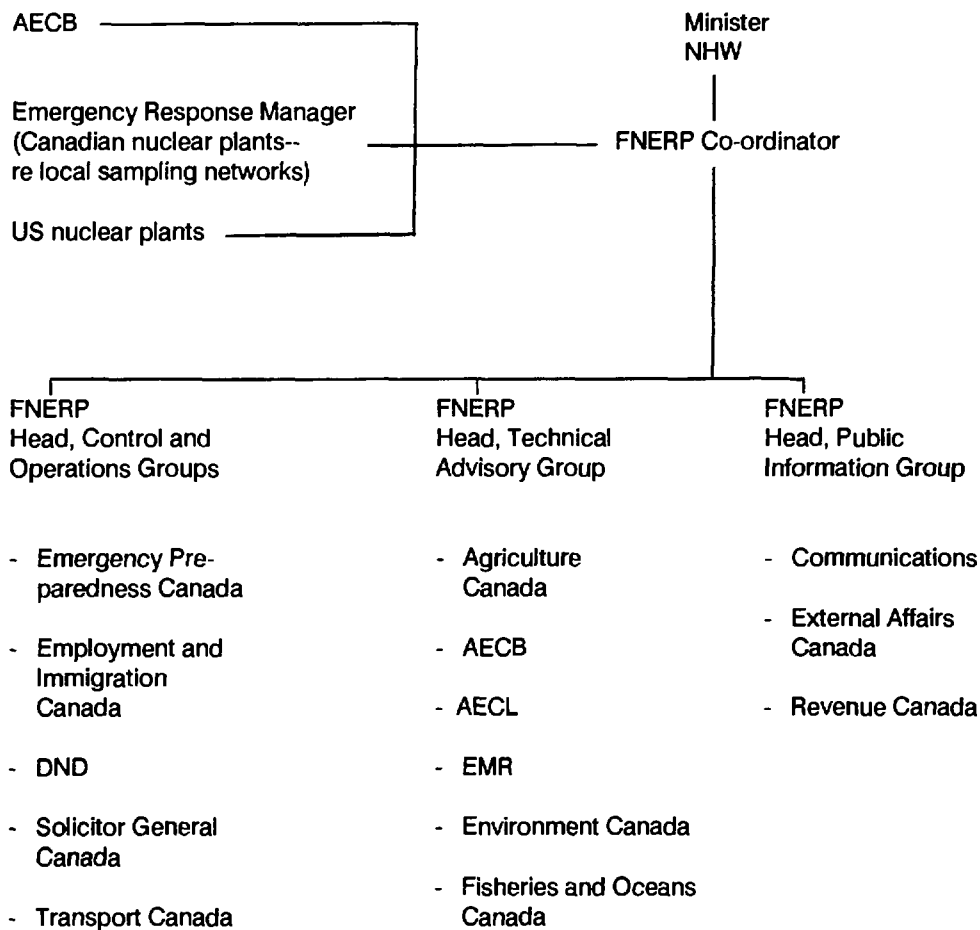
**(e) Comments on development of FNERP**

103. The need for a central federal co-ordinating agency was exemplified by the Three Mile Island accident in March 1979. A long series of interdepartmental discussions took place until finally, in 1984, the Prime Minister assigned NHW as the lead agency. The 5-yr hiatus in establishing this nuclear emergency arrangement presumably was some measure of the low priority given it by the government. Subsequently, only limited resources were assigned to FNERP by the department because of priorities given to chemical compared with radiation risks, along with the usual restraints imposed on growth of government agencies.

104. The Director of the Radiation Protection Bureau (now BRMD) serves as Federal Co-ordinator for FNERP. FNERP has for its support a well-equipped radiometric and chemical laboratory, which is now perhaps the best unit of its kind in Canada. Little support was given BRMD for emergency measures management, presumably because there were no nuclear emergencies to manage until Chernobyl occurred in May 1986. Because of the very low level of radiation detected in Canada, there was in fact no emergency, although considerable apprehension arose in the public because of the detection of even

Table VI-5

## Notification of FNERP re Nuclear Power Plants

Initiator

trace amounts of radiation. Chernobyl provided an ideal chance for a full dress rehearsal of FNERP, at least as an exercise. This did not occur, and the BRMD action was mostly a departmental exercise with useful inputs from some agencies, such as the Atmospheric Environment Service (AES) of Environment Canada. This agency sampled air and precipitation at weather stations throughout Canada. AES also did some remarkable forecasting of time and place of arrival of radiation from Chernobyl. Provincial agencies also contributed to the radiation surveys.

105. Severe weaknesses in the FNERP system showed up in the communications side, with the public, the media, and the many federal and provincial government agencies.

106. A lack of established radiation standards for air, water, and foods constituted another problem, not only for FNERP, but in general internationally; only now is the situation beginning to show some semblance of consensus.

107. A critique of the federal government's response to the Chernobyl accident (Prince 1987) was prepared by me, as a consultant to NHW.

108. During the past year, some progress has been made with respect to FNERP by improved co-ordination within NHW, with the retention of a communications consultant, and by consultation with Emergency Preparedness Canada. There remain problems on space and accommodation for housing an operations group, with little likelihood of early resolution in view of government curtailment of expenditures.

109. There are some promising signs of increased government concern with emergency measures, as seen by the introduction to Parliament for first reading by the Minister of DND of Bill 76, "An Act to provide for emergency preparedness and to make related amendments to the National Defence Act." A companion, Bill 77, has several parts--Part I deals with public welfare emergencies; the other parts are more directly concerned with national security.

110. A further indication of government concern for emergency measures is that the Auditor General has under way a review of emergency measures agencies in the federal government, which may lead to some rationalisation among the many such organisations.

**J. Transboundary Emergency Measures Agreements: US/Ontario, Interprovincial, and International**

111. In the terms of reference of this study, one item was "the existing agreements between American States and Canadian Provinces dealing with the possibility of an accident at one of the American nuclear generating stations located near the Canadian border." This instruction was interpreted to include interprovincial agreements for nuclear emergency measures. A brief statement is also included on international agreements.

**(a) Agreements between Ontario and adjacent US states**

112. A Provincial Plan for Transborder Emergencies, dated August 1986, has been prepared as part of Ontario's NEP (EPO 1986c). As stated in the plan, it was the intention to negotiate arrangements for emergency measures with the states of New York, Ohio, and Michigan, as well as with the Province of Quebec.

113. Considerable progress has been made in making satisfactory arrangements with the three states and with Quebec.

114. The Co-ordinator, EPO, has sent letters requesting agreement to the three adjacent Great Lakes states. The letters were sent to each of the State Emergency Management Organizations requesting prompt notification of Ontario of any nuclear emergency at any reactor in the state. Ontario has also requested permission to send its Ontario Liaison Team to an appropriate location in each state to obtain first-hand information in the event of an accident. Ontario has offered reciprocal arrangements for state liaison teams to come to Ontario in the event of an emergency in the province.

115. New York has sent a draft reply of agreement to be ratified later, if and when legal aspects are approved. Ohio has replied in agreement by letter from a very senior official, the State Adjutant General. Michigan has also agreed fully with Ontario's request, in a reply from the head of the State Emergency Management Organization. Although the states have agreed to the presence of the Ontario Liaison Team, they have not shown much support for reciprocal arrangements, but have not ruled out such action. US federal organisations are aware of Ontario's approach to the three states and have shown a co-operative attitude.

**(b) Agreement between Ontario and Quebec**

116. EMO Quebec was contacted by EPO in the same way as were the three US states, and agreement was reached for emergency co-operation, including the acceptance of the Ontario Liaison Team. In Quebec, the EMO comes under Protection Civile, which was formerly with the Department of Justice but now comes under Service et Approvisionnement. The Gentilly 2 reactor of Hydro Québec is near Trois-Rivières, Québec, some 200 km north-east of the Ontario boundary. Even though its distance from Ontario is much greater than that from most of the US Great Lakes plants, it is useful that an accord has been reached between the two provinces.

**(c) CANDU operators agreement (interprovincial)**

117. An interesting development in emergency measures concerning inter-provincial co-operation in Canada is a documented accord reached by CANDU operators. According to this agreement, operators in Ontario, Quebec, and New Brunswick have agreed to provide the maximum possible assistance to each other as required in the event of a nuclear power plant accident. This agreement is entirely a utility-to-utility arrangement, which presumably would be co-ordinated with provincial and federal government emergency plans.

#### **(d) International agreements**

118. Following the Three Mile Island nuclear reactor accident, improved arrangements were made for notification between members of the Organisation for Economic Co-operation and Development (OECD), Paris. This arrangement is still in effect and could be very useful in some cases of nuclear accidents. However, because the Western nations and Japan are members of OECD and the Eastern bloc countries are not, the information base is not world-wide.

119. In the early days after the Chernobyl accident, the Soviet Union provided very little information to neighbouring countries and to the world community. This lack of notification led to considerable resentment in Europe and elsewhere. As a result of post-Chernobyl meetings at the IAEA in Vienna in July and August 1986, arrangements were made for improved speed of notification by all members of IAEA, including representatives from both the Eastern bloc and the Western group. In the event of a future accident, much better notification on a global basis is assured.

#### **K. Emergency Measures Concerning the Great Lakes**

120. As mentioned in the introduction to this appendix, by far the greatest attention is given to the impact of possible nuclear accidents on the land. This approach is fully justified because of the consequences for people, property, and productive agricultural land. Although some concern is expressed about possible contamination of ground water by radioactive materials, it is not the most serious problem, because highly radioactive, short-half-life elements are largely decayed before reaching the aquifers or are retained in overlying soils.

121. The Great Lakes, their tributary streams, and connecting rivers could be subjected to direct fall-out of radioactive material from a nuclear accident to just as great an extent as corresponding land areas. On the Canadian side of the international boundary, there are eight nuclear power reactors at Bruce NGS on Lake Huron near Kincardine, Ontario. On the Canadian shore of Lake



Ontario, eight reactors are located at Pickering NGS near Ajax, Ontario. Four reactors are under construction at Darlington NGS about 20 km east of Pickering, near Bowmanville, Ontario. Of less direct concern for impact on the Great Lakes is the Gentilly 2 plant of Hydro Québec, located 13 km east of Trois-Rivières, Quebec, on the south bank of the St. Lawrence River. Gentilly 2 is about 200 km north-east of the Ontario border and about 400 km north-east of the east end of Lake Ontario.

122. On the US side of the Great Lakes, four nuclear reactors are located on Lake Ontario and three plants on Lake Erie, one of which is under construction. Lake Michigan, lying entirely in the United States, has four plants on the eastern shore of the lake in the State of Michigan. Others are located on the western shore of Lake Michigan, but are not listed here because of their greater distance from Ontario. Table VI-6 lists the Canadian and US plants having potential for impact on the waters of the Great Lakes and the adjacent terrain on both sides of the border.

123. From the data in Table VI-6 and the distribution figures in Table VI-7, it can be seen that Canada produces 60% and the United States 40% of the nuclear power in shoreside stations on the basis of plant ratings of operating units plus those under construction. For the Canadian total only, some measure of the enormous output potential (13 700 000 kW) would provide more than 500 W of continuous power per capita (population 25 million in Canada) or about 1500 W per capita to the 9 million residents of Ontario. Another measure of the magnitude of nuclear power in Ontario is a comparison with the largest single source of hydro power in the province, the Sir Adam Beck No. 2 plant at Queenston, with installed capacity of 1 223 600 kW (Thompson and Mucklestone 1987), which is equivalent to one Pickering unit plus one Bruce unit.

124. With no coal or oil resources, with a minor production of natural gas in the bed of Lake Erie, and with little reserve of untapped hydro power sites in the province, it is clear why nuclear power has been so quickly developed to utilise Ontario's only major indigenous energy resource--uranium. It appears that nuclear power is and will continue to be the principal source of electric

Table VI-6

## Power Reactors In or Near Ontario and Research Reactors in Canada\*

## A. POWER REACTORS

<u>Plant</u>	<u>Type</u>	<u>Location</u>	<u>In-service date</u>	<u>Net power (MWe)</u>	<u>Status as of October 1987</u>
<u>AECL/Ontario Hydro nuclear generating stations</u>					
1. Nuclear Power Demonstration (NPD)	CANDU	Rolphon (Ottawa Valley)	1962	21.5	Decommissioning, shut down 24 July 1987
2. Douglas Point NGS	CANDU	Bruce Township (Lake Huron shore)	1968	206	Decommissioning, shut down 4 May 1984
<u>Ontario Hydro nuclear generating stations</u>					
3. Pickering A NGS	CANDU	Pickering (Lake Ontario shore)			
Unit 1			1972	515	Returned to service after 1983-87 outage in September 1987
Unit 2			1971	515	Will return to service in 1988, after outage since 1983
Unit 3			1972	515	Operating. Spring 1989 outage scheduled for retrofits, inspection, and routine maintenance (90 d)
Unit 4			1973	515	Operating. Summer 1988 outage scheduled for retrofits, inspection, and routine maintenance (112 d)
4. Pickering B NGS	CANDU	Pickering (Lake Ontario shore)			
Unit 5			1983	516	Operating
Unit 6			1984	516	Operating
Unit 7			1985	516	Operating
Unit 8			1986	516	Operating

Table VI-6 (cont'd)

<u>Plant</u>	<u>Type</u>	<u>Location</u>	<u>In-service date</u>	<u>Net power (MWe)</u>	<u>Status as of October 1987</u>
5. Bruce A NGS	CANDU	Huron Township (Lake Huron shore)			
Unit 1			1977	759	Operating**
Unit 2			1977	769	Operating**
Unit 3			1978	759	Operating**
Unit 4			1979	769	Operating**
6. Bruce B NGS	CANDU	Huron Township (Lake Huron shore)			
Unit 5			1985	835	Operating
Unit 6			1984	837	Operating
Unit 7			1986	837	Operating
Unit 8			1987	837	Operating
7. Darlington NGS	CANDU	Town of Newcastle (Lake Ontario shore)			
Unit 1			1989	881	Advanced construction phase
Unit 2			1989	881	Advanced construction phase
Unit 3			1991	881	Under construction
Unit 4			1992	881	Under construction
<u>AECL power reactor</u>					
8. Gently 1	CANDU/ BLW	13 km east of Trois-Rivières	1971	250	Decommissioned 1986 (AECL)
<u>Other Canadian power reactors (not AECL or Ontario Hydro)</u>					
9. Gently 2	CANDU	13 km east of Trois-Rivières	1983	640	Operating (Hydro Québec)
10. Point Lepreau	CANDU	39 km south of Saint John	1983	640	Operating (New Brunswick Electric Power Commission)

Table VI-6 (cont'd)

<u>Plant</u>	<u>Type</u>	<u>Location</u>	<u>In-service date</u>	<u>Net power (MWe)</u>	<u>Status as of October 1987</u>
<u>Illinois State power reactors</u>					
11. Dresden Station		Morris			
Unit 2	BWR		1970	772	Operating
Unit 3	BWR		1971	773	Operating
12. Zion Plant		Zion			
Unit 1	PWR		1973	1040	Operating
Unit 2	PWR		1974	1040	Operating
13. Quad Cities Station		Cordova			
Unit 1	BWR		1973	769	Operating
Unit 2	BWR		1973	769	Operating
14. La Salle County Station		Seneca			
Unit 1	BWR		1984	1078	Operating
Unit 2	BWR		1984	1078	Operating
15. Byron Station		Byron			
Unit 1	PWR		1985	1120	Operating
Unit 2	PWR		1987	1120	Operating
16. Braidwood Station		Braidwood			
Unit 1	PWR		1987	1120	Operating
Unit 2	PWR		1988	1120	Under construction
17. Clinton		Clinton			
Unit 1	BWR		1986	950	Operating
<u>Michigan State power reactors</u>					
18. Big Rock Point	BWR	Charlevoix County	1965	72	Operating
19. Palisades	PWR	Van Buren County	1971	882	Operating

Table VI-6 (cont'd)

<u>Plant</u>	<u>Type</u>	<u>Location</u>	<u>In-service date</u>	<u>Net power (MWe)</u>	<u>Status as of October 1987</u>
20. D.C. Cook	PWR	Berrien County			
Unit 1			1975	1020	Operating
Unit 2			1978	1060	Operating
21. Erico Fermi	BWR	Monroe County	1986	1093	Operating
<u>New York State power reactors</u>					
22. Nine Mile Point	BWR	13 km north-east of Oswego			
Unit 1			1969	610	Operating
Unit 2			1988	1080	Operating
23. J.A. Fitzpatrick	BWR	13 km north-east of Oswego	1975	821	Operating
24. Indian Point	PWR	40 km north of New York City			
Unit 2			1974	864	Operating
Unit 3			1976	965	Operating
25. R.E. Ginna	PWR	20 km east-north-east of Rochester	1970	470	Operating
<u>Northern New England States power reactors</u>					
26. Main Yankee	PWR	Wiscasset, Maine	1972	810	Operating
27. Seabrook		Seabrook, New Hampshire			
Unit 1	PWR		-	1198	Opening deferred
Unit 2	PWR		-	1198	Opening deferred
28. Vermont Yankee Station	PWR	Vernon, Vermont	1972	504	Operating
<u>Ohio State power reactors</u>					
29. Davis-Besse	PWR	30 km east of Toledo	1977	906	Operating

Table VI-6 (cont'd)

<u>Plant</u>	<u>Type</u>	<u>Location</u>	<u>In-service date</u>	<u>Net power (MWe)</u>	<u>Status as of October 1987</u>
30. Perry	BWR	35 km north-east of Cleveland	1987	1205	Operating
<u>Wisconsin State power reactors</u>					
31. La Crosse (Genoa)	BWR	La Crosse	1969	48	Operating
32. Point Beach Station		Two Creeks			
Unit 1			1970	495	Operating
Unit 2			1972	495	Operating
33. Kewaunee Plant	PWR	Kewaunee	1974	515	Operating

## B. RESEARCH REACTORS

<u>Station/Unit</u>	<u>Operator</u>	<u>Location</u>	<u>Start-up date</u>	<u>Heat output (W)</u>	<u>Status as of 31 March 1987</u>
NRU	AECL	Chalk River, Ont.	1957	137 000 000	
NRX	AECL	Chalk River, Ont.	1947	42 000 000	In hot stand-by condition since 1986. Operates 4-8 h weekly
Swimming Pool	McMaster Univ.	Hamilton, Ont.	1959	5 000 000	
Pool Test Reactor (PTR)	AECL	Chalk River, Ont.	1957	100	
Zed 2 Experimental Reactor	AECL	Chalk River, Ont.	1957	150	
ZEEP	AECL	Chalk River, Ont.	1945	10	Retired 1970
Slowpoke Demonstration Reactor	AECL	Whiteshell, Man.	1987	2 000 000	Developmental, now at criticality
WRI	AECL	Whiteshell, Man.	1965	60 000 000	Closed down, 1985
Slowpoke II	AECL	AECL - Tunney's Pasture, Ont.	1971	20 000	Retired 1984
Slowpoke II	Univ. of Toronto	Toronto, Ont.	1976	20 000	

Table VI-6 (cont'd)

<u>Station/Unit</u>	<u>Operator</u>	<u>Location</u>	<u>Start-up date</u>	<u>Heat output (W)</u>	<u>Status as of 31 March 1987</u>
Slowpoke II	École Polytechnique	Montreal, Que.	1976	20 000	
Slowpoke II	Univ. of Edmonton	Edmonton, Alta.	1977	20 000	
Slowpoke II	Saskatchewan Research Council	Saskatoon, Sask.	1981	20 000	
Slowpoke II	AECL	Kanata, Ont.	1984	20 000	
Slowpoke II	Royal Military College	Kingston, Ont.	1985	20 000	

\* Excluding US military reactors.

