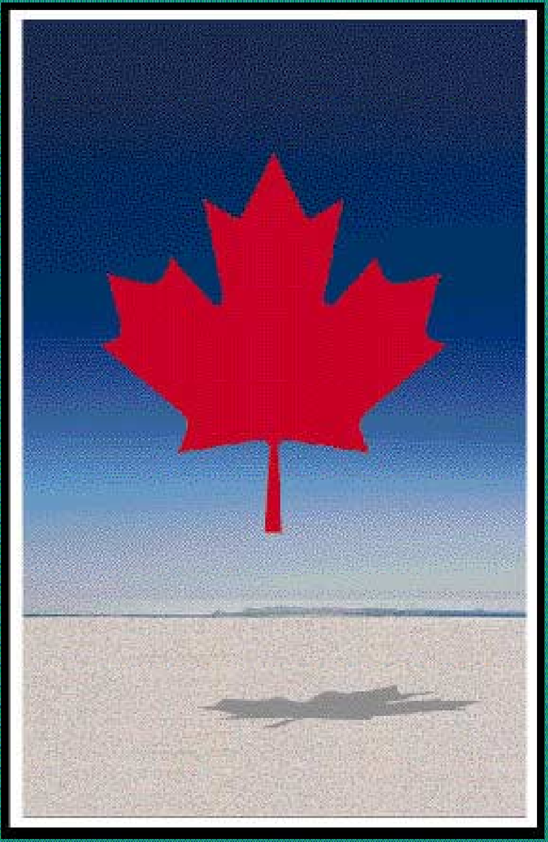
Government of Canada Gouvernement du Canada



Canadian National Report for the Convention on Nuclear Safety

Canada

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© Minister of Public Works and Government Services Canada 1998
Catalogue number CC2-0690E
ISBN 0-662-27207-2

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CANADIAN NATIONAL REPORT
FOR
THE CONVENTION
ON NUCLEAR SAFETY

September 1998

PREFACE

The Canadian National Report for the Convention on Nuclear Safety is prepared in fulfilment of Canada's obligation as a signatory of the Convention on Nuclear Safety coordinated by the International Atomic Energy Agency (IAEA). The report is to demonstrate how Canada has implemented its obligations under the Convention.

The report follows closely the guidelines, regarding form and structure, that were established by the contracting parties under Article 22 of the Convention. In addition, two sections are provided in the introduction to the report to address the basic characteristics of the CANDU reactor, and the Canadian philosophy and approach to safety of nuclear power stations. These sections reflect the historical development of the nuclear industry in Canada, and the uniqueness of the CANDU design.

This report is produced by the Atomic Energy Control Board on behalf of Canada. Contributions to the report were made by Atomic Energy of Canada Limited, Ontario Hydro Nuclear, New Brunswick Power, Hydro-Québec, CANDU Owners Group, Health Canada, Natural Resources Canada, and the Department of Foreign Affairs and International Trade.

TABLE OF CONTENTS

ANNEXES

LIST OF ATTACHMENTS

(provided under a separate cover)

7.1 Atomic Energy Control Act (R.S.C., 1985, c. A-16; and C-125, 1993)

7.2 Atomic Energy Control Regulations (Office Consolidation, with amendments to August 27, 1992)

7.3 Physical Security Regulations (SOR/83-77, 1983, and amendments to October 10, 1991)

7.4 Transport Packaging of Radioactive Materials Regulations (Office Consolidation, with amendments to February 27, 1992)

7.5 AECB Cost Recovery Fees Regulations, 1996 (SOR/96-412)

7.6 Nuclear Liability Act (R.S.C, 1985, c. N-28)

7.7 Nuclear Safety and Control Act (S.C., 1997, c. 9)

7.8 Canadian Environmental Assessment Act (S.C., 1992, c. 37)

7.9 Regulatory Guidance Documents:

- R-7: Requirements for Containment Systems for CANDU Nuclear Power Plants (1991)
- R-8: Requirements for Shutdown Systems for CANDU Nuclear Power Plants (1991)
- R-9: Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants (1991)
- R-10: The Use of Two Shutdown Systems in Reactors (1977)
- R-77: Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems (1987)
- R-90: Policy on the Decommissioning of Nuclear Facilities (1988)
- R-99: Reporting Requirements for Operating Nuclear Power Facilities (1995)

7.10 Consultative Document C-6, Requirements for the Safety Analysis of CANDU Nuclear Power Plants (1980)

15.1 Regulatory Standard S-106, Technical and Quality Assurance Standards for Dosimetry Services in Canada (1998)

LIST OF ACRONYMS

ACNS	Advisory Committee on Nuclear Safety
ACRP	Advisory Committee on Radiological Protection
AEC	Atomic Energy Control
AECB	Atomic Energy Control Board
AECL	Atomic Energy of Canada Limited
AIM	Abnormal Incident Manual
ALARA	As Low As Reasonably Achievable
BCO	Basis for Continued Operation
BMD	Board Member Document
CAMM	Canadian Adaptive Machine Model
CANATEX	Canadian National Exercises
CANDU	Canadian Deuterium Uranium
CANNET	CANDU Network
CANUTEC	Canadian Transport Emergency Centre
CCP	Critical Channel Power
CEA	Canadian Environmental Assessment
CEAA	Canadian Environmental Assessment Act
CEAAR	Canadian Environmental Assessment Act and Regulations
CHF	Critical Heat Flux
CNA	Canadian Nuclear Association
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
CPR	Critical Power Ratio
CSA	Canadian Standards Association
DLR	Derived Release Limits
DND	Department of National Defence
ECCS	Emergency Core Cooling System
EMO	Emergency Measures Organization
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
FAF	Fuelling Against the Flow
FEARP	Federal Environmental Assessment and Review Process
FNEP	Federal Nuclear Emergency Plan
FWF	Fuelling With the Flow
GAI	Generic Action Item
GCRM	Groupe communications et relations avec le milieu
GMA	Group of Medical Advisors
HCLPF	High Confidence of Low Probability of Failure
HFEPP	Human Factors Engineering Program Plan
HMS	Hydro Media Centre
HTS	Heat Transport System
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IIPA	Independent and Integrated Performance Assessment
INES	International Nuclear Event Scale
INEX	International Nuclear Exercises
INPO	Institute of Nuclear Power Operations