\*\* These units can produce 300 MWt of steam in addition.

Table VI-7

Distribution of Nuclear Power Production  
Between Canada and the United States

<u>Location</u>	<u>Production (MWe)</u>		
	<u>Canada</u>	<u>USA</u>	<u>Total</u>
Lake Ontario	7 720	3006	10 726
Lake Erie	None	3211	3 211
Lake Huron	6 000	None	6 000
Lake Michigan	None	2926	2 926
<b>Total</b>	<b>13 720</b> <b>(60%)</b>	<b>9143</b> <b>(40%)</b>	<b>22 863</b>



power, and society will have to learn to live with it or accept a more costly source of energy, such as coal, with its inherent acid rain and respiratory problems.

125. Emergency measures are part of the nuclear package, just as fire trucks and smoke detectors are part of normal living. It is hoped that emergency measures for nuclear accidents will continue to be required infrequently. Yet the consequences of a major accident can be extreme in all respects. In this section, the impact on the body of the Great Lakes is considered.

126. Nearly 37 million people live in the Great Lakes Basin--29.3 million in the United States and 7.4 million in Canada. Most of the population relies on withdrawals from the lakes for domestic and industrial water supplies. Alternatives for meeting the supplies from the lakes because of possible radioactive contamination are difficult to provide and in most cases would require months to years to construct by diversions, well fields for ground water, catchment basins, or other sources.

127. Scenarios of various source terms of radioactive release could be used over various areas of one of the lakes. For the Chernobyl accident, it is estimated that at least  $1 \times 10^{18}$  to  $2 \times 10^{18}$  Bq\* ( $\sim 3 \times 10^7$  Ci) were released (AECB 1987), based on measurements in the Soviet Union and especially in a 30-km-radius zone about the plant. With the Chernobyl release as the maximum, scaled-down amounts could also be used as source terms to evaluate their impacts on water supplies. Such evaluations are difficult to make because of the need to make assumptions of dilution, mixing rates, sedimentation, and current distribution.

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\* A comment on the two systems of radioactivity measurement now in use must be made. The old system of curies, rems, and rads is in the process of being replaced by the Système International (SI) with becquerels, sieverts, and grays. The use of the old system is being continued in the United States, Soviet Union, and Ontario. Canada (federal) and most European countries are changing over or have changed over to the SI. It is suggested here that in Canada, or at least in Ontario, for the next decade or so, both unit systems should be used, with the old units in parentheses, until complete change-over is made. Discussion should be held on this point between Ontario and federal agencies.

128. The postulated distribution and uptake of radioactivity in Lake Ontario can be modelled, as there has been considerable research on its current patterns, thermal regimes, and physical limnology in general. Early work was done by scientists at the Canada Centre for Inland Waters (CCIW) in support of the 1964 Reference of the governments of Canada and the United States to the International Joint Commission (IJC). This work was done in the mid- to late-1960s and is recorded in an Advisory Board (1969) report to the IJC. Studies of lake circulation showed that pollution did cross the international boundary and that each country was in effect polluting the other, thereby invoking the need for the IJC to propose corrective measures.

129. The Advisory Board study of Lake Ontario currents established that the statistical circulation of the main mass of lake water is counter-clockwise, westbound on the Canadian shore and eastbound on the US shore. At the same time, the Advisory Board recognised the existence of complex circulation patterns and more rapid flows in the near-shore areas than in the main water mass. The influence of wind effects was also noted, and comments were made on the need for, but the great complexity of, any mathematical model to rationalise the shore effects.

130. With the construction and operation of the two major nuclear generating stations at Pickering and Bruce, it became necessary to fully understand the characteristics of the discharge plumes of heated water from the turbine condensers and other sources in the plants. In addition, it was essential to know the distribution and dilution of small quantities of radioactive materials, such as tritium, released to the lakes.

131. Beginning in the early 1970s, many investigations were made by academics and institutional scientists and engineers concerning the near-shore current structures. Most of the work was done with respect to conditions within 12 km of the shore, and current flows were found to be highly wind-directed, with movement dominantly eastward in Lake Ontario, for example, because of the

high frequency of westerly wind sets. Several reports\* (Advisory Board 1969; CCIW/Ontario Hydro 1976; Bull et al. 1980; Bull and Murthy 1980; OME/Ontario Hydro/ NWRI 1981; Murthy and Dunbar 1981; Lam and Durham 1984; Simons et al. 1985; Murthy et al. 1986) illustrate the major programme that has been undertaken, the excellent work done, and the high degree of co-operation between Ontario Hydro and federal and provincial agencies. It seems clear that there is now a good understanding of near-shore currents in the vicinity of the nuclear plants and their possible impact on water supplies some distance away.

132. It now appears necessary to address the question of co-ordinating near-shore current characteristics with the main circulation of the lakes as a whole, on the assumption that a nuclear accident could spread radioactivity over, for example, a 50-km radius of the lake surface, in the case of Lake Ontario. If an accident were to occur at the Oswego, New York, site, would the dominant shore currents tend to remove most of the local contamination via the St. Lawrence River, thus affecting Montreal intakes, or would there be sufficient counter-clockwise circulation in the main body of the lake to bring significant radioactivity to the western end of Lake Ontario, affecting municipal water supplies at Toronto, Hamilton, and other cities?

133. Similarly, if an accident were to occur at the Pickering or Darlington sites, affecting most of the western half of the lake, would the dominant easterly shore currents rapidly transfer the contaminants to the east, relieving the stress on west-end municipal water supplies, thus causing stress on east-end intakes? Or would a central counter-clockwise gyre retain a significant amount of radioactivity in the main mass of water, releasing it gradually to the shore currents? Similar questions must be asked regarding a potential accident at the Bruce site, where perhaps the largest concentration of nuclear power in the world is located. Canada and Ontario have a particular responsibility to address this question at Bruce, because it is the only nuclear plant on Lake Huron. Would an accident at Bruce be likely to affect the water supplies as far away

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\* Copies of these publications were provided by the National Water Research Institute, CCIW, Burlington, Ontario.

as the Detroit-Windsor complex? Would it have any impact on Georgian Bay if most of the fall-out were to occur on Lake Huron?

134. The answers to some of these broader questions may be known, or reasonable guesses may be made as to their significance. However, it appears as if some major field programmes may be necessary to establish the broader aspects of lake circulation dynamics. This question is one that ONSR may wish to address to the Ontario government for the support of lake-wide investigations and for consideration of alternative sources of water supply in the event of, for example, Lake Ontario contamination.

135. With regard to alternative sources of water supply, a feasibility study could be made of a pipeline from Georgian Bay via Lake Simcoe (as reservoir or passing under it) to Toronto, with branches to Hamilton and Oshawa. Perhaps such considerations are already in hand, because the contamination of Lake Ontario by chemical poisons from upstream and local outfalls is already a problem. Unlike a nuclear accident, such an event is not a low-probability risk for the future.

136. With regard to a possible pipeline for water supply from Georgian Bay, the following data on elevations above sea-level are of interest:

Georgian Bay	Elevation 176.2 m (577 ft)	
Lake Simcoe	Elevation 218.8 m (718 ft)	Lift 42.6 m (140 ft)
Lake Ontario	Elevation 74.4 m (243 ft)	Fall 144.4 m (475 ft)

137. Assuming 455 L (100 gal) per person per day for 4 million people in the Oshawa-Toronto-Hamilton region, the flow requirement would be 1820 million L, or 1.82 million m<sup>3</sup>/d. A flow rate of about 21.25 m<sup>3</sup>/s (750 ft<sup>3</sup>/s), equivalent to that of a medium to small river, would be needed; by comparison, the Niagara River delivers 5800 m<sup>3</sup>/s (205 ft<sup>3</sup>/s) to Lake Ontario.

138. There is some possibility of power generation from the large fall from Lake Simcoe, to off-set the lift from Georgian Bay.

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## Appendix VII

### The Safety Implications of the Legal, Regulatory, and Organisational Framework Within Which Ontario's CANDU Reactors Operate

by  
Margaret C. Grisdale

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## A. Preamble

1. I have read the briefs pertaining to regulation that were submitted to the Ontario Nuclear Safety Review (ONSR), some of the principal background literature on the Atomic Energy Control Board (AECB)--the Doern (1976) critique and the Sims (1982) history were particularly useful--and certain regulatory and consultative documents of the AECB (see Annex VII-2) and of the US Nuclear Regulatory Commission (US NRC; see Annex VII-2). The conclusions and recommendations of previous studies of the Ontario nuclear power industry were also consulted (see Annex VII-2). I did, of course, peruse the Atomic Energy Control (AEC) Act and other relevant federal and provincial acts and regulations and referred to them throughout the preparation of this paper. A great deal of valuable information was collected through personal interviews and meetings, notably with representatives of the AECB, US NRC, British Nuclear Installations Inspectorate (NII), and Ontario Hydro. A complete bibliography (Annex VII-2) and list of interviews (Annex VII-1) are appended.

2. The AEC Act of Canada (1946, revised in 1954) is, unquestionably, the principal item of legislation in the regulation of the nuclear power industry in Ontario. There are, however, various other pieces of federal and provincial legislation that bear upon this industry and its safe operation. As well, account must be taken of the managerial structure of Ontario Hydro, the organisation that owns and operates Ontario's CANDU (Canada Deuterium Uranium) reactors, and the ethics and professional dedication of the people who form this organisation. When the concerns raised by Review briefers over the effectiveness of this framework are addressed, it becomes apparent that the perception of the effectiveness of the nuclear regulatory system must be considered as well as the actual effectiveness, and that safety considerations arise as soon as a nuclear power plant is contemplated.

**B. The Atomic Energy Control Act, 1946, and the Atomic Energy Control Board (AECB)\***

Whereas it is essential in the national interest to make provision for the control and supervision of the development, application and use of atomic energy, and to enable Canada to participate effectively in measures of international control of atomic energy which may hereafter be agreed upon; Therefore, His Majesty, by and with the advice and consent of the Senate and House of Commons of Canada, enacts as follows: . . . The Atomic Energy Control Act, 1946.<sup>1</sup>

3. Thus did the Government of Canada, in 1946, establish that the matter of atomic energy was an area of federal jurisdiction. The basis for this decision was the "peace, order and good government" clause of the Constitution Act, 1867 (formerly the British North America Act, 1867) and the interpretation of this clause as giving the federal Parliament power to legislate in matters of "national concern."<sup>2</sup> This interpretation has been upheld by two major judicial decisions: Pronto Uranium Mines Ltd. v. The Ontario Labour Relations Board et al.; Algom Uranium Mines Limited v. The Ontario Labour Relations Board et al., (1956) O.R. 862 (Ont. H.C.); and Denison Mines Ltd. v. Attorney General of Canada, (1973) I.O.R. 797 (Ont. H.C.)<sup>3</sup>

4. The AECB was created by section 3(1) of the AEC Act in 1946 and has regulated the Canadian nuclear industry since then with some modification and expansion of its powers. The AEC Act is short, the 1954 revised version covering only nine pages in both official languages. Although it grants sweeping powers to the AECB, the AEC Act is silent about how such powers

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\*In this appendix, "AECB" refers to the organisation; "Board" refers to the five-member board.

<sup>1</sup>The Atomic Energy Control Act, 1946 [Canada]. 10 George VI, Chapter 37.

<sup>2</sup>Lang Michener Lash Johnston, 1987a, The legal, regulatory and constitutional framework within which Ontario Hydro's CANDU nuclear reactor plants operate. A submission to the Ontario Nuclear Safety Review. Toronto, 8 September. pp. 9-10.

<sup>3</sup>Lang Michener, 1987a. pp. 10-16.

shall be exercised. Sections 7 to 10 of the AEC Act establish that the AECB is a self-regulating body, subject only to directions from the designated Minister, currently the Minister of Energy, Mines and Resources (EMR), and the Governor in Council (the federal Cabinet). The AECB is required by section 20(1) of the AEC Act to report annually to Parliament through the designated Minister, and its annual requests for funds are reviewed by parliamentary committee. The AEC Act was amended in 1954 to remove the development of the nuclear industry in Canada from the responsibilities of the AECB and to turn this function over to the newly created Atomic Energy of Canada Limited (AECL). (AECL research facilities are licensed by the AECB by means of a single licence for a site, covering all nuclear facilities on that site.) With the 1954 amendment of the AEC Act, the AECB retained control of "all aspects of nuclear facilities, substances and equipment, to assure that such facilities, substances and equipment are utilized with proper consideration of health, safety and security."<sup>4</sup>

5. The 1946 AEC Act, section 9, gave the AECB power to make Regulations by means of which the AECB controls all aspects of the nuclear industry in Canada, and many of the regulatory powers listed could be exercised directly by the Board without the approval of Cabinet.<sup>5</sup> The development of the Regulations is indicative of the evolution of the AECB and of the Canadian nuclear industry.

6. The Board is composed of five members, one of whom is appointed President and Chief Executive Officer and who is the only full-time member. The appointments are made on the informal recommendation of the Prime Minister at the pleasure of the federal Cabinet, but the part-time members are usually appointed for terms of 2 yr. The President of the National Research Council of Canada (NRCC) is an ex officio member. The Board exercises its responsibilities with the assistance of its staff, currently numbering about 270

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<sup>4</sup>AECB, 1987. Annual Report 1986-87. The Atomic Energy Control Board. Minister of Supply and Services Canada, Ottawa. p. 1.

<sup>5</sup>Sims, G.H.E., 1982. A History of the Atomic Energy Control Board. Minister of Supply and Services Canada, Ottawa (second printing). p. 24.

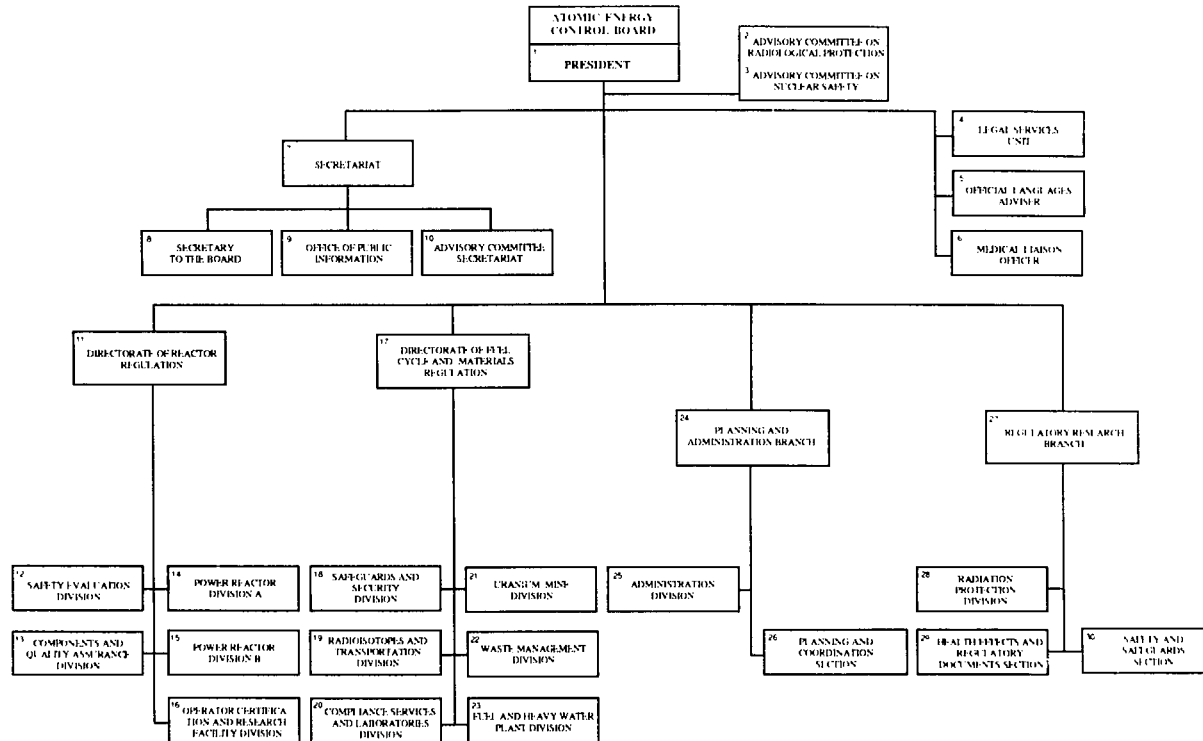
(Figures VII-1 and VII-2), and of standing advisory committees. The size of the Board and the method by which it is appointed, the size and composition of the AECB staff, and the Board's reliance on advisory committees have been the subjects of public comment and criticism.

7. The AECB's status in comparison with that of the nuclear power regulatory bodies in the United States, the United Kingdom, and France is shown in Table VII-1. It should be noted, however, that these three countries have larger populations than that of Canada, and that the regulation of nuclear power in these other countries may be affected by the fact that they have nuclear weapons programmes. Staff of the Review conducted personal meetings with representatives of the US NRC and of the UK NII; a consultant's report was commissioned on the differences between the US and Canadian systems of regulating nuclear power; and a consultant spent a day in Paris with senior representatives of the French Service central de la sûreté des installations nucléaires (SCSIN) and produced a detailed report on its method of operation. The comparison of Canadian nuclear power regulation with that of the United States and the United Kingdom was most useful because of the similarities in the government and culture of these three countries. The comparison was, however, extended to France because of the extent to which France relies on nuclear power (in 1986, 70% of all electricity used in France and 75% of all electricity produced by Electricité de France [EDF] came from the nuclear source) and the relative acceptance of the use of nuclear power in France.

8. The exercise of comparison helped the Review to focus upon many of the important regulatory issues. None of the other systems examined, however, was identified as a model for the regulation of nuclear power in Ontario and in Canada. The US legalistic, prescriptive way of regulation is, as stated elsewhere in this document, not suited to the Canadian approach to public administration. Furthermore, the US system has the inherent danger of leading to a false sense of security, to complacency, and to lack of initiative on the part of the utilities. Not surprisingly, given that much of Canadian governmental practice is borrowed from Great Britain, the UK system of regulation was much more appealing from a Canadian perspective. The organisational structure of

**Figure VII-1**

**Organisation chart, Atomic Energy Control Board,  
9 September 1987**



**Figure VII-2 1986/87 AECB staff effort by project**

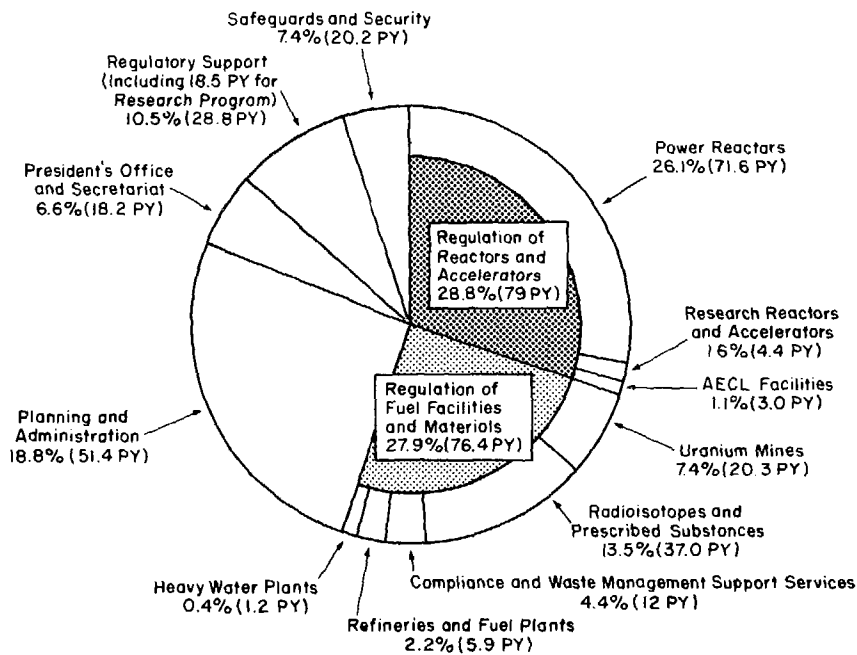


TABLE VII-1

## A Comparison of the Regulation of Nuclear Power Industries

<u>Characteristics</u>	<u>Canada</u>	<u>United States</u>	<u>United Kingdom</u>	<u>France</u>
Ownership of utilities	State	Some state, largely private	State	State
Responsibilities of regulators	Commercial reactors Non-government research reactors Materials handling licences High-level waste disposal Uranium mines Uranium mills Accelerators	Commercial reactors Non-government research reactors Non-defence reactors Materials handling licences High-level waste disposal Uranium mills Mill tailings	All licensed nuclear sites, <sup>a</sup> including: - commercial reactors - fuel reprocessing plants - fuel manufacturing plants - high-level waste disposal sites	Commercial reactors Non-government research reactors Government reactors Materials handling licences <sup>b</sup> High-level waste disposal Accelerators
No. of utilities regulated	3	60	3	1
No. of reactor vendors	1	4 (17 architect-engineers)	1	1 <sup>c</sup>
No. of commercial reactors				
- operating	18	107	40	49 <sup>d</sup>
- under construction	4	14	1	14
Agency employees	270	3400	160	170 (532) <sup>e</sup>
Employees devoted to reactor regulation (approximate)	120	2200	120	140 (398) <sup>e</sup>
No. of employees per reactor	5.5	18.5	2.9	2.70 (6.32) <sup>e</sup>
No. of lawyers in agency	2	96	0 <sup>f</sup>	0 <sup>g</sup>
Annual budget	\$24 million (Cdn)	\$410 million (US)	£9.5 million	36 million French francs (FF) <sup>h</sup>
Research budget	\$4.6 million (Cdn)	\$110 million (US)	£1.6 million	4 million French francs <sup>h</sup> (284) million French francs <sup>i</sup>

a Under the Health and Safety Work Act, the NII has two sets of powers: licensing of nuclear facilities, and health and safety.

b Including enrichment, fuel fabrication, reprocessing, and waste management.

c Excluding the fast breeder reactors.

d As of 31 December 1986.

e Including the personnel from IPSN (Institut de protection et de sûreté nucléaire), who perform safety analyses for the regulatory agency.

f Draws upon panel of four lawyers in Health and Safety Executive.

g As in UK, draws upon central pool of legal experts.

h 1 FF = \$0.23 (Cdn).

i 280 million FF go to CEA (Commissariat à l'Énergie Atomique), equivalent to AECL, for its technical support (IPSN belongs to CEA). Thus, the French situation seems to be similar to that in the UK.

Source: Based on Ahearn, 1987, Table 1.



the NII within the Health and Safety Executive is, however, not something that could be duplicated in Canada given our federal system of government. Canadian politics and culture require a uniquely Canadian approach to the regulation of nuclear power. This is being provided by the AECB, and the issue in question is how this approach can be strengthened.

9. The AECB exercises control over nuclear generating stations in two ways: through the approval and licensing processes and through the qualification of the control room operators. Although section 8 of the Regulations gives the AECB power to exempt the utility running a nuclear facility from the requirements of a licence,<sup>6</sup> the approval and licensing processes are in fact applied rigorously. The AECB's requirements of licence applicants contrast dramatically with the detailed and voluminous requirements of the US NRC.<sup>7</sup> The US regulatory documents are so prescriptive that it has been suggested that the regulator in effect designs the nuclear generating stations and determines how they will be run. The approach of the AECB is similar to that of the British NII in that it takes the view that the onus is on the utility to design its own plants and to prove to the regulator that they are safe.

10. The utility is in contact with the AECB from the time it declares its intention to build a nuclear generating station. The AECB's requirements can and frequently do extend well beyond those contained in its regulatory and consultative documents. The fact that many of its requirements are not formally documented has been seen as a weakness of the process, but the AECB maintains that it can require a licensee to comply with any requirement that it considers necessary. In addition to its own requirements and guidelines, the AECB applies independently developed codes and standards in its oversight of the safety of nuclear power plants.

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<sup>6</sup>Lang Michener, 1987a. p. 26.

<sup>7</sup>Ahcarne, J.F., 1987. A comparison of nuclear power regulation in Canada and the United States. A submission by Resources for the Future to the Ontario Nuclear Safety Review. Washington, 1 October. p. 10.

11. Lord Flowers has expressed the view that "safety is built into the design as it progresses"<sup>8</sup> in his criticism of the lack of NII involvement during the design phase of British nuclear generating stations. (In the 12 yr since this statement was made, NII involvement in the design phase has progressively changed.) It would be contrary to AECB policy to instruct the utility on how to design a nuclear facility, but the AECB is very much involved in the design phase of a nuclear generating station. The AECB is provided by the utility with preliminary design descriptions, safety design guides, draft design manuals, final design manuals, and the accident analyses conducted by the utility. The AECB can and does note potential design defects indicated by these documents and requires the utility to address its concerns. The AECB also requires the utility to have quality assurance (QA) programmes in place during the design stage to ensure the good management of the design work, and it periodically audits these QA programmes.

12. The AECB has the authority to refuse a construction licence if it objects to the site selected by the utility for its nuclear facility. The site acceptance process has been the subject of much criticism from public interest groups, arising mainly from the fact that the AEC Act makes no provision for public hearings at this stage. The AECB could, however, require a public hearing as a result of the 1984 amendment to the Guidelines of the federal Environmental Assessment and Review Process (EARP).

13. During the construction phase of a nuclear power plant, the AECB can, according to section 10(4) of the Regulations, "subject its approval to such conditions as it 'deems necessary in the interests of health, safety and security.'"<sup>9</sup> The AECB's resources do not permit it to maintain on-site inspectors with respect to the construction of a nuclear generating station. AECB inspectors are, indeed, on-site during the latter part of the construction stage, but not to monitor the construction. Rather, they monitor the functions

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<sup>8</sup>Flowers, Sir Brian, 1976 [has since become Lord Flowers]. Nuclear power and the environment. Sixth Report of the Royal Commission on Environmental Pollution. Her Majesty's Stationery Office, London, September. p. 114.

<sup>9</sup>Lang Michener, 1987a. p. 25.

carried out by the utility's operating staff during this stage. The manufacture and installation of pressure-retaining components are particularly crucial aspects of construction, and the AECB would likely inspect these procedures if they were not monitored by the Ministry of Consumer and Commercial Relations (MCCR). As explained in greater detail later in this appendix, the AECB has entered into an agreement with the Ontario MCCR for the latter to undertake inspections of pressure-retaining vessels on its behalf. The MCCR, for example, currently has two inspectors resident on the construction site of the Darlington Nuclear Generating Station (NGS). Just as it does during the design phase, however, the AECB conducts audits of the utility's QA programme during the construction phase.

14. The AECB resident inspectors (properly called "resident engineers") move onto the nuclear generating station site at about the same time as the utility's operating staff and "in association with specialists based in Ottawa review design, construction and safety analyses, and monitor the commissioning of reactors."<sup>10</sup> An operating licence is granted only after the utility has fulfilled the requirements of section 9 of the Regulations and has complied with the conditions imposed with respect to construction approval.<sup>11</sup> Agreements reached between the AECB and the utility with respect to the operation of the station are set out as conditions of the operating licence or are in reference documents. The latter, which are enforceable under the licence, are often operating policies and principles developed by Ontario Hydro and approved by the AECB.

15. Representatives of the utility deal with AECB staff throughout the above process. When a segment of the review, i.e., with respect to siting, construction, and operation, is complete, the staff makes recommendations to the Board, which makes a collective decision. The Board is not required by law to make decisions on a judicial or quasi-judicial basis and is thus exempt from judicial review.<sup>12</sup> It is possible for the licence applicant or its representative

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<sup>10</sup>AECB, 1987. p. 4.

<sup>11</sup>Lang Michener, 1987a. p. 26.

<sup>12</sup>Lang Michener, 1987a. p. 27.

to be present when the Board makes a decision with respect to a licence, and the AECB has developed formal procedures for attendance at meetings. Minutes of Board meetings and the documents upon which the Board decisions are made, with the exception of those pertaining to security or of a proprietary nature, are available to the public in the AECB reading room.

16. The AECB enforces its requirements of the utility in several ways. The Regulations provide it with broad powers to request information. In addition to the documents and analyses required during the siting and construction phases, the Board requires the utility to keep detailed records of the operation of a nuclear generating station, and to report all significant events. The AECB, which has maintained resident inspectors from the beginning of the Canadian nuclear power programme, has three or more inspectors resident at each nuclear generating station. This is a practice similar to that followed by the US NRC, but the latter has maintained resident inspectors only since 1980. The AECB, in fact, has more inspectors assigned to each nuclear site than does the US NRC, but, as Ahearne points out, the number per reactor is less, as "most Canadian sites have 4-8 reactors, whereas the US sites have one."<sup>13</sup> The UK NII, on the other hand, does not have inspectors resident on-site, but this is mainly because of the much smaller geographic area that it covers.

17. Questions arise of the AECB inspectors' ability to maintain their independence in the field, and whether, particularly at such remote sites as the Bruce Nuclear Power Development, they might come to identify more with the utility than with the regulator. The Review has asked these questions and has identified no evidence of a problem. The AECB resident inspectors have a fair amount of authority to approve actions of the licensee,<sup>14</sup> and this is seen by both the AECB and the utility as simplifying the administrative and reporting processes.

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<sup>13</sup>Ahearne, J.F., 1987. p. 14.

<sup>14</sup>Ahearne, J.F., 1987. p. 14.

18. Another issue considered by the Review was whether the number of resident inspectors at each station was adequate. The danger of too few inspectors is fairly obvious: the AECB would not be able to monitor adequately the utility's compliance with its regulations. Too many inspectors, however, can be as detrimental to effective regulation as too few: the inspectors may lose sight of broad policy objectives and focus attention on minor details that are rightly the responsibility of the utility. It was the Review's assessment that, whether by design or chance, the right balance of numbers had been struck. The AECB resident inspectors appeared to function effectively and to have gained the respect of the utility representatives with whom they worked.

19. The licence is the AECB's chief means of enforcing its requirements. In the event of non-compliance with licence conditions, the AECB does not have the authority to impose fines upon the utility, as does the US NRC (AECB must go to court to do so, and this is only done with respect to radioisotope licence violations), nor has it shut down plants during the construction and operating stages, as the US NRC has been known to do.<sup>15</sup> It does, however, have this latter power, through its ability to withhold construction approval, to withhold a licence, to withhold a licence renewal, and to impose conditions upon an existing licence. Restrictions were imposed, for example, on the outputs of Douglas Point and Ontario Hydro Bruce A NGSs until these stations met AECB requirements. Flowers makes the point that the British NII,<sup>16</sup> too, has found the licence to be an effective tool with which to enforce its regulations and indicates that although the NII can prosecute the operators of nuclear facilities for licence infractions, it has not needed to do so.<sup>17</sup> Unlike the situation in the United States in which an operating licence is granted for the life of a reactor, the AECB requires the utility to renew a licence every 2 yr.

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<sup>15</sup>Ahearn, J.F., 1987. p. 17.

<sup>16</sup>The NII issues a site licence that identifies a particular area of land to be used to construct and operate a particular reactor, rather than licensing a reactor. Although the UK licence is valid for the lifetime of a reactor, there is a requirement for each reactor to be shut down for maintenance every 2 yr and for approval to start up again.

<sup>17</sup>Flowers, Sir Brian, 1976. p. 113.

Although the AECB has never failed to grant or to renew a licence, it has come close to doing so. During the course of its nuclear operations, Ontario Hydro has received a number of letters asking it in effect to show cause why licences for reactors with which the AECB had identified problems should not be revoked. A former member of Ontario Hydro's senior management who was responsible for responding to seven "show cause" letters indicated that these letters were taken very seriously, and resources were not spared in addressing the AECB's concerns.<sup>18</sup> This is typical of the Canadian and British approach to regulation: the utility is not told how to make its nuclear operations safe, but is asked to prove that these operations are safe. Although some, including Schrecker,<sup>19</sup> see the licence as a meagre tool of enforcement, the economic and political penalties for the utility are considerable should the AECB withhold or revoke a licence.

20. The AECB finds that negative publicity or the threat of it is a very effective lever in obtaining utility compliance with its requirements. Were the AECB to issue an order for a utility to shut down one of its reactors, it would also issue a press release: the AECB believes that a situation requiring such an order is one about which the public should be aware. The utility realises this. Therefore, if the AECB indicates that a reactor should not be running under certain conditions, the utility is likely to shut down voluntarily.

## CONCLUSION

There is no evidence that the AECB lacks the proper means of enforcing its regulations. The licence and the threat of unfavourable publicity if the utility does not comply with licence conditions are adequate tools of enforcement.

21. In addition to its approval and licensing processes, the AECB reserves the right to certify the control room operators and shift supervisors of the utility

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<sup>18</sup>Mencley, D.A., 1988. Letter to the Ontario Nuclear Safety Review, 10 January.

<sup>19</sup>Schrecker, T., 1987. The Atomic Energy Control Board: assessing its role in reactor safety regulation. A submission by Energy Probe to the Ontario Nuclear Safety Review. Peterborough, 1 September. p. 9.

and to approve the appointments of the station managers, the production managers, and health physicists. The training programmes for the operators and shift supervisors are designed and conducted by the utility, but are reviewed by the AECB. The training period is lengthy and demanding. Before these staff members are certified by the AECB, they must pass a series of five AECB written or oral examinations: two dealing with science fundamentals, two dealing with specific aspects of each station, and one dealing with radiological protection. The failure rate is high, and both Ontario Hydro and the union representing the operating staff (CUPE Local 1000) find themselves at odds with the AECB over the examinations: they feel that the AECB's standards are too high, that the AECB is too critical in its marking of the examinations, and that the examinations require excessive memory work. The US NRC also examines operators of nuclear generating stations, but this is a fairly recent development, introduced only after the Three Mile Island accident.<sup>20</sup> The UK NII does not examine operating staff, taking the view that it is the responsibility of the utility to ensure that its plants are operated safely. As a result of the Chernobyl accident and the part played in it by operator performance, however, the NII is considering the possibility of qualifying operators.<sup>21</sup> The French SCSIN also takes the view that the owner is totally responsible for the safe functioning of its nuclear plants, and there is no equivalent of AECB examinations for French operators.

### C. Criticism of the Atomic Energy Control Board

22. As indicated above, the AEC Act of 1946, although all-powerful, was very general as to the specific duties and responsibilities of the Board. The only major revision of the AEC Act was that of 1954, which transferred responsibility for the development of Canada's nuclear industry from the AECB to AECL. The AEC Act's emphasis on secrecy and protection reflects the concerns of the post-war political and cultural climate in which it was drafted.

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<sup>20</sup>Ahearne, J.F., 1987. p. 17.

<sup>21</sup>Interview with UK NII representatives, 5 August 1987.

In the 40 yr that have followed the proclamation of the AEC Act, there have been a great many developments in the nuclear industry in Canada that have increased the demands placed upon the AECB: the use of radioisotopes, the increasing importance of uranium mining and refining, the proliferation of nuclear waste, the production of heavy water, and, of particular importance to this discussion, the establishment and growth of the nuclear power industry.<sup>22</sup> The AECB evolved in size and focus to keep pace with these developments in the industry, and, through its power to enact Regulations, was able to exert its authority in new areas of the nuclear industry without amendments to the AEC Act. Dissatisfaction developed both within and without the AECB over the years concerning its legislative basis, and, after wide consultation, Bill C-14 was introduced in 1977 to amend the AEC Act. Regrettably, it died on the order paper.