IRS	Incident Reporting System
LOECC	Loss Of Emergency Core Cooling
LLOCA	Large Loss Of Coolant Accident
LOCA	Loss Of Coolant Accident
LOR	Loss Of Regulation
LWR	Light Water Reactor
MACSTOR	Modular Air-Cooled STORage
MCR	Main Control Room
mSv	millisieverts
NAOP	Nuclear Asset Optimization Plan
NBEMO	New Brunswick Emergency Measures Organization
NB Power	New Brunswick Power
NEA	Nuclear Energy Agency
NOC	Nuclear Oversight Committee
NPAG	Nuclear Performance Advisory Group
NPVs	Nuclear-Powered Vessels
NRCan	Natural Resources Canada
NSC(ACT)	Nuclear Safety and Control (Act)
OCP	On-site Contingency Plan
OEB	Ontario Energy Board
OEC	Off-site Emergency Centre
OECD	Organization for Economic Cooperation and Development
OHN	Ontario Hydro Nuclear
OPEX	Operating Experience
OP&P	Operating Policies and Principles
OSCQ	Organisation de la Sécurité Civile du Québec
PAG	Public Affairs Group
PEAC	Provincial Emergency Action Committee
PHTS	Primary Heat Transport System
PIP	Performance Improvement Program
PLGS	Point Lepreau Generating Station
PMUNE-G2	Plan des mesures d'urgence nucléaire externe à la centrale Gentilly 2
PNEP	Provincial Nuclear Emergency Plan
PNGS	Pickering Nuclear Generating Station
PSA	Probabilistic Safety Analysis
PSAR	Preliminary Safety Analysis Report
QA	Quality Assurance
RCP	Radiation Contingency Plan
RLE	Review Level Earthquake
ROPT	Regional Overpower Protection Trip
RPCG	Reactor Projects Coordinating Group
SCA	Secondary Control Area
SDS1	Shutdown System One
SDS2	Shutdown System Two
SDSE	Shutdown System Enhancement
SER	Significant Event Report
SMA	Seismic Margin Assessment

SOE	Safe Operating Envelope
SSFI	Safety System Functional Inspections
TAPNS	Technical Advisory Panel on Nuclear Safety
TOE	Technical Operability Evaluation
UEPS	Unusual Event Processing System
UER	Unplanned Event Report
USNRC	U.S. Nuclear Regulatory Commission
WANO	World Association of Nuclear Operators

INTRODUCTION

1. CANADA’S NUCLEAR POLICY AND THE RELATED GOVERNMENT STRUCTURE

The Canadian government gives top priority to the safety and protection of the populace and the environment from the operation of nuclear facilities. As a result, the nuclear industry is one of the most intensely regulated industries in Canada. The Atomic Energy Control (AEC) Act, passed by Parliament in 1946, soon to be replaced by the Nuclear Safety and Control (NSC) Act, along with the Nuclear Liability Act are the centrepieces of Canada’s legislative and regulatory framework (refer to Article7). Canada has updated its nuclear regulatory requirements to make sure that the most current standards and practices are in place.

The two lead government organizations in the nuclear field in Canada are:

- Atomic Energy of Canada Limited (AECL), responsible for the design, marketing and construction of CANDU power reactors;
- The Atomic Energy Control Board (AECB), created through the AEC Act, responsible for the legislative and regulatory requirements for the nuclear industry. Throught AECB, the federal government regulates the development, application and use of nuclear energy in Canada.

The federal government has funded nuclear research, supported the development and the use of nuclear energy and related applications for several decades. Federal government funds for research are approximately $100 million for research and development activities related to CANDU technology. Three utilities (Ontario Hydro, Hydro-Québec and New Brunswick Power) also contribute funding for the program.

Canadians have benefited in many ways from this investment:

- Nuclear energy supplies on average about 16% of Canada’s electricity. For the Province of Ontario, nuclear power provides cost-competitive base-load electricity.
- The medical world has improved cancer therapy and diagnostic techniques.
- The Canadian nuclear industry, which includes power generation, contributes several billions of dollars a year to the gross domestic product and results in the creation of more than 26,000 highly skilled jobs. These jobs are concentrated in the uranium industry, the three provincial nuclear utilities, and in approximately 150 engineering firms and private sector suppliers of CANDU equipment and services.
- Uranium continues to rank among the top 10 metal commodities in Canada for value of production.

2. NATIONAL NUCLEAR PROGRAMS PERTAINING TO NUCLEAR POWER STATIONS

2.1 AECL and its Role in National Nuclear Programs

AECL is a Canadian Crown corporation, established in 1952, that reports to the Parliament of Canada through the Minister of Natural Resources. AECL develops, markets, and manages the construction of CANDU power reactors and MAPLE research reactors. It also plays a leadership role in a vibrant Canadian nuclear industry. The CANDU success is a result of close collaboration with Canadian utilities and the private sector, and continues to make an important contribution to job and wealth creation throughout Canada.