23. Many of the concerns prevalent today were addressed by Bill C-14. As Lang Michener point out, most of the dissatisfaction with the AECB "does not stem so much from immediate criticism over demonstrated failings as from increased concern with the adequacy of present checks and balances in view of the potentially catastrophic results of error or bad judgement. One is reminded of the basic principle of the common law judicial system that 'justice must not only be done but must also be seen to be done.'"<sup>23</sup>

24. The AECB is not always perceived to be independent, and its proceedings are not always perceived to be democratic. This leads to questions of accountability and effectiveness. Further questions of accountability and effectiveness are raised by the ad hoc nature of some of the AECB's dealings with licensees and its access to sanctions to enforce its requirements. The overall issue of the effectiveness of the AECB is raised when its size and resources are compared with those of other regulators (see Table VII-1) and with those of Ontario Hydro, the principal nuclear power utility in Canada.

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<sup>22</sup>Domaratzki, Z. and T.J. Molloy, 1987. Nuclear regulation--in tune with the times. Proceedings 27th Annual Conference of the Canadian Nuclear Association, Saint John, New Brunswick. p. 31.

<sup>23</sup>Lang Michener, 1987a. p. 33.



25. Much of the independence issue arises from the history of the nuclear power industry in Canada, notably that it was developed by a handful of pioneering scientists, some of whom are still influential in the industry, and that it was developed by the federal government, which is still actively fostering and promoting nuclear power. This situation and the resulting problems are summarised by Lang Michener:

Because of the small and tight-knit circle of individuals capable of holding a position on a board concerned with such a technologically advanced field as nuclear power, and the fact that the federal government through Crown corporations such as Atomic Energy of Canada Limited (AECL), Eldorado Nuclear Limited and Uranium Canada Limited is Canada's largest nuclear energy entrepreneur, the federal government's conflicting mandate of being the main proponent and chief watchdog of the Canadian nuclear industry has raised doubts about whether the AECB maintains enough independence to function as effectively as it should. This is especially so when the same Minister is responsible for the promotion as well as the regulatory functions.<sup>24</sup>

26. The small pool of technical expertise to be drawn upon in appointing professional staff to the AECB has frequently been raised as an issue of concern. It is true that nuclear power is such a technologically advanced field that the number of people qualified to regulate it is restricted. It is also true that the CANDU system is essentially a stand-alone system, and, as the Review discovered in seeking to engage independent consultants on an international basis, the expertise in the CANDU system is limited to Canada, and most of the experts are employed by Ontario Hydro and AECL. John Ahearne was critical of what he describes as the "revolving door phenomenon" in which scientists and engineers move freely among Ontario Hydro, AECL, and the AECB.<sup>25</sup> He points out that there are regulations prohibiting such freedom of movement in the United States and encourages the AECB to concentrate on drawing its staff expertise from Canadian universities. The British NII, which, like Canada,

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<sup>24</sup>Lang Michener, 1987a. p. 29.

<sup>25</sup>Ahearne, J.F., 1987. p. 24.

faces the problem of regulating a non-universal type of reactor--the MAGNOX, has no such prohibitions and finds the poacher-turned-gamekeeper phenomenon quite acceptable. The NII rarely recruits technical staff under 30 yr of age and seeks those with a solid background in the industry.<sup>26</sup>

27. The Review was unable to find evidence that the recruitment and future career paths of AECB professional staff are matters for concern. As Sims states, "there does not appear to have existed between the Board staff members and their former colleagues in the industry much of the professional 'cosiness' which has been suggested as characterizing the relationship between them."<sup>27</sup> AECB professional staff members usually hold engineering degrees and usually have some prior experience, either in industry or in the military. In numerous conversations with representatives of the AECB and Ontario Hydro about their relationship, the Review formed the impression that although there was little antagonism, the authority of the AECB was clearly recognised and respected. It is not unheard of for an AECB staff member to accept a position in the industry regulated. But this occurs so seldom that it cannot be considered an issue. Restriction of the movement of technical staff, furthermore, could lead to a deterioration of the overall quality of the regulatory body.

## CONCLUSION

Capable technical expertise in Canada is scarce. The scientists and engineers engaged in the nuclear industry are professionals, guided by their professional ethics. The Review has discovered no evidence of an incestuous relationship between the industry and its regulator, nor of staff movement between the AECB and the utilities to the extent that might be described as movement through a "revolving door." The present situation is a healthy one and should be preserved in the future.

28. The pre-eminence of the Canadian government in promoting nuclear power certainly clouds the independence issue, as does the fact that the operators of

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<sup>26</sup>Interview 5 August 1987.

<sup>27</sup>Sims, G.H.E., 1982. p. 129.

the power reactors are provincial government utilities. These facts of the Canadian nuclear power industry notwithstanding, it is worth noting that the nuclear power industries in the United Kingdom and in France are also largely state-developed and state-run. The British have addressed the independence of the regulator issue by removing the NII from the Department of Energy, the sponsoring department for nuclear power, and placing it within the Health and Safety Executive, a regulatory body that is completely independent of the sponsoring government department.<sup>28</sup>

29. The situation in France is different: the research and development of nuclear energy (civil and defence use) are the responsibility of the CEA (Commissariat à l'Énergie Atomique), the equivalent of AECL; the civil use of nuclear power is the responsibility of EDF, the equivalent of Ontario Hydro; and the regulation of civil nuclear reactors is the responsibility of the SCSIN, the equivalent of the AECB. EDF and SCSIN both report to the Minister of Industry, Telecommunications and Tourism. CEA also reports to this minister with respect to safety-related matters. (CEA reports to other ministers with respect to research and defence.) The Minister of Industry is also responsible for safety matters related to defence reactors through a body other than SCSIN.

30. Political conflicts between "promoters," "users," and "regulators" in France are handled by the same minister. This practice is considered preferable to the alternative of having the Prime Minister arbitrate between several ministers if responsibility for nuclear power were divided. One of the reasons that this is not perceived as a problem in France may be the fact that the nuclear power programme there is characterised by a common political will to make it a success.

## CONCLUSION

Public confidence in the AECB as an independent watchdog of the nuclear industry would be increased if

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<sup>28</sup>Under the Health and Safety at Work Act 1974, the NII forms part of the Health and Safety Executive along with the Factory Inspectorate, the Alkali and Clean Air Inspectorate, and the Mines and Quarries Inspectorate.

the Atomic Energy Control Board and Atomic Energy of Canada Limited were to report to Parliament through different ministers of the Crown.

## RECOMMENDATION

1. **THAT** the Atomic Energy Control Board and Atomic Energy of Canada Limited report to Parliament through different ministers of the Crown.

31. Some of the criticism levied against the AECB can be traced to the way in which it perceives itself: "The AECB has always been a technical body which reaches technical conclusions and decisions on health, safety and security matters. The Board has re-affirmed the view that it should retain that mode of operation."<sup>29</sup> Several of the Review's intervenors took the view that this interpretation by the AECB of its role was too narrow. Many of the AECB's decisions, as Schrecker points out, pertain to "complex policy choices about the nature, the acceptable level and distribution of risk."<sup>30</sup> They are much broader decisions than those relating simply to technical matters. In applying the ALARA (as low as reasonably achievable) principle in determining permissible levels of radiation exposure, the AECB takes into account socio-economic considerations. Many of its critics point out that the AECB is thus much more than a technical body and should include among its members representatives of the social sciences and humanities and of the labour force of the nuclear industry. The AECB responds to such criticism by pointing out that there are other bodies in which the social and economic concerns over the use of nuclear power can be debated, and that it does not have the resources to conduct social and economic analyses. If the Board were enlarged, it would, of course, be necessary for Parliament to amend the AEC Act.

32. It is further argued by critics of the AECB that there should be a much greater degree of openness and public involvement in the selection and

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<sup>29</sup>Domaratzki, Z. and T.J. Molloy, 1987. p. 40.

<sup>30</sup>Schrecker, T., 1987. p. 17.

appointment of Board members and in Board decision making. The confusion surrounding the naming of Dr. Ursula Franklin of the University of Toronto as a member of the Board and the subsequent denial of her appointment are cited as an example of the untidiness of the appointment process.<sup>31</sup> The public has the opportunity to influence Board decision making through the consultative document process by means of which the AECB issues papers on matters of significance that are before the Board and invites written comments within a specified period of time. Public participation in Board decision making is seen by some critics, however, as participation in Board meetings.<sup>32</sup> Meetings are not currently open, although there is a formal process by which permission to attend may be sought.

33. The AECB's interpretation of its role with respect to environmental issues has also been criticised as being too narrow. The Canadian Environmental Law Association (CELA) has recommended that "a broad definition of the environment be adopted" by the AECB to ensure that the full environmental impact and socio-economic effects of a nuclear generating station are given full consideration during the licensing process, either by the AECB or by the provincial Ministry of the Environment.<sup>33</sup> This concern is addressed in Section H of this appendix.

34. The way in which the AECB interprets its role and the claimed lack of democracy in its decision making are discussed in detail by Doern in his critique prepared for the Law Reform Commission of Canada, one of the background documents to Bill C-14. He offers two models of regulation:

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<sup>31</sup>Adams, T. and M. Jerrett, 1987. The Atomic Energy Control Board: its role and performance in the regulation of nuclear reactor safety. A submission to the Ontario Nuclear Safety Review. Toronto, 2 September. pp. 18-22.

<sup>32</sup>Adams, T. and M. Jerrett, 1987. p. 23.

<sup>33</sup>CELA, 1987. Regulatory control of nuclear safety. A submission by the Canadian Environmental Law Association to the Ontario Nuclear Safety Review. Toronto, September. p. 54.

Model 1, the professionally-open model, is characterised by a high degree of trust. Its proponents assert that it is internally open, fostering frank criticism and evaluation among professional and technically competent people. Advocates of Model 1 suggest that regulators using this model are viewed by regulated utilities as professionals trying to achieve common goals, health and safety, as well as production. As a result, professionals in the utilities are more likely to reveal to their regulating peers both what is working well, as well as what isn't . . . . . Model II, the democratically-open model, parallels the nuclear regulatory regime in the United States that allows broad participation in regulation-making, licensing and compliance proceedings. Extensive hearings are a component of Model II. So too are greater opportunities for interventions and judicial review. Model II's opponents claim that it promotes confrontation.<sup>34</sup>

Doern considers it appropriate for the AECB to move toward his "democratically open model," acknowledging that it is much more in keeping with the American style of regulation. It is surprising, given that Doern is a political scientist, that he fails to consider at any length the fundamental differences between Canadian and American societies, and whether a Model II system of regulation would be in keeping with the Canadian way of doing things. The professionally open characteristics of the AECB's style of regulation have, however, led to the perception of too close a relationship between the regulator and the regulated. It has further led to serious concerns over the weakness of the ad hoc nature of many of the AECB's licensing decisions and interactions with the utility.

35. Ahearne notes the collegial atmosphere of the relationship between the AECB and the utility, a sharp contrast with the confrontational situation in the United States, and indicates that "the Canadian system is based upon the concept of a family working together."<sup>35</sup> As a result of this trusting

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<sup>34</sup>Doern, G.B., 1976. *The Atomic Energy Control Board: an evaluation of regulatory and administrative processes and procedures.* Prepared for the Law Reform Commission of Canada. Minister of Supply and Services Canada, Ottawa. p. 33.

<sup>35</sup>Ahearne, J.F., 1987. p. 27.

atmosphere, there is a tendency not to write down decisions and, as important, the reasons for these decisions.

36. Although the codified, prescriptive approach to regulation may not be desirable for Canada and is not in keeping with Canadian history and culture, there are some problems with the Canadian system as it exists. Ahearne points out that should a disagreement arise between the AECB and the utility it is hard to settle it if there is no written record of the decision on which the disagreement is based.<sup>36</sup> The AECB makes the final decision in the event of a disagreement, but the utility exercises its right to defend itself: a written record of the basis for the decision under dispute would benefit both parties. CELA sees the problem as being more serious, noting that uncertain or flexible rules invite resistance, and that "given the bewildering mix of guidelines, consultative documents, and advisory papers that constitute AECB design 'requirements,' enforcement by way of prosecution would not be possible."<sup>37</sup> Reliance on undocumented agreements between the AECB and the utility could also present difficulties should the AECB wish to license a new utility, the representatives of which were not part of this shared history.

37. It has also been suggested that in the absence of clear definitions of the AECB's requirements the utility may be confused.<sup>38</sup> The AECB, however, has little doubt that the utility understands its requirements. It points out that an attempt to define and record precise meanings of the terms used in its requirements could be counter-productive: unless the recorded definitions were thoroughly comprehensive, they could provide the utility with the opportunity for less-than-complete response to requirements, and a great deal of time could be squandered on quibbling over definitions and meanings.<sup>39</sup>

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<sup>36</sup>Ahearne, J.F., 1987. p. 27.

<sup>37</sup>CELA, 1987. pp. 4, 5.

<sup>38</sup>Ontario Legislature Select Committee on Ontario Hydro Affairs, 1980. The safety of Ontario's nuclear reactors: final report to the 4th Session, 31st Parliament, 29 Elizabeth II. Toronto, June. pp. 41, 42.

<sup>39</sup>Interview with AECB representative, 27 November 1987.

38. The scarcity of documentation relating to decisions may be indicative of, or the cause of, "the apparent elasticity of reactor licensing requirements."<sup>40</sup> This issue was raised by several Review intervenors<sup>41</sup> and was the subject of critical comment by the 1980 Ontario Legislature Select Committee on Ontario Hydro Affairs. The specific illustration upon which these intervenors focus is the licensing in 1977 of the reactors at the Ontario Hydro Bruce A NGS. The Board overruled its staff, as is its prerogative, and gave Ontario Hydro permission to begin operating the reactors before all the necessary documentation was provided. When the documentation was eventually received, it did not demonstrate hazard, but it failed to prove safety if the reactors were operated above 63% of full power. Yet Ontario Hydro was given permission by the Board to operate at 88% of full power, on the understanding that it would proceed toward the installation of an improved emergency core cooling system to address the problem. The situation was not remedied until November 1986. This delay is partly explained by the complexity of the \$123 million remedial operation. If, however, the Board's reasons for overruling its staff had been more fully documented, this case might not now be surrounded by the suspicion of political interference. Furthermore, the accusation of "elasticity" of the regulations could be better defended by the AECB.

## CONCLUSION

Strengthening of the regulatory system need not do away with the existing atmosphere of mutual respect, nor need it require the adoption of a prescriptive approach to regulation. It does, however, call for more extensive documentation by the AECB of decisions and licensing requirements--decisions and requirements often negotiated in detailed face-to-face discussion with the utility--and the reasons behind these decisions and requirements. Such strengthening would make it easier for the regulator and the utility to settle disputes and more difficult for the utility to resist the requirements of the regulator.

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<sup>40</sup>Schrecker, T., 1987. p. 9.

<sup>41</sup>Adams, T. and M. Jerrett, CELA, Schrecker, T., 1987.



## RECOMMENDATION

2. **THAT** the AECB adopt a more formal approach in its dealings with utilities, and that its licensing and other requirements and the reasons behind them be clearly documented.

39. A question equally germane to the issue of safety is whether the atmosphere of collegiality and trust that exists between the AECB and Ontario Hydro can last. Maturation of the nuclear power industry and the complexity of its regulation could make personal interaction among representatives of the utility and the AECB more difficult. Times are also changing in that, largely as a result of the new Constitution, Canadian society as a whole is moving towards a greater reliance on the judicial system than on Parliament, and this may well reflect upon the way in which the nuclear power industry is regulated.

40. The extent to which the AECB and Ontario Hydro are mismatched in terms of size and resources also raises questions about the effectiveness of the AECB. Ahearne points out that "Ontario Hydro is essentially the nuclear utility in Canada . . . . Consequently, nuclear regulation in Canada is the regulation of Ontario Hydro."<sup>42</sup> It has been stressed throughout this section that the AECB places the onus on Ontario Hydro to prove to it that its proposals with respect to nuclear generating stations are safe. This proof largely takes the form of safety analyses, often conducted to AECB specification. The AECB, however, does not verify on a consistent basis the results of such licensee-directed research.<sup>43</sup>

41. The research programme in support of safety and licensing for the Canadian nuclear industry is run by AECL, funded jointly by it and the other three members of the CANDU Owners' Group (COG). It is difficult to identify and separate research that pertains specifically to licensing and safety, as the total amount committed to any specific investigation often includes a portion of the cost of the ongoing, fundamental research being carried out by AECL that

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<sup>42</sup>Ahearne, J.F., 1987. p. 8.

<sup>43</sup>Ahearne, J.F., 1987. p. 36.

supports a specific investigation. The annual COG nuclear research and development budgets for the 3 most recent yr are as follows: 1986-87--\$20.6 million; 1987-88--\$65.0 million; and 1988-89--\$92.0 million. AECL provides about 50% of the annual budget, Ontario Hydro provides about 44%, and Hydro Québec and New Brunswick Power together provide about 3%. One of the main reasons for the significant increases in these research and development budgets in 1987-88 and 1988-89 is that the utilities have endeavoured to cover part of the shortfall in the AECL budget that resulted from federal government cut-backs in funding. (The federal government has contributed \$200 million a year to all nuclear research and development, but this will be reduced to \$100 million by 1991.)

42. The portion of the research and development budget contributed by the COG utilities is directed towards research specifically related to keeping their nuclear stations operating and research necessary to meet AECB licensing concerns. AECL is free to use the portion of the research and development budget that it contributes for research into matters of interest to it or for research that it feels would be good for the utilities. The research agenda is established in a committee composed of AECL and utility representatives, and there is a considerable amount of give and take in establishing the programme. AECL and other members of COG periodically conduct meetings at the research laboratories in which they report upon their research to the AECB.

43. In addition to the research and development conducted by AECL, the CANDU utilities conduct small programmes under contract. This includes contracts with Westinghouse Canada and certain universities. Ontario Hydro also maintains an active nuclear research programme at its W.P. Dobson Laboratory in Toronto.

44. The AECB's own research programme covers only small projects and is run on a budget of \$3.5 million annually. This programme should not be undervalued, and some modest expansion may be justified. But one of the principal functions of a regulator is as auditor of licensee-directed research. By not conducting research on a large scale, the AECB is able to retain its

independence as a reviewer of licensee research results. The problem, however, is that the AECB's resources are not sufficient for it to review all of the research results with which it is presented.

45. The ability to verify independently the results of research presented to it and to conduct some research itself is essential to a nuclear power regulator, as shown by the US NRC, the British NII, and the French SCSIN. "The NRC has a large research programme, although the NRC itself has no research facilities. The programme is conducted primarily at US national laboratories (e.g., Brookhaven and Argonne) and in universities. In addition to exploring the boundaries of existing regulations, the programme provides the NRC with a capability to test claims of licensees."<sup>44</sup> The UK NII has similar access to independent research: "Within the inspectorate there is the technical capability to check reactor design in considerable detail. The NII do not have a research laboratory of their own, but they can and do commission such research as they need from outside bodies."<sup>45</sup> The French nuclear programme is noted for its strong laboratories (government-run), and within these is a group devoted to safety that provides independent assessment of the utility's safety analyses.

46. A fundamental difference between the way in which the US NRC and the AECB regulate, noted by Ahearne, is the fact that the AECB does not benchmark the computer codes used by Ontario Hydro and AECL.<sup>46</sup> These are highly sophisticated codes upon which the safe operation of the CANDU reactors is largely dependent. The AECB recognises that the reliability and accuracy of Ontario Hydro's computer codes are important areas of nuclear safety. Benchmarking, however, may not be the most satisfactory method of independently checking the claims of the utility, as the process restricts the checking to comparison with a single other code that may not be flawless. Another method of checking claims more widely used in Canada is validating, in

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<sup>44</sup>Ahearne, J.F., 1987. p. 38.

<sup>45</sup>Flowers, Sir Brian, 1976. p. 113.

<sup>46</sup>Ahearne, J.F., 1987. p. 39.

which computer codes are checked against results of experiments. The AECB currently reviews codes and does sensitivity analyses in related areas, but it does not benchmark or validate in any systematic fashion.<sup>47</sup> The AECB is restricted in this area by its current resources. If, for example, it were to validate in a consistent fashion, another 15 to 20 staff members and an additional half million dollars in the annual computer budget would be required.

47. Until 1975 the AECB had funded pure research in Canadian universities, mainly in nuclear physics and nuclear engineering. In 1975, however, the AECB's existing grant programme and related funds were turned over to the NRCC, and the AECB decided to concentrate on research directly related to its regulatory activities. As Doern points out, however, the appropriate role of the AECB vis-à-vis research was not so easily settled: "The AECB's research funding function, both past and present, has important regulatory implications. In one sense the old granting programme could be seen as a promotion of the development of atomic energy and competent research personnel. But the granting programme could also be viewed as a means of co-opting the one pool of nuclear expertise, namely university nuclear physicists, capable of criticising the state-owned industry centred on AECL."<sup>48</sup>

48. The aspects of the research issue relevant to this consideration are (i) whether the ongoing research and development programme in support of the Canadian nuclear industry is adequate to ensure safety, and whether the way in which it is handled can be perceived as providing the necessary assurance of safety; and (ii) whether the AECB has sufficient resources at its disposal to verify the safety claims made by its licensees.

49. Robertson points out that:

most modern technologies require large interdisciplinary programmes with expensive, efficient support services, combining underlying and applied science, engineering

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<sup>47</sup>Interview with AECB representative, 27 November 1987.

<sup>48</sup>Doern, G.B., 1976. p. 25.

development and design, safety, economic and market analysis, and the operation of test facilities as well as pilot and prototype plants. Universities, with other priorities, do not provide this broad capability, nor do most of the laboratories of Canadian industry, dominated as it is by multinational corporations. In these circumstances, Canada has evolved some excellent government laboratories.<sup>49</sup>

It would be hard to dispute the excellence of the AECL research facilities at Chalk River and Whiteshell, but they may not be perceived as independent because they are part of COG (i.e., AECL as a corporate entity, not the research company *per se*). It may be feared by some that the objectivity of the researchers within AECL may be clouded by corporate goals, or that analysts may be tempted to use research results selectively. This, however, is highly unlikely: the AECL researchers are independent of Ontario Hydro and the other two utility members of COG in that AECL matches the research funds provided by the utilities; the research staff of AECL are professionals, most of them guided by a sense of dedication and by professional ethics (a matter considered in detail later in this appendix), and not easily compromised; and the AECB is free to inspect the research facilities.

50. The AECL research laboratories are a valuable resource of the Canadian nuclear power industry. The expertise and the equipment for research and development in support of the CANDU system exist within these laboratories, and neither Canadian universities nor the AECB could replace AECL's work. It would not be desirable for AECB to attempt to do so: if it were to assume an expanded role in research and development, it is quite likely that industry would cut back on its efforts in this area. The approach dictated by the Canadian regulatory philosophy is to leave the onus on the licensee to conduct safety research on the understanding that it must be able to satisfy the regulator. It is also desirable for research to be conducted at arm's length from those having to pass judgement on it, in this case the AECB.

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<sup>49</sup>Robertson, J.A.L., 1987a. Fuel for thought. Proceedings of the Engineering Centennial Conference, Montreal, May. p. 5.

## CONCLUSION

Research is essential to the safe operation of Ontario's CANDU reactors.

Arrangements for safety-related research currently in force, namely that AECL is primarily responsible, are workable and effective. Sufficient funds are, however, required.

Ontario Hydro, along with the other two Canadian nuclear utilities, contributes significantly to licensing and safety research conducted by AECL and has assumed an increasing portion of these costs as the federal government has cut back its funding of AECL. This is appropriate, as is Ontario Hydro's desire to control the way in which its research funds are directed by AECL. It is, nonetheless, important that the resources and facilities of the AECL laboratories not be eroded as the result of decreasing federal government support.

## RECOMMENDATION

3. **THAT** the federal government maintain its financing of AECL's safety-related research for CANDU reactors, regardless of future export prospects.

51. The second aspect of the issue is whether the AECB has sufficient resources at its disposal to verify the safety claims made by its licensees. The AECB has the power to request analyses and information until it is satisfied with licensee claims. It is a reasonable assumption that a certain amount of independent checking is required of even the most honest and dedicated utility representatives. The AECB does not dispute this and, indeed, does a considerable amount of auditing and verification. But it acknowledges that it could do more of this if its resources permitted, and that it would prefer to do more. Although it is obviously not necessary that the regulator match the size and resources of the utility person by person and dollar by dollar, a proportional balance is required between the two. Ontario Hydro is now such a dominant force that no such balance exists.

52. It may be argued by some that there is little need to increase the resources of the AECB: with the completion of Darlington NGS within the

next 3 or 4 yr there will be no new plants under construction, and the demands upon the AECB will consequently be reduced. Those who put forward such arguments, however, must be reminded that the ageing of existing plants will increase the demand upon the AECB, and that when the stations have served useful lives the decommissioning process must be overseen by the AECB.

53. The perception of the AECB that has been discussed in the preceding sections of this appendix is that of the AECB itself, of the industry that it regulates, and of the organised anti-nuclear-power lobby groups. The discussion has not considered the perception of the AECB by the general Canadian public. The AECB is not very visible in Canada, and its work and activities are little known to those outside the immediate circle of the industry and its organised anti-nuclear-power lobbyists. Public opinion surveys in Ontario indicate that the public tends to turn to Ontario Hydro for information on nuclear safety, although the AECB makes such information available. In 1980, 3 yr before the proclamation of the Access to Information Act, the AECB introduced a policy on public access to licensing information. It maintains an office of public information and a public documents room in which licensing information is freely provided. The extent to which these facilities are underused is illustrated by the fact that seven people made use of the public documents room in the first seven months of 1987.

54. The AECB's low public profile is not surprising, as it does not promote itself the way the US NRC does. For example, AECB resident inspectors at nuclear generating stations do not participate in community information sessions,<sup>50</sup> and the AECB Annual Report is considerably less comprehensive than that of the US NRC.

55. Nor does the AECB have access to a resource similar to the French Conseil supérieur de la sûreté nucléaire (CSSN), which has an exceedingly broad mandate to address issues related to the use of nuclear power. This so-called council of wise persons is made up of 39 high-ranking representatives of trade

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<sup>50</sup>Ahcarne, J.F., 1987. p. 22.

unions and associations, industry and government, and members-at-large selected for their scientific, technical, economic, or social competence and is chaired by Nobel Laureate Louis Neel. The Council may address to the Minister of Industry recommendations concerning any matter that it deems related to nuclear safety. The CSSN may create working groups to examine such technical matters as spent fuel and nuclear waste management. The mandate and composition of the Council are currently being modified to enable it to deal with all matters concerning public information related to the safety of the nuclear installations that are the responsibility of the Department of Industry (nuclear installations under the jurisdiction of the Department of Defence are not covered). This enlarged mandate of the CSSN will include relations with the news media and responsibility for the dissemination of information with respect to accidents that may occur in civil reactors. Additional members of the Council will, therefore, be selected for their communication skills.

56. It has been suggested by Review intervenors that the AECB should assume a greater role in the dissemination of information about nuclear power to the public.<sup>51</sup> It is a well-accepted principle of nuclear power regulation in Canada that the licensee assumes full responsibility for the operation of its sites. The licensee is, accordingly, the appropriate spokesperson for issues relating to either the normal or abnormal operation of these sites. The suggestions of the Review intervenors are not intended to confuse or to complicate the existing situation. The information that the AECB might disseminate would be generic and non-site-specific and might cover such issues as the effects of exposure to low-level radiation or the socio-economic benefits of nuclear power.

57. The issue of public visibility is one of which the new AECB President, Dr. R.J.A. Levesque, is aware, and which he is attempting to address. He has, for example, already attempted to obtain media coverage of his visits to nuclear generating stations in Canada.

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<sup>51</sup>CELA, 1987.



## CONCLUSION

Many, if not most, members of the Canadian public are unaware that an independent watchdog of the nuclear power industry exists in the form of the AECB.

If the AECB were more visible, and if its effectiveness were better perceived, some of the public uncertainty and fear about the use of nuclear power might be allayed.

58. The antithesis of the perception of the AECB being too much under the influence of the federal government is the concern raised almost as frequently: that the AECB is ignored by government and that there is little accountability to government. Ahearne states that Parliament lacks interest in the AECB's affairs.<sup>52</sup> Adams and Jerrett point out the complete lack of personal communication between the Hon. Patricia Carney during her term as Minister of EMR and the President of the AECB and indicate that "inconsistent ministerial supervision can lead to policy problems."<sup>53</sup>

## CONCLUSION

A regulatory body should have a clearly defined set of policies, understood by all, in order to protect both the regulator and the regulated. Without such policy, it is difficult to hold the regulator accountable, and there is the danger of unnecessary regulation being imposed upon the industry regulated. The matter of direction and policy of the AECB, however, was a matter identified by the Task Force on Program Review (Nielsen Task Force) and is currently being studied by the AECB.

59. It would be inaccurate to assume that the AECB is left entirely on its own by government. The AEC Act, section 20(1), requires the AECB to report to Parliament annually through the designated minister, and the AECB appears before parliamentary committee annually to defend its estimates. A transcript from Hansard indicates that the meeting lasts for 2 h, with about 20 fairly

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<sup>52</sup>Ahearne, J.F., 1987. p. 20.

<sup>53</sup>Adams, T. and M. Jerrett, 1987. pp. 7, 8.

routine questions being put to the AECB representatives by the 10-member committee. The AECB, as are other government departments and agencies, is subjected to scrutiny by the Auditor General of Canada, the Comptroller General of Canada, and Treasury Board. In addition, the AECB is subjected to ad hoc review and assessment by various government bodies from time to time. The following summary of recent and current reviews of the AECB, as they relate to the concerns of the Review, is drawn from the Domaratzki and Molloy paper presented to the 1987 meeting of the Canadian Nuclear Association.<sup>54</sup> It indicates that many of the above concerns as to the size, composition, and modus operandi of the AECB are being studied by government.

60. In 1983, the AECB was part of a government-wide review of management practices by the Office of the Comptroller General, and, in 1985, in addition to its routine annual financial audit, the Office of the Auditor General subjected the AECB to a comprehensive audit. Although the former resulted in some recommendations for improvement, the outcome of both reviews was favourable.

61. As part of the 1984-85 Task Force on Program Review chaired by the Hon. Erik Nielsen, the AECB was reviewed by two different study teams. One team examined the efficiency and effectiveness of the AECB and found that its administration and financial management were sound. This study team also took into account the criticism of the Board's composition, of the Board for not maintaining an arms's-length relationship with industry, and of the Board for administering the Regulations inconsistently. "To deal with these criticisms, the study team recommended that the structure of the Board be reviewed and the number of members expanded to include more permanent members representing industry, labour and public interests."<sup>55</sup>

62. The second study team examined the accountability of regulatory agencies to the government and took the view that Parliament should specify more clearly the policy objectives of the AECB. This team also recommended that

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<sup>54</sup>Domaratzki, Z. and T.J. Molloy, 1987. pp. 32-37.

<sup>55</sup>Domaratzki, Z. and T.J. Molloy, 1987. pp. 33, 34.

the Board should be enlarged to include representatives of industry, labour, and public interest groups, that the AECB should be left with the discretion to decide when to hold public hearings, and that the government should consider requiring the AECB to hold a public hearing to review the manner in which it would exercise this discretion.

63. The questions of the size, composition, and operation of the Board and the possibility of holding public hearings are still being studied by the Board and its staff, after which recommendations will be made to the designated minister. This is proceeding slowly because the Board's study stopped during the period in 1987 in which the presidency was vacant. "In defining the scope of the study, the Board expressed the opinion that it should continue to be a technical body which would make technical decisions on health, safety and security matters. Social and economic impacts of nuclear activities would continue to be left to other agencies, e.g., provincial authorities and the Federal Environmental Assessment Review Office."<sup>56</sup> The AECB recognises that if its size is to be enlarged, the AEC Act will have to be amended by Parliament.

## CONCLUSION

Nuclear power is a technologically advanced field that must be regulated by people capable of understanding nuclear science and technology. The regulation of nuclear power within the Canadian context, however, is not restricted to purely scientific and technical decisions. In the application of the ALARA (as low as reasonably achievable) principle in establishing permissible radiation doses, there are social and economic considerations implicit in AECB decisions. It is desirable, therefore, that the expertise available within the Board be broad. In order to ensure this breadth of outlook and skills, the size of the Board should be enlarged.

Board members should not, however, be appointed to represent industry, labour, special interest groups, etc. Other bodies exist in which the special concerns of these groups can be aired and debated.

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<sup>56</sup>Domaratzki, Z. and T.J. Molloy, 1987. p. 34.

Given the responsibility borne by members of the Board, it is desirable that the process by which members are appointed be open.

## RECOMMENDATION

4. **THAT** the AECB, in studying the recommendations of the Nielsen Task Force, give serious consideration (i) to enlarging its membership, to increase its resources of skill and knowledge; and (ii) to developing a mechanism for the identification of useful Board members.

64. In March 1986, the Government of Canada announced a Regulatory Reform Strategy, including the Citizens' Code of Regulatory Fairness, and, in May 1986, the Regulatory Process Action Plan. The AECB already complies with the 15 statements of the Code. The following statements in the Regulatory Process Action Plan, however, have implications for the AECB: "the government will encourage and facilitate a full opportunity for consultation and participation by Canadians in the federal regulatory process,"<sup>57</sup> and "the government will enhance the predictability of the exercise of discretionary powers by federal regulatory authorities."<sup>58</sup> The AECB's current practices with respect to the first statement are to issue consultative documents, to give advance notice to the public of the dates on which licensing decisions are to be made, and to make available upon request technical information submitted in support of licence applications. The extent to which these practices meet the first of the above-cited objectives is debatable. Several of the briefs submitted to the Review called for a much greater degree of involvement. The second objective, of enhancing the predictability of the exercise of discretionary powers, is somewhat difficult to reconcile with the AECB's non-prescriptive approach to licensing nuclear reactors.

65. Yet another body whose recommendations may affect the modus operandi of the AECB is the Standing Joint Committee of the Senate and the House of Commons on Regulations and other Statutory Instruments. "In order to protect

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<sup>57</sup>Domaratzki, Z. and T.J. Molloy, 1987. p. 35.

<sup>58</sup>Domaratzki, Z. and T.J. Molloy, 1987. p. 35.

applicants and licensees, as well as the people of Canada, from arbitrary decisions by regulators this joint committee wishes to see clear criteria to define how discretionary powers will be used."<sup>59</sup> The AECB supports the objectives of this joint committee, but hopes that they can be achieved without requiring regulations to be too detailed or prescriptive. And, finally, the Regulatory Process Action Plan of the federal government may affect the future modus operandi of the AECB: it puts "greater emphasis on the analysis of the impact of any proposed regulations and on public consultation."<sup>60</sup>

66. Much is expected of the AECB in addressing concerns raised with respect to its modus operandi by formal watchdog and regulatory reform bodies, such as those mentioned above, and by public interest groups, such as those that presented briefs to the Review. The AECB is not oblivious to these calls for change. Formal recommendations of review bodies are taken seriously and in many cases acted upon. This applies equally to recommendations coming from provincial review bodies, as indicated by the consideration given by the AECB to the recommendations directed to it by the 1980 Ontario Legislature Select Committee on Ontario Hydro Affairs. In fact, some impetus for change comes from within the AECB itself, as shown by the proposed changes to the AEC Regulations,<sup>61</sup> and from its Advisory Committees, as shown by a recent report prepared by the advisory committee on Nuclear Safety in which risk is examined from a much broader perspective than that taken by the AECB in the past.<sup>62</sup> There is still, however, much to be done.

67. If the AECB is to respond adequately to these various calls for change, it will require a major increase in its resources of funds and staff. Although

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<sup>59</sup>Domaratzki, Z. and T.J. Molloy, 1987. p. 36.

<sup>60</sup>Domaratzki, Z. and T.J. Molloy, 1987. p. 37.

<sup>61</sup>AECB, 1986. Proposed general amendments to the Atomic Energy Control Regulations. Atomic Energy Control Board Consultative Document C-83, 28 April.

<sup>62</sup>ACNS, 1986. A report on the public perception of risk. Advisory Committee on Nuclear Safety (Atomic Energy Control Board) ACNS-8, July.

there is much to be said for the efficiency of a lean organisation without a large bureaucratic overhead, it is remarkable that the AECB achieves its current level of effectiveness with such a small staff. It has about 120 employees devoted to reactor regulation. As indicated in paragraphs 51 and 52, the AECB feels that it could do considerably more auditing and verifying of licensee research results. Work related to licensing is ongoing for the AECB, and resident site inspectors currently devote a fair amount of time to it. Much of the licensing work they now undertake could be better handled elsewhere if resources permitted. The AECB would like to undertake probabilistic risk assessment studies for existing plants, and it would like to monitor international research in the nuclear power field. These aspirations are currently hampered by lack of resources. The AECB's size and scarcity of resources are further cause for concern when they are compared with those of the organisations it regulates, notably those of Ontario Hydro.