AECL and its Canadian and international business partners have designed, engineered, supplied components, and managed the construction of CANDU units on four continents. AECL is a leading supplier of nuclear power products and continues to advance the research and engineering that supports its products as follows:

- provides research and development, and engineering and consulting services to CANDU plants at home and abroad;
- offers radioactive waste management products and services;
- carries out underlying reactor research;
- supplies CANDU and Light Water Reactor (LWR) support services;
- offers radioactive waste management products and services.

AECL is dedicated to continuous improvement and sustainable development focussing mainly on the CANDU business. It has more than 3,500 staff at sites and offices in the following locations throughout the world:

- the Canadian Provinces of Ontario, Manitoba and Quebec
- Seoul, South Korea
- Beijing, China
- Bucharest, Romania
- Ankara, Turkey
- Jakarta, Indonesia
- Moscow, Russia
- Buenos Aires, Argentina
- The Netherlands
- Cairo, Egypt

AECL UTILITY SERVICES
AECL supports CANDU utilities with a wide range of site, engineering, and research and development services to keep each nuclear power station operating at top efficiency. At the same time, it helps nuclear power stations to decrease operating and maintenance costs. Predictive Maintenance Services help utilities determine the current status and potential service-life of reactor systems and components. AECL's expertise includes corrosion and fouling analysis techniques and hot cell examinations of materials and components.

AECL's Engineering and Consulting Services are an effective resource for CANDU stations around the world. AECL experts provide engineering support and problem-solving, often as part of an AECL-utility team. AECL also provides new technologies to enhance maintenance practices for existing CANDU reactors. A reliable supply of heavy water is available from AECL.

AECL WASTE MANAGEMENT PRODUCTS AND SERVICES
AECL's expertise includes:

- high- and low-level radioactive waste management
- decontamination and decommissioning
- radiation protection
- tritium remediation technology

AECL's Underground Research Laboratory in Manitoba has been used by agencies from the United States, France, Japan, and other countries for contract and collaborative research.

AECL developed and markets MACSTOR (Modular Air-Cooled STORage), an above ground dry-storage systems for all spent nuclear fuel types. MACSTOR delivers superior cooling and shielding performance. Its features include easy fuel retrievability and exceptionally low construction and operating costs. A MACSTOR system is operating successfully at Hydro-Québec's Gentilly site.

2.2 Research and Development

The research and development activities to support the operating stations in Canada and the CANDU technology are administered by the CANDU Owners Group (COG) organization. COG is jointly funded by the three Canadian nuclear utilities and AECL.

The organizational structure of the COG program consists of:

- Directing Committee
- COG Operations
- Technical Committees
- Working Parties

The Directing Committee is formed of members of the COG partners and is responsible for:

- setting the objectives of the program;
- assessing the progress being made;
- administering the program's budget.

The COG Operations is a separate entity that administers and monitors the technical program. Seven Technical Committees manage the research work in different technical areas. For some of these committees, the work is divided among separate Working

Parties who are responsible for specific work disciplines. The Working Parties interact directly with the researchers, and provide information and recommendations to the Technical and the Directing committees. Members of the Technical Committees and Working parties represent all COG partners. The research work is mainly done at the laboratories of:

- AECL in Chalk River and Mississauga, Ontario; and Pinawa, Manitoba
- OHN in Toronto, Ontario
- Stern in Hamilton, Ontario

in addition to many subcontractors including some Canadian universities.

The size of the COG program can be indicated by the 1997/98 budget of M$92.7. The program has 126 staff members serving on the Technical Committees and Working Parties, and 380 contract officers responsible for administering one or more of the research projects.

See Annex 1.1 for COG organization structure, and technical description of the research work done under this program.

3. BASIC CHARACTERISTICS OF THE CANDU REACTOR

3.1 Brief Description of the CANDU Reactor

A CANDU reactor utilizes controlled fission in the reactor core as a heat source to supply steam and electrical power. However, unlike other reactors, the CANDU is fuelled with natural uranium fuel that is distributed among several hundred fuel channels. Each six-metre-long fuel channel contains 12 or 13 fuel bundles. The fuel channels are housed in a horizontal cylindrical tank (called a calandria) that contains cool heavy water (D_2O) moderator at low pressure. Fuelling machines connect to each fuel channel as necessary to provide on-power refuelling; this eliminates the need for refuelling outages. The on-power refuelling system can also be used to remove a defective fuel bundle in the event that a fuel defect develops. CANDU reactors have systems to identify and locate defective fuel.

The CANDU reactor heat transport system is shown schematically in Figure 1.1. Pressurized heavy water (D_2O) coolant is circulated through the fuel channels and steam heavy water coolant flowing through the fuel channels. The coolant carries the heat to steam generators, where it is transferred to light water to produce steam. The steam is used to drive the turbine generator to produce electricity.

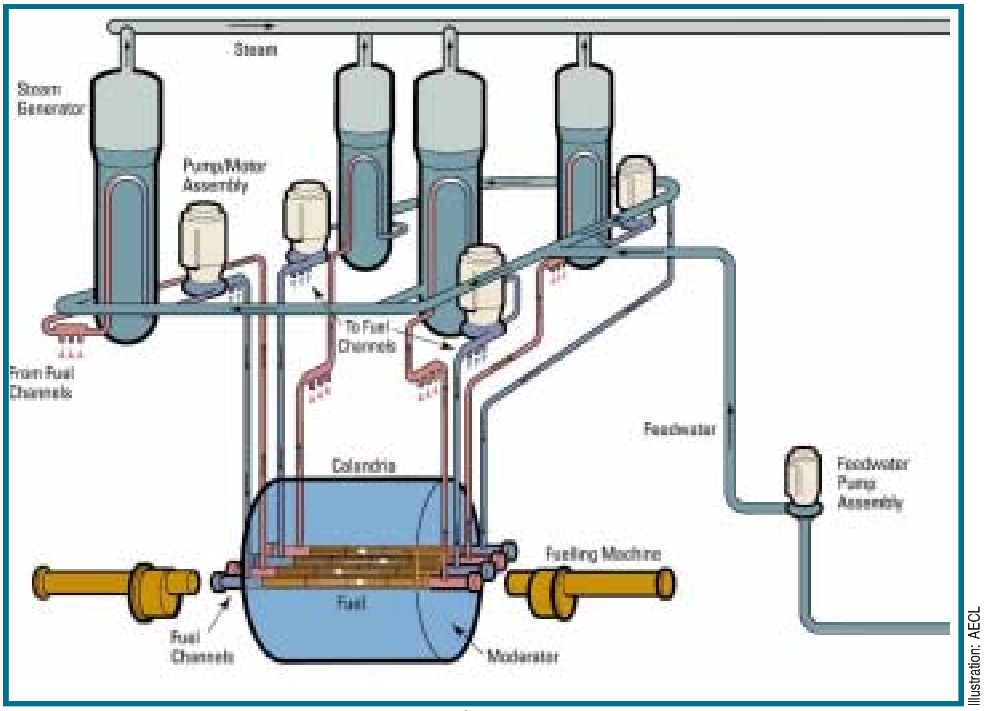

Figure 1.1: CANDU Reactor Heat Transport System

What follows are further descriptions of main reactor, fuel, and fuel changing systems:

REACTOR

The reactor comprises a stainless steel horizontal cylinder, the calandria, closed at each end by end shields, which support the horizontal fuel channels that span the calandria, and provide personnel shielding. The calandria is housed in and supported by a light water-filled, steel lined concrete structure (the reactor vault) which provides thermal shielding. The calandria contains heavy water (D_2O) moderator at low temperature and pressure, reactivity control mechanisms, and several hundred fuel channels.

FUEL HANDLING SYSTEM

The fuel handling system refuels the reactor with new fuel bundles without interruption of normal reactor operation; it is designed to operate at all reactor power levels. The system also provides for the secure handling and temporary storage of new and irradiated fuel.

HEAT TRANSPORT SYSTEM

The heat transport system circulates pressurized heavy water coolant (D_2O) through the reactor fuel channels to remove heat produced by fission in the uranium fuel. The heat is carried by the reactor coolant to the steam generators, where it is transferred to light water to produce steam. The coolant leaving the steam generators is returned to the inlet of the fuel channels.

MODERATOR SYSTEM

Neutrons produced by nuclear fission are moderated (slowed) by the D_2O in the calandria. The moderator D_2O is circulated through systems that cool and purify it, and control the concentrations of soluble neutron absorbers used for adjusting the reactivity.

FEEDWATER AND STEAM GENERATOR SYSTEM

The steam generators transfer heat from the heavy water reactor coolant to light water (H_2O) to form steam, which drives the turbine generator. The low pressure steam exhausted by the low pressure turbine is condensed in the condensers by a flow of condenser cooling water. The feedwater system processes condensed steam from the condensers and returns it to the steam generators via pumps and a series of heaters.

REACTOR REGULATING SYSTEM

This system controls reactor power within specific limits and makes sure that station load demands are met. It also monitors and controls power distribution within the reactor core, to optimize fuel bundle and fuel channel power within their design specifications.

SAFETY SYSTEMS

Four special safety systems (Shutdown System One (SDS1), Shutdown System Two (SDS2), the Emergency Core Cooling Systems (ECCS), and containment system) are provided to minimize and mitigate the impact of any postulated failure in the principal nuclear steam plant systems. Safety support systems provide services as required (electric power, cooling water, and compressed air) to the special safety systems.