68. Present indications are that rather than seeing its resources enlarged in the near future, the AECB can expect them to diminish: "downsizing of the AECB (along with the rest of the public service) is in effect. At the AECB our approved staff complement will shrink from 285 in 1986 to 263 in 1991, a reduction of 8% over the five years."<sup>63</sup> Regardless of the desirability of developing and enhancing the role of the AECB, it is not going to happen unless the resources exist. The AECB is currently considering at the request of the federal government the practicality of attempting to recover some of its costs by levying fees against those using its services.<sup>64</sup> Such an enterprise would not increase the resources at the disposal of the AECB, but would merely result in a reduction of the parliamentary appropriations for the AECB.

## CONCLUSION

The resources available to the AECB are not keeping pace with the demands of the changing nuclear power industry. The AECB is stretched too thin. It is not necessary that it match the size and resources of the

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<sup>63</sup>Domaratzki, Z. and T.J. Molloy, 1987. p. 37.

<sup>64</sup>Domaratzki, Z. and T.J. Molloy, 1987. p. 37.

major nuclear power utility that it regulates, namely Ontario Hydro. But it must have sufficient strength to stay abreast of Ontario Hydro's nuclear programme and regulatory requirements.

## RECOMMENDATION

5. **THAT** the Government of Canada authorise an increase in staff for AECB, to allow appointments in professional fields not currently represented, and that it ensure that the AECB has sufficient access to computer and other equipment.

### D. The Nuclear Liability Act

69. The Nuclear Liability Act, which came into force in 1976, "makes the operator and only the operator of a nuclear generating station liable for damages, limits the liability of the operator to a maximum of \$75,000,000, and provides a mechanism for federal financial responsibility in the event that the injury or damage exceeds \$75,000,000."<sup>65</sup> The AECB administers this Act by "designating nuclear installations and, with the approval of Treasury Board, prescribing the amount of basic insurance to be maintained by the operator."<sup>66</sup> The Act expressly binds the Crown in right of a province, although this is not an issue with respect to Ontario Hydro, as it is not considered to be a Crown agency.

70. Considerable controversy surrounds this Act. Energy Probe, the City of Toronto, and 11 individuals recently challenged its constitutional validity on the basis that (i) it contravenes the Canadian Charter of Rights and Freedoms by limiting the right of individuals to bring actions against persons who may otherwise be legally responsible for damage or injury; and (ii) it deals with a subject with which the provinces have the exclusive powers to legislate. The Ontario Supreme Court did not uphold this challenge, taking the view that court

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<sup>65</sup>Lang Michener, 1987a. pp. 16, 17.

<sup>66</sup>AECB, 1987. p. 10.

action was premature as Canada had not had a major nuclear accident, and that none of the applicants had been affected by a nuclear accident. The court took the further view that Energy Probe and the City of Toronto were corporations and could not sustain injury.

71. Valid though this debate over the constitutionality of the Act may be, the issue relevant to this discussion is the one raised by Energy Probe: that by restricting the liability in the event of an accident solely to the operator of a nuclear facility, the Act shields contractors and suppliers and may encourage careless performance.<sup>67</sup> In announcing that Energy Probe would appeal the decision of the Ontario Supreme Court, David Poch, lawyer for Energy Probe, stated that "the elimination of the Act would make the nuclear industry financially responsible and thereby increase its incentive to undertake more safety efforts and avoid unneeded expansion."<sup>68</sup>

72. The amount of compensation that could be made under the Act is virtually limitless, as it provides for the federal government to intervene if the cost of injury and damage in the event of an accident exceeds \$75 million. Critics of the Act argue, however, that if government intervention were required, it would be the citizens of Canada who would bear the financial burden of the accident, and that, given the rate of inflation since the Act was proclaimed, the \$75 million ceiling on the liability of the operators is too low.

73. It is worth noting, however, that were an accident to occur in a nuclear facility, the chief impact would be felt within the facility itself. Given the high costs of building such a facility and the loss of revenue that would be suffered if the facility were shut down by an accident, the operator of the facility would bear a serious financial loss. Lang Michener point out that unless the contracts between the operator and its contractors and suppliers provide otherwise, the latter would be liable to the operator for loss or damage

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<sup>67</sup>Lang Michener Lash Johnston, 1987b. Correspondence with the Ontario Nuclear Safety Review, 21 October.

<sup>68</sup>Claridge, T., 1987. Challenge to limited awards in nuclear accidents rejected, *The Globe and Mail*, 9 September. p. A13.



to a nuclear facility as the result of negligence, and for all damage arising from a non-nuclear incident as the result of negligence.<sup>69</sup> It is very much in the financial interests of all of the contractors, suppliers, and nuclear power plant operators to perform safely.

74. Because of the Nuclear Liability Act, the Canadian public has greater assurance of receiving, without undue delay, compensation for injury or loss suffered as a result of a nuclear power accident than does the American public. The Act endeavours to ensure that funds are available to meet the liability it imposes and removes from the public the burden of having to prove third-party negligence to obtain compensation. Lang Michener note, though, that the Act requires action to be taken within 10 yr of an accident. Given the length of time sometimes required for the effects of radiation exposure to become apparent, this stipulation may have the effect of denying compensation to some victims of a nuclear accident.<sup>70</sup>

## CONCLUSION

The Nuclear Liability Act is good in principle, but could be improved.

Despite the criticisms levied by public interest groups, there is no evidence that the Nuclear Liability Act has an adverse effect on safety by reducing the financial *incentive for Ontario Hydro, its contractors, and suppliers* to make Ontario's nuclear generating stations safe. In the event of a serious accident in a nuclear generating station, the owner-operator would suffer a large financial loss. It is also clear that unless contracts provide otherwise, the utility could seek compensation for loss or damage from suppliers and contractors. It is in the financial interests of the contractors and suppliers, as well as of the nuclear power plant owners, to keep the plant working safely.

The Nuclear Liability Act provides the Canadian public with a greater assurance of compensation, without undue delay, for injury and loss in the event of a nuclear

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<sup>69</sup>Lang Michener, 1987b. p. 3.

<sup>70</sup>Lang Michener, 1987a. pp. 66, 67.

power plant accident than would be the case otherwise. There are, however, a couple of troubling aspects of the Nuclear Liability Act. The \$75 million ceiling on the liability of the utility may not be realistic. Given the length of time required for the effects of radiation exposure to become apparent moreover, the requirement that legal action be taken within 10 yr of an accident is not reasonable.

## RECOMMENDATION

6. **THAT** the Nuclear Liability Act be amended (i) to increase the \$75 million ceiling on the extent of the liability to be borne by the utility to a more realistic level; and (ii) to eliminate the requirement that action be taken within 10 yr of an accident and establish a more realistic period within which action must be taken.

### E. **Other Legislation and Regulations Bearing Upon the Safety of Ontario's CANDU Reactors**

75. Ontario Hydro, as is any corporation doing business in the province, is subject to the requirements of a myriad of federal, provincial, and municipal laws and regulations. Given the size of the Ontario Hydro corporation and the diversity of its operations, the impact of these requirements upon it is great. Nearly all Ontario Hydro activities, from its financial management, audit, and employment practices to its food service operations in corporation-run cafeterias, are monitored and regulated in some way. Furthermore, safety in the conventional side of Ontario Hydro's nuclear power plants is regulated in a number of ways, mainly by provincial ministries. It is, however, the legislation and regulation bearing upon the safety of Ontario Hydro's nuclear operations with which this discussion is concerned and upon which it has and will continue to focus.

## F. Other Federal Legislation and Regulation

76. Although the AEC Act and the Nuclear Liability Act are the principal laws pertaining to the nuclear industry in Canada, one other federal act and one federal regulation affect or have the potential to affect the safe operation of the Ontario nuclear power industry: the Environmental Contaminants Act and the Environmental Assessment and Review Process (EARP) Guidelines Order.

77. The Environmental Contaminants Act "is essentially directed at chemical substances which may cause harm to the environment and does not specifically identify radiation or radioactivity as a contaminant."<sup>71</sup> Lang Michener do not see it as being of particular significance in regulating the safety of Ontario's nuclear plants: "The Environmental Contaminants Act has not been considered to be a very effective tool in the curbing of environmental pollution, which is why the new Environmental Protection Bill (C-74) has been put forward."<sup>72</sup> The latter bill, however, has been criticised for not being forceful enough and does not specifically cover radioactive material.

78. The EARP Guidelines Order, brought into effect in 1984 under the Government Organisations Act, is administered by the Federal Environmental Assessment Review Office (FEARO). As indicated earlier in this appendix, "the federal Crown corporations involved in the Canadian nuclear industry are not required to participate in EARP, but are merely 'invited' to participate."<sup>73</sup> The 1984 amendment of EARP, however, requires the regulator, in this case the AECB, to become the proponent of the environmental assessment should the federal Crown corporation decline the invitation to participate. The AECB has not yet had to require an environmental assessment under this revised provision of EARP, but should this ever happen, it would be in a most awkward position.

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<sup>71</sup>Lang Michener, 1987a. p. 53.

<sup>72</sup>Lang Michener, 1987a. p. 53.

<sup>73</sup>Lang Michener, 1987a. p. 25.

## G. The Application of Federal Legislation to Provincial Crown Corporations

79. Some ambiguity exists as to whether Ontario Hydro is a provincial Crown corporation. The status of Ontario Hydro in this respect is important because of the complexity of the application of federal legislation to provincial Crown corporations. As the Review's legal consultants, Lang Michener, explain, "provincial Crown corporations can be bound by federal legislation when they operate in areas of federal legislative jurisdiction provided that certain legislative requirements are met. The basic requirement is that the legislation must have been clearly intended to bind manifestations of the Crown." Lang Michener go on to identify a potential problem: "since the Atomic Energy Control Act (Canada) does not expressly state that it is binding on the Crown, a question arises as to whether it applies to Ontario Hydro's CANDU reactors. If it does not, [Ontario] Hydro could at any time refuse to comply with the Act and with the requirements of AECB."<sup>74</sup> Recent court decisions have indicated that provincial Crown corporations are not bound by federal legislation by "necessary implication." Some of the Review's briefers have recommended that the AEC Act (Canada) and all other federal and provincial statutes concerning the regulation of CANDU reactors be amended in order to expressly bind the Crown in right of Canada and the provinces.<sup>75</sup>

80. Lang Michener indicate that section 3 of the Crown Agency Act of Ontario "stipulates that the Act 'does not affect Ontario Hydro' [emphasis added]. The Power Corporation Act, which established Ontario Hydro, is silent on whether or not Ontario Hydro is a Crown Agent."<sup>76</sup> Lang Michener point out that it is difficult to conclude whether Ontario Hydro is or is not a Crown agency, but that according to common law principles it would likely be considered an agent of the Crown, and that it is unclear whether or not section 3 of the Crown Agency Act is sufficient to negate this.<sup>77</sup>

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<sup>74</sup>Lang Michener, 1987a. p. 37.

<sup>75</sup>Lang Michener, 1987a; CELA, 1987.

<sup>76</sup>Lang Michener, 1987b. p. 1.

<sup>77</sup>Lang Michener, 1987b. p. 1.

81. There is, however, a body of opinion that contradicts that of Lang Michener concerning Ontario Hydro's status as a Crown agency. The Ontario Attorney General, R. Roy McMurtry, is on record as stating:

It is apparent from these sections [4(1) and 58] and the general scheme of The Power Corporation Act that Ontario Hydro is an independent corporation managed by its Board of Directors. As a statutory corporation, its powers are those granted to it by its enabling statute. . . . Ontario Hydro is not a government ministry, and The Power Corporation Act does not permit the Minister [of Energy] to govern its actions.<sup>78</sup>

Blake, Cassels and Graydon offered a similar opinion.<sup>79</sup> Many statutes that refer to agents of the Crown refer specifically to Ontario Hydro in addition when the intent is to bind Ontario Hydro: The Environmental Assessment Act (s.3, Reg. s.3), The Land Speculation Tax Act (s.4 [j]), and The Canada-Ontario Anti-Inflation Agreement (Schedule A).<sup>80</sup> Furthermore, the non-government-department status of Ontario Hydro was confirmed by the Privy Council decision in St. Catherines v. The Hydro Electric Power Commission of Ontario (1930).<sup>81</sup> If the validity of the exemption of Ontario Hydro from the Crown Agency Act were to be legally challenged, it is likely that the decision would be based on the extent of governmental control over the corporation. Courts have tended to apply this test strictly, basing their assessment on the formal, documented control mechanisms that exist rather than the actual control exerted. It is felt that Ontario Hydro would not be deemed to be a Crown agent.

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<sup>78</sup>Attorney General R. Roy McMurtry, 1978. Letter to the Honourable James A.C. Auld, Minister of Energy dated 16 October.

<sup>79</sup>Blake, Cassels and Graydon, 1976. Letter from A.J. MacIntosh to W.E. Rancy, Esq., O.C., Secretary and General Counsel of Ontario Hydro dated 4 August.

<sup>80</sup>Ministry of Energy, 1977. Internal memorandum dated 28 April, regarding Application of Rent Review to Ontario Hydro.

<sup>81</sup>[1930] D.L.R. 409, see Crown Counsel, Ontario, Robin Scott, 5 October 1972.

## CONCLUSION

Despite Lang Michener's concern, there is sufficient opinion and supporting evidence upon which to conclude reasonably that Ontario Hydro is not a Crown agency. If it is not a Crown agency, the concerns raised by Lang Michener and several Review intervenors over the applicability of federal and provincial legislation, including the Atomic Energy Control Act, to Ontario Hydro's activities simply are not valid. Ontario Hydro, as a statutory corporation, is bound by all relevant provincial and federal legislation.

### H. Provincial Legislation and Regulations

82. "The complexity of regulating the Canadian nuclear industry stems primarily from two dichotomies in the system: (i) the fact that Canada is a federal state and therefore the jurisdiction to legislate over various matters may reside in either the provincial or the federal parliament, or sometimes both, and (ii) the fact that both the federal and provincial governments are involved in both the commercial and regulatory aspects of the nuclear industry in Canada."<sup>82</sup> The role played by Ontario in regulating the CANDU reactors within the province is significant. The main items of Ontario legislation relevant to the safe operation of the nuclear power stations are the Environmental Assessment Act, 1980, the Environmental Protection Act, 1980, the Ontario Water Resources Act, 1980, and the Emergency Plans Act, 1983.

83. There is some question of the applicability of this provincial environmental legislation to an enterprise under federal jurisdiction:

At present Ontario has the strictest environmental legislation in the country (including the federal jurisdiction) and Ontario Hydro is the most experienced utility in the Canadian nuclear power business. However, as previously discussed, the decisions in Pronto and Denison, which have granted the federal government formal jurisdiction over all aspects of atomic energy,

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<sup>82</sup>Lang Michener, 1987a. p. 5.

possibly preclude or limit the application of provincial environmental controls to nuclear power.<sup>83</sup>

Lang Michener then go on to cite several court decisions that indicate that "the provisions of Ontario's environmental statutes which apply to radioactive contaminants or pollutants may be valid."<sup>84</sup>

84. The most important of the three provincial environmental acts to this discussion is the Environmental Assessment Act:

The Environmental Assessment Act provides for the assessment of projects by or on behalf of the government of Ontario, public bodies, or municipalities (s.3). The Act requires Ministerial approval for certain undertakings to proceed, approval which is based upon acceptance of an environmental assessment performed by the proponent of the undertaking (s.14(1) and s.6(1)). The Act makes it possible for public hearings to be held, but it also allows for the exemption of major public undertakings from complying with the assessment review process contained in the Act.<sup>85</sup>

85. The environmental assessment process is seen by some as an important vehicle for involving the Ontario government in decisions concerning the safety of its CANDU reactors. CELA takes the view that it is necessary to justify the need for a nuclear plant, pointing out that "the time to deal with the basic questions of safety and risk and nuclear power is during the early stages of the planning process," and that "the Ontario environmental assessment process offers the appropriate regulatory regime for addressing *inter alia* the risks of nuclear power during the system planning process."<sup>86</sup> CELA applauds the comprehensive definition of the environment provided by the Act, and the explicit inclusion of economic and social concerns within its purview. The Act's

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<sup>83</sup>Lang Michener, 1987a. p. 48.

<sup>84</sup>Lang Michener, 1987a. p. 49.

<sup>85</sup>Lang Michener, 1987a. p. 49.

<sup>86</sup>CELA, 1987. pp. 29, 30.

requirement of public hearings addresses the concerns of many public interest groups about the existing lack of public involvement during the siting stage of a nuclear power plant. If the requirements of the Act were applied rigorously, it might be unnecessary to amend the Atomic Energy Control Regulations to provide for public hearings during the siting process. As indicated, however, the Act is not rigorously applied, and exemptions may be granted under the Act:

Where the Minister is of the opinion that it is in the public interest, having regard to the purpose of this Act and weighing the same against the injury, damage or interference that might be caused to any person or property by the application of this Act to any undertaking, the Minister, with the approval of the Lieutenant Governor in Council or of such Minister of the Crown as the Lieutenant Governor in Council may designate, may by order,

- (a) exempt the undertaking or the proponent of the undertaking from the application of this Act or the regulations or any matter or matters provided for in this Act or the regulations subject to such terms and conditions as the Minister may impose.<sup>87</sup>

Such an exemption was granted with respect to the Ontario Hydro Darlington NGS, and "to date the environmental assessment process has been used only, with respect to [Ontario] Hydro matters, for the purpose of transmission siting and certain routine activities."<sup>88</sup> Although planning for the Darlington station began before the final proclamation of the Environmental Assessment Act, Ontario Hydro did, in fact, prepare the documentation required by the Act. This documentation was circulated within the Pickering community, and comment invited. Very little public response was generated: there were fewer than a dozen written comments. Given this apparent lack of public concern, the urgency to proceed with the project, and the fact that the Environmental

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<sup>87</sup>Environmental Assessment Act, Revised Statutes of Ontario, 1980, Chapter 140, Section 29.

<sup>88</sup>CELA, 1987. p. 32.