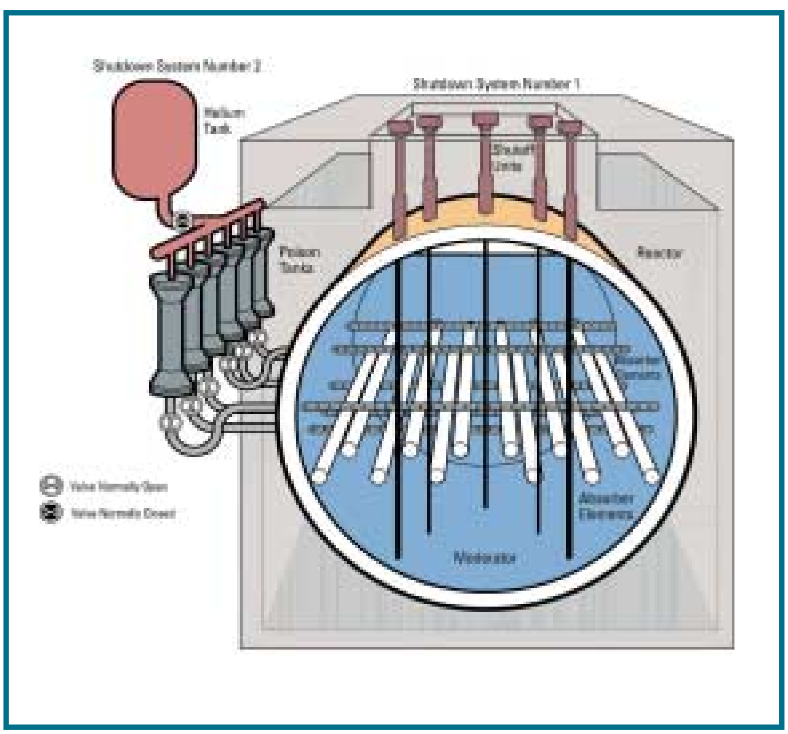

Illustration: AECL

Figure 1.2: CANDU Two Shutdown Systems

The two shutdown systems are shown schematically in Figure 1.2. SDS1 uses shutoff rods while SDS2 uses poison injection.

REACTOR ASSEMBLY

The CANDU reactor assembly, shown in Figure 1.3, includes several hundred channels contained in and supported by a horizontal cylindrical tank known as the calandria. Figure 1.4 shows the reactor face during construction. The calandria is closed and supported by end shields at each end. Each end shield comprises an inner and an outer tubesheet joined by lattice tubes at each fuel channel location and a peripheral shell. The inner space of the end shields is filled with steel balls and water, and is water cooled. The fuel channels, supported by the end shields, are located on a square lattice pitch. The calandria is filled with heavy water moderator at low temperature and pressure. The calandria is located in a light water filled shield tank. In the case of CANDU 6, this comprises a steel lined, water filled concrete vault, while CANDU 9 and most other CANDU designs utilize a water filled steel shield tank.

Horizontal and vertical reactivity measurement and control devices are located between rows and columns of fuel channels, and are perpendicular to the fuel channels.