Assessment Act had not been in force when planning began, the public hearings requirement was waived.

## CONCLUSION

The demand for public hearings on future nuclear projects in Ontario can best be met by application of the relevant procedures of the Environmental Assessment Act. Public concern about such projects is of such a nature that use of this Act is more appropriate than reliance on AECB's procedures, which do not include public hearings.

## RECOMMENDATION

7. **THAT** the provisions of the Environmental Assessment Act be applied rigorously to future nuclear projects, and that exemptions from the requirements of the Act be granted only in the most unusual and extreme cases.

86. The Environmental Protection Act (Ontario) prohibits the discharge of a contaminant (radiation being included in the definition of contaminant) into the environment and prohibits any discharge likely to damage the environment or harm individuals. Section 17 of this Act is particularly important in that it:

allows the Director appointed under the Act to make orders specifying the construction and/or provision of specified equipment, devices, etc., and the implementation of specified procedures at the project site, where he is of the opinion, on reasonable and probable grounds, that the undertaking is such that the discharge of a contaminant from it would impair the environment within the terms of s.1(1)(c) of the Act.<sup>89</sup>

Although Lang Michener then go on to observe that there is no such power in either the AEC Act or the Nuclear Liability Act, it must be noted that the AECB has the power to require the provision of equipment, devices, and the like as licence conditions. Where the powers of the Environmental Protection Director and the AECB inspectors overlap, those of the AECB take precedence.

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<sup>89</sup>Lang Michener, 1987a. pp. 50, 51.

87. The Ontario Water Resources Act is intended to prevent the contamination of Ontario's water bodies and "contains provisions similar in scope to those of the Environmental Protection Act,"<sup>90</sup> including provisions similar to those of section 17. There is a Canada Water Act, administered by Environment Canada, but Ontario Hydro is normally monitored by the provincial authorities. Should Environment Canada wish to be involved, however, it would likely do so through restrictions imposed by AECB licences.

88. Section 21 of the AEC Regulations provides for the full reporting of "any occurrence likely to cause the exposure of any person to radiation."<sup>91</sup> This section of the regulations goes on to stipulate that the manager of a nuclear facility shall minimise the exposure of persons to radiation and shall comply with the instructions of the inspector for the area appointed by the AECB. It is clear, however, that other than monitoring the situation, the AECB's responsibility for dealing with a nuclear emergency ends at the plant gate. In Ontario, the Emergency Plans Act, 1983, administered by the Ministry of the Solicitor General, forms the legal basis for off-site emergency planning with respect to a nuclear accident. The way in which this Act is implemented is discussed in detail elsewhere in the Review report.

## I. Shared Jurisdiction

89. The AECB's area of jurisdiction overlaps with several areas of provincial authority, the important ones for this discussion being the inspection and licensing of pressure-retaining components of nuclear generating stations and occupational health and safety. Although "the provincial authorities generally do not provide another level of regulatory control, but, rather, look after their

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<sup>90</sup>Lang Michener, 1987a. p. 51.

<sup>91</sup>Lang Michener, 1987a. p. 56.

own specific areas of expertise,"<sup>92</sup> they tend to negotiate with the AECB on how authority can be exercised jointly in areas of such closely overlapping jurisdiction as those cited above. Adams and Jerrett caution that "there are inherent accountability problems with the complex and largely informal joint regulatory process."<sup>93</sup> Although the AECB's arrangements with respect to pressure-retaining components appear to be working well, there is evidence of accountability difficulties with respect to occupational health and safety.

90. The AECB and the Pressure Vessels Safety Branch, Technical Services Division, of the Ontario MCCR share responsibility for inspecting and licensing the pressure-retaining components of a nuclear station. The division of responsibility between the two agencies is set out in a document dated March 1985 titled "General Liaison Procedures Between Staff of the Atomic Energy Control Board and Staff of the Pressure Vessels Safety Branch, Technical Services Division, Ministry of Consumer and Commercial Relations." This covers all phases of monitoring, from the procurement of QA programmes through the replacement, repair, and modification of pressure-retaining components in operating nuclear generating stations. These arrangements have worked to the satisfaction of both parties, and, as indicated with respect to the construction of a nuclear facility, the AECB would be required to increase its inspection activity if the MCCR did not act on its behalf.

91. The situation with respect to occupational health and safety in Ontario's nuclear power plants is less tidy. Under a 1980 letter of understanding, the Ontario Ministry of Labour was given jurisdiction over conventional safety, and the AECB retained jurisdiction over matters of radiological protection. This is not a situation with which the Ontario Hydro Employees' Union (CUPE Local 1000) is satisfied. They do not believe that the letter of understanding is a document that would be recognised by law, and they would prefer more formal arrangements. Furthermore, they would prefer to deal with a single regulator

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<sup>92</sup>Robertson, J.A.L., 1987b. Comments to the Ontario Nuclear Safety Review on the submission by Atomic Energy of Canada Limited to the ONSR, 17 August. p. 3.

<sup>93</sup>Adams, T. and M. Jerrett, 1987. p. 29.

in the area of health and safety and recommend that the Occupational Health and Safety Act of Ontario be used to cover all health and safety matters affecting them. Ontario Hydro shares the views of the union in this matter. The Ontario Ministry of Labour, however, is not willing to settle labour disputes involving radiation exposure.

92. As nuclear energy is a matter under federal jurisdiction, health and safety in Ontario Hydro's nuclear generating stations come under the control of Labour Canada. If Ontario Hydro, CUPE, and the Ontario Ministry of Labour cannot reach a mutually acceptable agreement, the provisions of the Canada Labour Code would apply according to law. The AECB is flexible about the way the matter is to be handled.

## CONCLUSION

Intervention is not required with respect to the monitoring of occupational health and safety at Ontario Hydro's nuclear generating stations. If Ontario Hydro, CUPE Local 1000, and the Ontario Ministry of Labour cannot agree on a satisfactory division of responsibility for settling labour disputes over radiological matters, the Canada Labour Code provides appropriate mechanisms.

## J. International Monitoring of Ontario's CANDU Reactors

93. Canada is a member of the International Atomic Energy Agency (IAEA) and is one of the signatories of the Nuclear Non-Proliferation Treaty (NPT). This involvement provides an indirect added measure of safety for Ontario's CANDU reactors.

94. As a member nation of the IAEA, Canada is in a position to request it to send an Operational Safety Review Team (OSART) to review the operations of nuclear power plants within the country. At the suggestion of the Ontario government, Canada made such a request late in 1986, and the OSART review of the Ontario Hydro Pickering NGS that took place in June 1987 made a valuable contribution to the Review's work.

95. As indicated in the AECB Annual Report, "staff members participate in activities of the International Atomic Energy Agency, the Nuclear Energy Agency of the Organisation for Economic Cooperation and Development, and other international organisations concerned with the peaceful uses of nuclear energy."<sup>94</sup> This international activity enhances the AECB's staff awareness of issues related to nuclear safety and keeps them abreast of international standards for nuclear safety. It is also encouraging to see the high regard with which Canadians are held within these international organisations. For example, the immediate past-President of the AECB, Mr. Jon Jennekens, was appointed early in 1987 to head the IAEA safeguards programme.

### **K. The Relationship Between the Government of Ontario and Ontario Hydro**

96. The public accountability of Ontario Hydro is relevant to this consideration because the issues of risk, benefit, and necessity of the use of nuclear power are fundamental safety issues. Schrecker states that "it is impossible to separate the question of risk from nuclear power production from the complex and controversial question of necessity."<sup>95</sup> Value judgement, however, is exercised by the decision makers not only with respect to the future need for electricity in the province, but also with respect to the degree of risk that is tolerable in the production of this electricity. Given the magnitude of such decisions, it is reasonable to expect that they be influenced by the public or its elected representatives.

97. Concern was expressed by some of the Review's intervenors over the adequacy of such public influence. CELA, for example, stated in its brief:

Pursuant to sections 4 and 5 of The Power Corporation Act, responsibility for long-range strategic corporate planning, the process that will determine Ontario Hydro's commitment to nuclear energy, is vested in the utility's

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<sup>94</sup>AECB, 1987. p. 10.

<sup>95</sup>Schrecker, T., 1987. p. 2.

Board of Directors. Currently, no regulatory control or oversight whatsoever is exercised with respect to this seminal stage of the planning process. Neither is Ontario Hydro's planning undertaken within the context of an overall electrical energy plan developed by the province, for there is none. Rather, the role of the Ministry of Energy is reactive, with government involvement occurring only after the review and assessment process is concluded.<sup>96</sup>

In an attempt to address such concerns, the Review examined the relationship between the Government of Ontario and Ontario Hydro.

98. The Hydro-Electric Power Commission of Ontario was created in 1906, and its name changed to Ontario Hydro under The Power Corporation Act of 1974. As discussed earlier, Ontario Hydro's status as a Crown agency is somewhat ambiguous, but there is a significant body of evidence that suggests that it is not a Crown agency. The purpose of Ontario Hydro, as set out in The Power Corporation Act, amended in 1983, is the generation, transmission, distribution, supply, sale, and use of power, the provision of energy conservation programmes, and the production, sale, supply, and delivery of heat energy (i.e., steam, hot water, or hot air).<sup>97</sup> The Power Corporation Act makes no specific reference to the use of nuclear power.

99. The Ontario Ministry of Energy was created in 1974 under the Ministry of Energy Act, and the Minister of Energy was made responsible for the administration of The Power Corporation Act. The operating relationship between the Ontario Hydro Board of Directors and the Minister of Energy is set out in detail in a Memorandum of Understanding dated 11 March 1985.<sup>98</sup> Essentially, the Minister of Energy is Ontario Hydro's formal link with government: the

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<sup>96</sup>CELA, 1987. p. 27.

<sup>97</sup>The Power Corporation Act, Revised Statutes of Ontario, 1980, Chapter 384, as amended by 1981, Chapter 16; 1981, Chapter 41 and 1983, Chapter 15, sections 56, 56a(1), and 56c.

<sup>98</sup>Memorandum of Understanding between the Minister of Energy for Ontario and the Board of Directors of Ontario Hydro, 11 March 1985. p. 7.

Ontario Hydro Board's policy decisions are communicated in writing to the minister, minutes and supporting documents of Board meetings are sent to the minister, significant information on Ontario Hydro matters is transmitted to the minister by the Board, and requests by Ontario Hydro for Orders in Council are made through the minister. The exception in the latter case has to do with Ontario Hydro financial matters. In such cases, Ontario Hydro deals with the Provincial Treasurer, keeping the Minister of Energy informed.

100. The Ministry of Energy and the Ministry of Treasury and Economics provide useful formal links between Ontario Hydro and the Government of Ontario, but the influence and control over Ontario Hydro by government go well beyond the jurisdiction of these two ministries. It is, in fact, the Provincial Cabinet with which authority over Ontario Hydro rests. This is not surprising, given the importance of Ontario Hydro to the economic and social status of the province.

101. The Ontario Hydro Chairman and Board of Directors (with the exception of the President, who is appointed by the Board) are appointed by the Lieutenant Governor in Council (the Provincial Cabinet). The Board of Directors is responsible for the direction of the business affairs of Ontario Hydro, but The Power Corporation Act distinguishes between those things that Ontario Hydro may do independently and those that Ontario Hydro may do only with the prior approval of the Lieutenant Governor in Council. The Act states:

- (2) In particular, but without limiting the generality of subsection (1), the Lieutenant Governor in Council, upon the recommendation of the Corporation, may authorize the Corporation to,
  - (a) acquire by purchase, lease or otherwise, land, waters, water privileges, water powers, buildings and works used for, or adapted or useful for, or capable of being used or made useful for generating, transforming, transmitting, distributing or selling power; enter upon, take possession of, expropriate, acquire and use any such land, waters, water privileges, water powers, buildings and works without the consent of the owner thereof, or of any person in any manner entitled to any right, title,

interest, claim or demand thereto or therein; and have and hold them however acquired or obtained, and develop, utilize, use, maintain, operate and improve them for any of the purposes of this Act;

- (c) generate and produce power at places in Ontario by the use of water, coal, steam or oil, or by any other means, and transform, transmit, make available for use, distribute, deliver, sell, supply and generally use for the purposes of the Corporation such power and connect the works constructed or installed for these purposes with any other power works and with any system.<sup>99</sup>

102. Through the appointment of the Board of Directors, and by requiring Ontario Hydro to obtain Orders in Council for the crucial activities of acquiring land and equipment and generating power, The Power Corporation Act gives the Cabinet a tremendous amount of control over Ontario Hydro.

103. To return to the point made by CELA, the legislative requirements described above do not indicate formal government involvement in Ontario Hydro's strategic planning. The reality of the situation, however, is that government, and particularly Cabinet, is constantly kept abreast of Ontario Hydro's planning process, and that there is a two-way flow of information between Ontario Hydro and Queen's Park. The Ontario Hydro Government Relations Section briefs officials of the Ministry of Energy, of other ministries, and of the Premier's Office on an ongoing basis and endeavours both to generate information and to respond to requests for information. But the flow of information also takes place at the most senior level: the Chairman and President of Ontario Hydro and members of Cabinet communicate directly and informally when the need arises. The Cabinet is kept fully informed of the review and decision-making processes, often extending over a period of years, that lead to an application for an Order in Council, and may intervene at any time.

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<sup>99</sup>The Power Corporation Act, section 23(2), (a) and (c). I was unable to obtain an explanation why The Power Corporation Act, revised as recently as 1981, makes reference to a variety of power sources, but does not mention the nuclear option.



**104.** Government decisions regarding matters of such significance as the future siting of an electricity generating station are exceedingly complex, and Cabinet must do more than merely react to proposals put forward by Ontario Hydro. Indeed, a very careful balancing of social, economic, environmental, health, and federal-provincial issues must be achieved. The decision to build the Pickering NGS is an example of one made to satisfy various requirements. Ontario Hydro had identified the need for an additional generating station, nuclear being one option, and a coal-fired plant being another. The federal government was interested at the time in establishing a major project that incorporated AECL technology, and the provincial government was grappling with environmental problems that would be exacerbated by the further burning of coal. The decision to proceed with a nuclear generating station addressed the needs of the utility and of the two levels of government, and three-way financing of the Pickering project was arranged.

**105.** The recent draft Demand-Supply Planning Strategy document is an example of how Ontario Hydro endeavours to attain public involvement in its long-range corporate planning. The document, released in December 1987, is intended to serve as the basis for discussion of the future energy requirements of the province, and the Ontario Hydro Board of Directors expects to modify it as required. It will remain to be seen whether this draft strategy document will be dealt with by the recently appointed Select Committee on Energy of the Ontario Legislature, but public input is expected before the strategy document is finalised. The release of this draft document, however, has not satisfied all those concerned with Ontario Hydro's strategic planning. Critics object to the fact that the establishment of future power need is divorced from a consideration of the source of future supply and the siting of future generating stations.

## **CONCLUSION**

Involvement of elected public representatives in the Ontario Hydro strategic planning process appears to be the crux of the Ontario Hydro accountability issue. Given the potential impact of decisions made as part of this process on the Ontario public, the issue is understandable. An examination of the legislative

framework within which Ontario Hydro operates and the day-to-day reality of the relationship of the Government of Ontario and Ontario Hydro, however, indicates that Ontario Hydro can be and is indeed held accountable to government. The Cabinet is kept informed of all Ontario Hydro Board decisions and may intervene informally in its strategic corporate planning at any time. Furthermore, Cabinet retains the formal power to approve any action emanating from the strategic planning process.

#### **L. The Contribution of Individuals to the Regulatory Organisation**

106. As all first-year students of organisational theory learn, organisations are made up of people. This discussion, which began with an account of the macro nuclear power regulatory organisation in Ontario, the AECB, and attempted to cover the supporting and supplementing legal, regulatory, and organisational structures, would not be complete without considering the micro element: the people within the industry. The safety system surrounding Ontario's CANDU reactors has, in addition to the formal checks and balances described above, the conscience of each individual member of the system (albeit more highly developed in some than in others) to sustain it.

107. In its brief to the Review, the Federation of Engineering and Scientific Associations (FESA) estimates that 5000 professionals (engineers and scientists) are engaged in the nuclear power industry in Ontario.<sup>100</sup> Many of these individuals are directed by a professional code of ethics, the engineers among them subscribing to the Association of Professional Engineers of Ontario (APEO) Code of Ethics. Section 2(a) of this code states that the engineer's duty to public welfare is paramount.<sup>101</sup> FESA states that its members obviously try to work within the structure of their employer organisations to remedy safety problems that they may identify. But, "in extreme situations, where more than

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<sup>100</sup>FESA, 1987. Professional aspects of nuclear safety. A submission to the Ontario Nuclear Safety Review by the Federation of Engineering and Scientific Associations. Toronto, 1 September.

<sup>101</sup>FESA, 1987. Appendix B, p. i.

a question of differing professional judgement is involved, and it is considered that questions of improper or unethical professional conduct are involved, the individual then has the obligation to raise the issue, under clause 5(a) [of the APEO Code of Ethics] with the appropriate public authorities or bodies."<sup>102</sup>

108. The AECB's attitude towards whistle blowers is very clearly stated in the proposed amendments to the Regulations.<sup>103</sup> It is indicated there that any employee of a nuclear facility who has reason to believe that safety is being compromised should report the situation to the AECB, and that no licensee can discipline an employee for taking such action in good faith. The amendments to the Regulations do not say anything about whistle blowers within the AECB, but it is the AECB's attitude that they must apply the same rules to their own people as they do to those in the employ of licensees. Whistle blowers have, on the odd occasion, surfaced. There might be more of them were it not for the fact that the AECB's on-site inspectors discover in their day-to-day interaction with nuclear plant staff many of the problems that whistle blowers might identify.

109. The integrity of the nuclear power station work-force is further emphasised by the AECB's belief that conditions within the stations remain unchanged when their inspectors go off duty, and that the workers would advise them otherwise. The AECB is not averse from conducting unannounced inspections, as strongly recommended by one of the Review briefers,<sup>104</sup> but it is not convinced that much new knowledge would be gained as a result. (It is amusing to note that a certain amount of machination would be required of the AECB if it were to conduct a "midnight raid," as its own Physical Security Regulations endeavour to make it impossible for anyone to enter a nuclear power plant without reporting to a security guard who is an employee of the utility.)

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<sup>102</sup>FESA, 1987. p. 7.

<sup>103</sup>AECB, 1986.