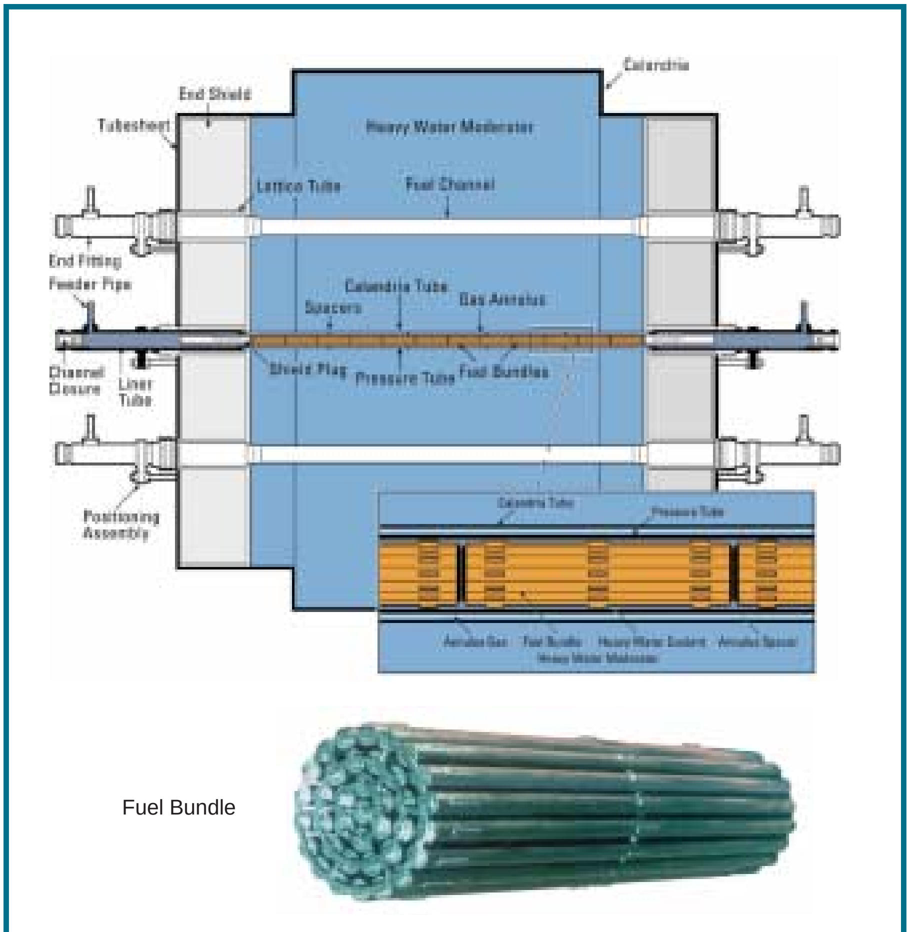

Illustration : AECL

Figure 1.3: CANDU Reactor Assembly

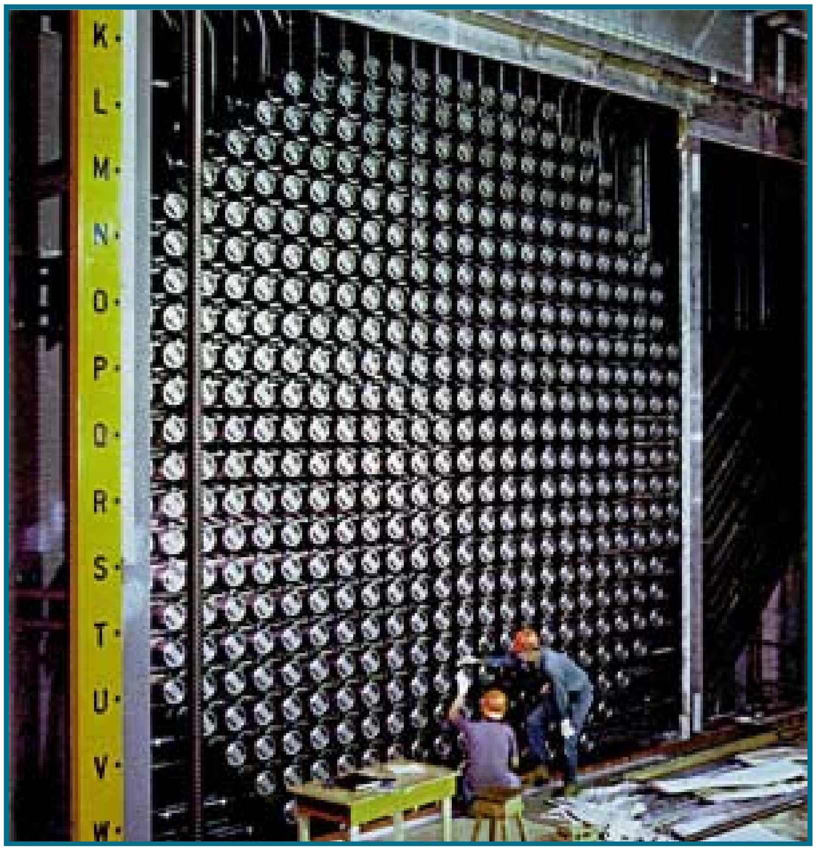

Photo : AECL

Figure 1.4: Reactor Face (during construction)

The fuel channels are also shown in Figure 1.3, with additional detail provided in the accompanying figure. Each fuel channel locates and supports 12 or 13 fuel bundles in the reactor core. The fuel channel assembly includes:

- a zirconium-niobium alloy pressure tube
- a zirconium calandria tube
- stainless steel end fittings at each end
- four spacers which maintain separation of the pressure tube and calandria tube

Each pressure tube is thermally insulated from the cool, low pressure moderator, by the CO_2 filled gas annulus formed between the pressure tube and the concentric calandria tube.

Each end fitting incorporates a feeder connection through which heavy water coolant enters/leaves the fuel channel. Pressurized heavy water coolant flows around and through the fuel bundles in the fuel channel and removes the heat generated in the fuel by nuclear fission. Coolant flow through adjacent channels in the reactor is in opposite directions. During on-power refuelling, the fuelling machines gain access to the fuel channel by removing the closure plug and shield plug from both end fittings of the channel to be refuelled.

FUEL
The CANDU fuel bundle consists of 37 elements, arranged in circular rings as shown in Figure 1.3. Each element consists of natural uranium in the form of cylindrical pellets of sintered uranium dioxide contained in a zircaloy-4 sheath closed at each end by an end cap. The 37 elements are held together by end plates at each end to form the fuel bundle. The required separation of the fuel elements is maintained by spacers brazed to the fuel elements at the transverse mid-plane. The outer fuel elements have bearing pads brazed to the outer surface to support the fuel bundle in the pressure tube.

The fuel handling system:

- provides facilities for the storage and handling of new fuel;
- refuels the reactor remotely while it is operating at any level of power;
- transfers the irradiated fuel remotely from the reactor to the storage bay.

FUEL CHANGING
The fuel changing operation is based on the combined use of two remotely controlled fuelling machines, one operating on each end of a fuel channel. This is shown in the schematic of Figure 1.5. New fuel bundles, from one fuelling machine, are inserted into a fuel channel and the displaced irradiated fuel bundles are received into the second fuelling machine at the other end of the fuel channel. Typically, either four or eight fuel bundles in a fuel channel are exchanged during a refuelling operation. For a CANDU 6 size reactor (380 fuel channels), about 10 fuel channels per week are refuelled.

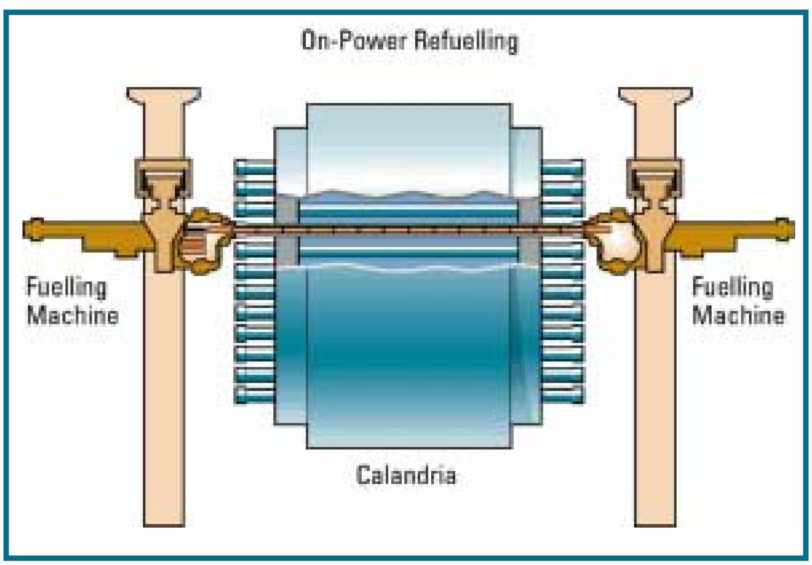

Illustration : AECL

Figure 1.5: CANDU On-power Fuelling

Either machine can load or receive fuel. The direction of loading depends upon the direction of coolant flow in the fuel channel being fuelled, which alternates from channel to channel.