<sup>104</sup>CELA, 1987. p. 6.

## M. Summary

110. The regulatory and organisational framework within which Ontario CANDU reactors operate appears to work very much better than its critics fear. Behaviour is governed to a large extent by a set of unwritten conventions, and the system is run by people with conviction and dedication. These attitudes are, no doubt, derived from the British political and social traditions. The system, however, requires strengthening: Canadian regulatory institutions are evolving, and people are requiring assurance that trust in these institutions is merited and that there are enforceable means of ensuring that their trust is honoured.

111. Indeed, there appears to be a need to emphasise that a regulatory system exists in Canada: "The general public has little knowledge of the AECB's identity and function. It is unaware, in general, of the extent of the measures taken to control the risks associated with the production of nuclear power."<sup>105</sup> This lack of public awareness at best leads to insecurity and suspicion, and at worst to panic. The situation in Greece during the month following the Chernobyl accident, May 1986, is a prime example of the effect of ignorance and misleading information: 2500 otherwise wanted pregnancies were artificially terminated because of conflicting data and false rumours about the radiation doses.<sup>106</sup> An effort must be exerted to make known to the Canadian public the programmes in effect to ensure the safe operation of the nuclear power industry, to make available to the public the facts about the risk of nuclear power, and to provide the means of some formal public participation in decision making about nuclear power. Such action could, of course, benefit only those members of the public who take the trouble to inform themselves about the issues and who attempt to influence the way decisions are made. This may well call for an expanded role for the AECB, but EARP and the

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<sup>105</sup>ACNS, 1986. p. vi.

<sup>106</sup>Anonymous, 1987a. Victims of Chernobyl in Greece: induced abortions after the accident. Br. Med. J. 295:1100, 31 October.

provincial legislation also provide opportunities to address the social, economic, and public information aspects of nuclear power.

112. Finally, the question of safety begins at the time a nuclear generating station is contemplated, for it is at that time that the issues of benefit, risk, and the level of tolerable risk must be confronted. It is at this stage, too, that regulation begins, because the public or its elected representatives must decide whether it is in the best public interest to incur the risks involved with the use of nuclear power. It is essential, therefore, that all such decisions be made in a democratic forum, i.e., one in which the public is directly involved, or one in which elected representatives of the public speak and act on its behalf.

## Annex VII-1

**Visits and Interviews  
Concerning Nuclear Power Regulation**

1987

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|-------------|--|
| 12 February | The Atomic Energy Control Board, President and officials, Ottawa                                   |
| 19 March    | The US Nuclear Regulatory Commission, Chairman and officials, Washington, DC                       |
| 27 April    | The Atomic Energy Control Board, officials, Ottawa   |
| 3 August    | Sir Frank Layfield, Q.C., London   |
| 4 August    | The Central Electricity Generating Board, Chairman, London   |
| 4 August    | Lord Flowers, London   |
| 5 August    | The British Nuclear Installations Inspectorate, H.M. Chief Nuclear Inspector and officials, London |
| 12 August   | Dr. Christopher Herzig, International Atomic Energy Agency, Vienna                                 |
| 18 August   | The Atomic Energy Control Board, officials, Ottawa   |
| 3 September | The Assistant Auditor General of Canada and officials, Ottawa                                      |
| 13 November | Mr. J.A.L. Robertson, Deep River   |
| 27 November | The Atomic Energy Control Board, official, telephone interview                                     |
| 3 December  | The Atomic Energy Control Board, President and official, Ottawa                                    |
| 9 December  | The British Nuclear Installations Inspectorate, official, telephone interview                      |

1988

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| 27 January | Mr. Barry Collingwood, Manager of CANDU Owners' Group |
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## Annex VII-3

### Conclusions and Recommendations

#### CONCLUSION

There is no evidence that the AECB lacks the proper means of enforcing its regulations. The licence and the threat of unfavourable publicity if the utility does not comply with licence conditions are adequate tools of enforcement.

#### CONCLUSION

Capable technical expertise in Canada is scarce. The scientists and engineers engaged in the nuclear industry are professionals, guided by their professional ethics. The ONSR has discovered no evidence of an incestuous relationship between the industry and its regulator, nor of staff movement between the AECB and the utilities to the extent that might be described as movement through a "revolving door." The present situation is a healthy one and should be preserved in the future.

#### CONCLUSION

Public confidence in the AECB as an independent watchdog of the nuclear industry would be increased if the Atomic Energy Control Board and Atomic Energy of Canada Limited were to report to Parliament through different ministers of the Crown.

#### RECOMMENDATION

1. **THAT** the Atomic Energy Control Board and Atomic Energy of Canada Limited report to Parliament through different ministers of the Crown.

## CONCLUSION

Strengthening of the regulatory system need not do away with the existing atmosphere of mutual respect, nor need it require the adoption of a prescriptive approach to regulation. It does, however, call for more extensive documentation by the AECB of decisions and licensing requirements--decisions and requirements often negotiated in detailed face-to-face discussion with the utility--and the reasons behind these decisions and requirements. Such strengthening would make it easier for the regulator and the utility to settle disputes and more difficult for the utility to resist the requirements of the regulator.

## RECOMMENDATION

2. **THAT** the AECB adopt a more formal approach in its dealings with utilities, and that its licensing and other requirements and the reasons behind them be clearly documented.

## CONCLUSION

Research is essential to the safe operation of Ontario's CANDU reactors.

Arrangements for safety-related research currently in force, namely that AECL is primarily responsible, are workable and effective. Sufficient funds are, however, required.

Ontario Hydro, along with the other two Canadian nuclear utilities, contributes significantly to licensing and safety research conducted by AECL and has assumed an increasing portion of these costs as the federal government has cut back its funding of AECL. This is appropriate, as is Ontario Hydro's desire to control the way in which its research funds are directed by AECL. It is, nonetheless, important that the resources and facilities of the AECL laboratories not be eroded as the result of decreasing federal government support.

## RECOMMENDATION

3. **THAT** the federal government maintain its financing of AECL's safety-related research for CANDU reactors, regardless of future export prospects.

## CONCLUSION

Many, if not most, members of the Canadian public are unaware that an independent watchdog of the nuclear power industry exists in the form of the AECB.

If the AECB were more visible, and if its effectiveness were better perceived, some of the public uncertainty and fear about the use of nuclear power might be allayed.

## CONCLUSION

A regulatory body should have a clearly defined set of policies, understood by all, in order to protect both the regulator and the regulated. Without such policy, it is difficult to hold the regulator accountable, and there is the danger of unnecessary regulation being imposed upon the industry regulated. The matter of direction and policy of the AECB, however, was a matter identified by the Task Force on Program Review (Nielsen Task Force) and is currently being studied by the AECB.

## CONCLUSION

Nuclear power is a technologically advanced field that must be regulated by people capable of understanding nuclear science and technology. The regulation of nuclear power within the Canadian context, however, is not restricted to

purely scientific and technical decisions. In the application of the ALARA (as low as reasonably achievable) principle in establishing permissible radiation doses, there are social and economic considerations implicit in AECB decisions. It is desirable, therefore, that the expertise available within the Board be broad. In order to ensure this breadth of outlook and skills, the size of the Board should be enlarged.

Board members should not, however, be appointed to represent industry, labour, special interest groups, etc. Other bodies exist in which the special concerns of these groups can be aired and debated.

Given the responsibility borne by members of the Board, it is desirable that the process by which members are appointed be open.

## **RECOMMENDATION**

4. **THAT** the AECB, in studying the recommendations of the Nielsen Task Force, give serious consideration (i) to enlarging its membership, to increase its resources of skill and knowledge; and (ii) to developing a mechanism for the identification of useful Board members.

## **CONCLUSION**

The resources available to the AECB are not keeping pace with the demands of the changing nuclear power industry. The AECB is stretched too thin. It is not necessary that it match the size and resources of the major nuclear power utility that it regulates, namely Ontario Hydro. But it must have sufficient strength to stay abreast of Ontario Hydro's nuclear programme and regulatory requirements.



## RECOMMENDATION

5. **THAT** the Government of Canada authorise an increase in staff for AECB, to allow appointments in professional fields not currently represented, and that it ensure that the AECB has sufficient access to computer and other equipment.

## CONCLUSION

The Nuclear Liability Act is good in principle, but could be improved.

Despite the criticisms levied by public interest groups, there is no evidence that the Nuclear Liability Act has an adverse effect on safety by reducing the financial incentive for Ontario Hydro, its contractors, and suppliers to make Ontario's nuclear generating stations safe. In the event of a serious accident in a nuclear generating station, the owner-operator would suffer a large financial loss. It is also clear that unless contracts provide otherwise, the utility could seek compensation for loss or damage from suppliers and contractors. It is in the financial interests of the contractors and suppliers, as well as of the nuclear power plant owners, to keep the plant working safely.

The Nuclear Liability Act provides the Canadian public with a greater assurance of compensation, without undue delay, for injury and loss in the event of a nuclear power plant accident than would be the case otherwise. There are, however, a couple of troubling aspects of the Nuclear Liability Act. The \$75 million ceiling on the liability of the utility may not be realistic. Given the length of time required for the effects of radiation exposure to become apparent, moreover, the requirement that legal action be taken within 10 yr of an accident is not reasonable.

## RECOMMENDATION

6. **THAT** the Nuclear Liability Act be amended (i) to increase the \$75 million ceiling on the extent of the liability to be borne by the utility to a more realistic level; and (ii) to eliminate the requirement that action be taken within 10 yr of an accident and establish a more realistic period within which action must be taken.

## CONCLUSION

Despite Lang Michener's concern, there is sufficient opinion and supporting evidence upon which to conclude reasonably that Ontario Hydro is not a Crown agency. If it is not a Crown agency, the concerns raised by Lang Michener and several Review intervenors over the applicability of federal and provincial legislation, including the Atomic Energy Control Act, to Ontario Hydro's activities simply are not valid. Ontario Hydro, as a statutory corporation, is bound by all relevant provincial and federal legislation.

## CONCLUSION

The demand for public hearings on future nuclear projects in Ontario can best be met by application of the relevant procedures of the Environmental Assessment Act. Public concern about such projects is of such a nature that use of this Act is more appropriate than reliance on AECB's procedures, which do not include public hearings.

## RECOMMENDATION

7. **THAT** the provisions of the Environmental Assessment Act be applied rigorously to future nuclear projects, and that exemptions from the

requirements of the Act be granted only in the most unusual and extreme cases.

## **CONCLUSION**

Intervention is not required with respect to the monitoring of occupational health and safety at Ontario Hydro's nuclear generating stations. If Ontario Hydro, CUPE Local 1000, and the Ontario Ministry of Labour cannot agree on a satisfactory division of responsibility for settling labour disputes over radiological matters, the Canada Labour Code provides appropriate mechanisms.

## **CONCLUSION**

Involvement of elected public representatives in the Ontario Hydro strategic planning process appears to be the crux of the Ontario Hydro accountability issue. Given the potential impact of decisions made as part of this process on the Ontario public, the issue is understandable. An examination of the legislative framework within which Ontario Hydro operates and the day-to-day reality of the relationship of the Government of Ontario and Ontario Hydro, however, indicates that Ontario Hydro can be and is indeed held accountable to government. The Cabinet is kept informed of all Ontario Hydro Board decisions and may intervene informally in its strategic corporate planning at any time. Furthermore, Cabinet retains the formal power to approve any action emanating from the strategic planning process.

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Council Review Panel on Strategic Energy Grants. He is a Fellow of the Royal Society of Canada.

## List of Acronyms and Abbreviations

ACNS	Advisory Committee on Nuclear Safety (Atomic Energy Control Board)
ACRP	Advisory Committee on Radiological Protection (Atomic Energy Control Board)
AEC Act	Atomic Energy Control Act
AECB	Atomic Energy Control Board (Canada)
AECL	Atomic Energy of Canada Limited
AES	Atmospheric Environment Service
AIM	Abnormal Incident Manual
ALARA	as low as reasonably achievable
ALI	Annual Limit on Intake
APEO	Association of Professional Engineers of Ontario
ARW	atomic radiation worker
ASME	American Society of Mechanical Engineers
BEIR	(Advisory Committee on the) Biological Effects of Ionizing Radiations
Bq	becquerel
BRMD	Bureau of Radiation and Medical Devices
BWR	boiling water reactor
CANDU	<u>C</u> anada <u>D</u> euterium <u>U</u> ranium reactor
CCAP	la Commission centrale des appareils a pression
CCIW	Canada Centre for Inland Waters
CCPA	Canadian Chemical Producers Association
CEA	Commissariat a l'Energie Atomique
CEGB	Central Electricity Generating Board
CELA	Canadian Environmental Law Association
CGE	la Compagnie générale d'électricité
Ci	curie
COG	CANDU Owners' Group
CRNL	Chalk River Nuclear Laboratories
CSSN	le Conseil supérieur de la sûreté nucléaire
CUPE	Canadian Union of Public Employees
d	day
DAI	derived air concentration
DEL	derived emission limit



**List of Aconyms and Abbreviations (cont'd)**

<b>DHC</b>	<b>delayed hydride cracking</b>
<b>DND</b>	<b>Department of National Defence</b>
<b>DPSE</b>	<b>Darlington Probabilistic Safety Evaluation</b>
<b>DRIR</b>	<b>directions regionales et de la recherche</b>
<b>DRL</b>	<b>derived release limit</b>
<b>EARP</b>	<b>Environmental Assessment and Review Process</b>
<b>ECC</b>	<b>emergency core cooling</b>
<b>ECCS</b>	<b>emergency core cooling system</b>
<b>ECIS</b>	<b>emergency coolant injection system</b>
<b>ECY</b>	<b>equivalent calendar year</b>
<b>EDF</b>	<b>Électricité de France</b>
<b>EMO</b>	<b>Emergency Measures Organisation</b>
<b>EMR</b>	<b>Energy, Mines and Resources Canada</b>
<b>EPC</b>	<b>Emergency Preparedness Canada</b>
<b>EPO</b>	<b>Emergency Planning Ontario</b>
<b>EPRC</b>	<b>ex-plant release category</b>
<b>FADS</b>	<b>filtered air discharge system</b>
<b>FDC</b>	<b>fuel damage category</b>
<b>FEARO</b>	<b>Federal Environmental Assessment Review Office</b>
<b>FESA</b>	<b>Federation of Engineering and Scientific Associations</b>
<b>FNERP</b>	<b>Federal Nuclear Emergency Response Plan</b>
<b>GPR</b>	<b>le Groupe permanent réacteurs</b>
<b>GWh</b>	<b>gigawatt-hour</b>
<b>GWyr</b>	<b>gigawatt-year</b>
<b>H</b>	<b>procédures "hors-dimensionnement"</b>
<b>h</b>	<b>hour</b>
<b>HTS</b>	<b>heat transport system</b>
<b>HWR</b>	<b>heavy-water reactor</b>
<b>IAEA</b>	<b>International Atomic Energy Agency</b>
<b>ICRP</b>	<b>International Commission on Radiological Protection</b>
<b>IIASA</b>	<b>International Institute of Applied Systems Analysis</b>
<b>IICPH</b>	<b>International Institute of Concern for Public Health</b>

## List of Acronyms and Abbreviations (cont'd)

IJC	International Joint Commission
INPO	Institute of Nuclear Power Operations
IPSN	l'Institut de protection et de sûreté nucléaire
J	joule
KEMA	Keuring van Elektrotechnische Materialen Arnhem
L	litre
LOCA	loss of coolant accident
LOECC	loss of emergency core cooling system
LWR	light-water reactor
m	metre
MACCS	Melcor Accident Consequence Code System
MAGNOX	magnesium oxide reactor
MCCR	Ministry of Consumer and Commercial Relations
MeV	mega electron volts
min	minute
MP	melting point
MWe	megawatt-electrical
MWt	megawatt-thermal
NAS	National Academy of Sciences (USA)
NCRP	National Council on Radiological Protection and Measurements
NEA	Nuclear Energy Agency
NEP	Nuclear Emergency Plan (Ontario)
NGD	Nuclear Generation Division (Ontario Hydro)
NGS	nuclear generating station
NHW	Department of National Health and Welfare
NII	Nuclear Installations Inspectorate (UK)
NIRC	Nuclear Integrity Review Committee
NOP	neutron overpower
NPD	Nuclear Power Demonstration reactor (Rolphton) Nuclear Power Development (Bruce)
NPT	Nuclear Non-Proliferation Treaty
NRCC	National Research Council of Canada
NRPB	National Radiological Protection Board (UK)

## List of Acronyms and Abbreviations (cont'd)

NRU	National Research Universal reactor (Chalk River Nuclear Laboratories, AECL)
NRX	National Research Experimental reactor
NSSD	Nuclear Studies and Safety Department
NWRI	National Water Research Institute
OECD	Organisation for Economic Co-operation and Development
OHEOC	Ontario Hydro Emergency Operations Centre
OME	Ontario Ministry of the Environment
OMS	Organisme modial de la santé
ONSR	Ontario Nuclear Safety Review
ORSEC	l'organisation de <u>secours</u>
ORSECRAD	<u>O</u> rganisation de la <u>sec</u> urité en cas de <u>rad</u> iations
OSART	Operational Safety Review Team (IAEA)
OTA	Office of Technology Assessment (USA)
Pa	pascal
Pc	poste de commande
PPI	le plan particulier d'intervention
PRA	Probabilistic Risk Assessment
PSE	Probabilistic Safety Evaluation
PTHW	pressure tube heavy-water reactor
PWR	pressurised water reactor
QA	quality assurance
QC	quality control
QE	quality engineering
RFS	Règles fondamentales de sûreté
RIH	reactor inlet header
RMEP	Radioactivity Management and Environmental Protection Department
s	second
SCOHA	Select Committee on Ontario Hydro Affairs of the Ontario Legislature
SCSIN	Service central de la sûreté des installations nucléaires
SDS	shut-down system

## List of Acronyms and Abbreviations (cont'd)

SER	Significant Event Report
SGHWR	steam-generating heavy water reactor
SI	Système International
SMR	standardised mortality ratio
SRDF	le système de recueil de données de fiabilité
Sv	sievert
TLD	thermoluminescent dosimetry dose
TMI	Three Mile Island Nuclear Generating Station of the General Public Utilities System
TWh	terrawatt-hour
U	procedures "ultimes"
UECC	Unit Emergency Control Centre
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
US NRC	US Nuclear Regulatory Commission
WNRE	Whiteshell Nuclear Research Establishment
yr	year

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