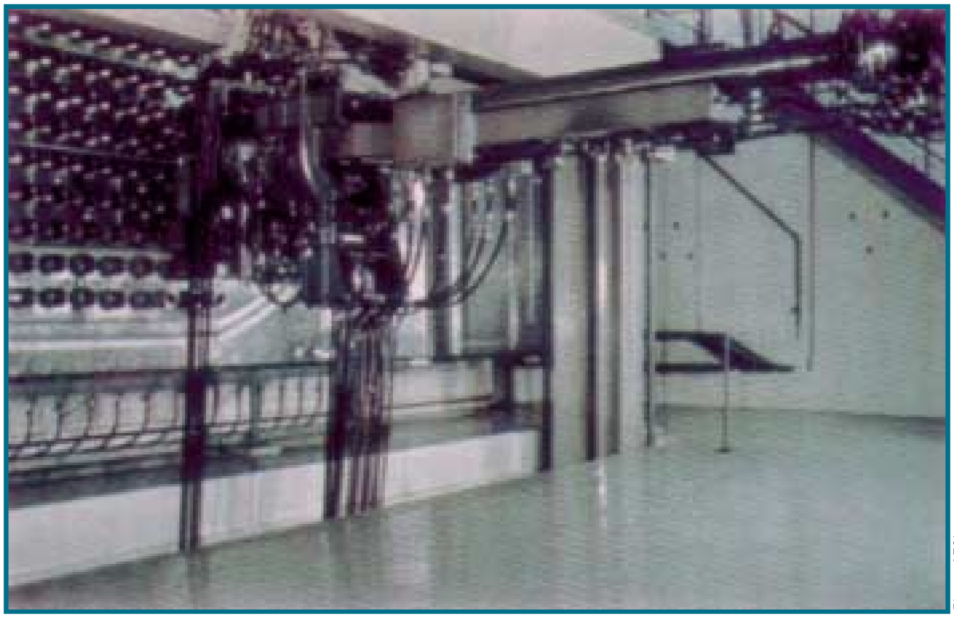

Figure 1.6 Fuelling Machine in Operating Position at Face of Reactor

The fuelling machines receive new fuel while they are connected to the new fuel port and discharge irradiated fuel while connected to the discharge port. The fuelling machine is shown in Figure 1.6.

The entire operation is directed from the control room through a preprogrammed computerized system. The control system provides a printed log of all operations and permits manual intervention by the operator.

3.2 CANDU Inherent Safety Features

There are a number of inherent safety features of the CANDU design that are summarized as follows:

- Reactivity devices work in the low pressure moderator, not in the high pressure and high temperature coolant, and therefore are not subject to pressure-assisted ejection.

- Criticality of CANDU bundles in ordinary (light) water is not possible, removing a concern in severe accidents. In fact, the CANDU core geometry is near the optimum reactivity so that severe core damage accidents that could rearrange the core structure will tend to make sure there is a shutdown.

- Because of on-power fuelling, the in-core power distribution reaches an equilibrium in less than a year and then remains virtually unchanged for the remainder of the reactor's operating life. Therefore analysis of the reactor core behaviour as a result of the postulated accidents is relatively simple.

- The pressure tube concept is used in identifying the location of fuel defects, and on-power fuelling permits removal of defective fuel from the core as soon as it is identified or whenever convenient. This helps to keep the heat transport system essentially free from fission product activity. A clean heat transport system allows maintenance work to be done with relatively little exposure of personnel to radiation.

- The effect of changes in operating parameters on reactor power is slow in CANDU reactors because of long neutron lifetime. This characteristic allows the use of slow-acting control devices. These regulating devices, acting alone, are capable of controlling reactor operation over the entire operating range.

- The use of natural uranium fuel and heavy water leads to a design characterized by good neutron economy and low excess reactivity. There is little reactivity worth in the control devices (~20 mk total and typically less than 1-2 mk per device) because the burnup compensation is done by on power refuelling. This limits the potential severity of accidents due to a loss of reactivity control. The largest positive reactivity insertion is from a large Loss Of Coolant Accident (LOCA) and is well within the capability of mechanical and hydraulic shutdown systems. The reactivity feedback from steam line breaks, cold or light water injection, or sudden turbine stop valve closure is negative.

- Low temperature and low pressure moderator provides an ideal location for neutronic measurements.

- The cold, low pressure heavy water moderator, about 1 cm away from the fuel in the channels, acts as an emergency heat sink following a loss of coolant even if the ECCS fails to inject water. The heat removal occurs from the fuel through the pressure tube and calandria tube to the moderator. The moderator can remove more than 4% of the total thermal power, enough to accept decay heat indefinitely.

- The pressure tubes are relatively small in size (10 cm diameter). In a severe accident, such as a LOCA combined with a loss of emergency core coolant injection, the pressure tubes will sag and/or strain into contact with the calandria tube where further deformation will be arrested by the cooling of the moderator system. Should channel failure occur (for example, due to a further equipment unavailability resulting in a loss of moderator heat removal), then such failures will be spread out in time “softening” the load on containment. Direct containment heating or containment damage due to massive failure of a reactor vessel at high pressure, is precluded.

- The bottom of the large calandria vessel provides a spreading and heat removal area for core debris following a severe core damage accident.

- The calandria vessel is surrounded by a shield tank containing light water for biological and thermal shielding. In severe core damage accidents, this tank also absorbs decay heat either from the moderator liquid or from direct conduction from debris inside the calandria vessel.

In addition to these inherent safety characteristics, CANDU includes engineered systems to enhance reactor safety. They include the following systems:

- Two redundant, independent, diverse, separated shutdown systems, testable in operation to demonstrate an unavailability of less than 10^{-3}. They share no devices with the control system nor with each other. This triple layer of defence removes the need during reactor design to consider the consequences of accidents without shutdown. Likewise they are not a significant risk contributor in Probabilistic Safety Analysis (PSA) terms.
- A shutdown cooling system, which removes decay heat at full temperature and pressure conditions, precluding the need for depressurization after a loss of heat sink.
- In addition to two standby diesel generators in Group 1 safety systems, there are also two independent, separately-located emergency diesel generators in Group 2 so that a loss of all AC power is of low frequency.
- Two control rooms, the Main Control Room (MCR) and Secondary Control Area (SCA), each of which can independently perform the safety functions of shutting down the reactor, removing decay heat, and monitoring the status of the plant.

Performance of these systems is evaluated as part of regulatory-required safety analysis included in Safety Report and Probabilistic Safety Assessment documents.

4. CANADIAN PHILOSOPHY AND APPROACH TO SAFETY OF NUCLEAR POWER STATIONS

In Canada, the prime responsibility for safety rests with the owner (licence holder) of the nuclear power station. It is the licensee’s responsibility to demonstrate to the satisfaction of the regulator that the nuclear power station can and will be operated safely. The licensee must show that the design meets all applicable performance standards and will continue to do so throughout its design life. In keeping with this principle, the AECB asthe regulator has produced only very general performance standards for nuclear power stations. It is the licensee’s responsibility to translate these into more detailed design requirements and to submit these design requirements for acceptance by the AECB. When accepted by the AECB, the licensee’s design requirement documents become a part of the licensing basis for the nuclear power station and form the basis for future regulatory activities (such as inspections and change approvals).

HISTORIC DEVELOPMENTS

The development of the Canadian philosophy and approach to nuclear power station safety was heavily influenced by a serious accident at a Canadian research reactor in 1952. The essential principles that evolved were derived from the recognition that even well-designed and well-built systems fail. Therefore, there is a need for separate, independent safety systems that can be tested periodically to demonstrate their availability to perform their design functions.

In the mid-1960s, these concepts were formalized by the AECB into a set of criteria commonly called the Siting Guide. These criteria were based on the separation of plant equipment into three categories depending on its safety function:

- the first category, process or normally operating equipment;
- the second category, protective equipment designed to prevent fuel failure in the event that process equipment fails;
- the third category, provisions to contain releases of radioactive material in the event of fuel failure.

Subsequently, the protective shutdown systems, the ECCS, and containment system were combined into a single category of “special safety systems”.

The criteria impose limits on the maximum permissible frequency of serious process failures, and on the fraction of time special safety systems may be in a condition that renders them unavailable to perform their safety functions. There is also a requirement[1] that the special safety systems be testable and that they be routinely tested, during normal operation, to demonstrate that the unavailability on test is less than the specified limit. The limits imposed (one per three years for a serious process failure and 10^{-3} unavailability of special safety systems) were selected so that it would be practical to demonstrate, within a reasonably short operational period, that at least these minimal requirements were being achieved. However, more stringent requirements were imposed, as necessary, to reflect the safety importance of specific systems. For example, there was and is a requirement that the reactor regulating system be designed so that a loss of reactivity control accident (a specific example of a serious process failure) has a frequency of less than 10^{-2} per reactor-year.

The criteria specify reference off-site dose limits to be used in safety analyses of:

- any serious process failure (single failure), and
- any combination of a serious process failure and failure of a special safety system (dual failure).

The criteria also require that the special safety systems be separate and independent of the process systems and of each other.

The reference dose limits (see Table 1.1) were determined on the basis of the maximum frequencies of the events. Since the maximum frequency for single process failures is one per three reactor-years, the reference dose limit for individuals in the public was chosen as equal to the corresponding one-year regulatory dose limit. For dual failures, with an assumed maximum frequency of one per 3000 reactor-years, the reference dose limit for individuals was that judged tolerable for a “once-in-a-lifetime” emergency dose.

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1. The meaning of the phrase “regulatory requirements” in this report is to be taken in the context described on the previous page (first paragraph under section 4). AECB regulatory documents are in general standards and guidelines that AECB expects the licensees to follow.

TABLE 1.1*

REFERENCE DOSE LIMITS UNDER NORMAL AND ACCIDENT CONDITIONS

Situation	Assumed Maximum Frequency	Meteorology to be used in calculation	Maximum Individual Dose Limits	Maximum Total Population Dose Limits
Normal Operation	100%	Weighted according to effect, i.e., frequency times dose for unit release	5 mSv/yr whole body	100 person-Sv/yr
Process Equipment Failure	1/3 yr.	As above	30 mSv/yr to thyroid **	100 person-Sv/yr thyroid
Process and Protective Equipment Failure	1/1000 yr.	Either worst weather existing at most 10% of time or Pasquill F condition if local data incomplete	250 mSv whole body 2.5 Sv thyroid ***	10^4 person-Sv 10^4 thyroid-Sv

* From Hurst D.G, and Boyd, F.C. paper, "Reactor Licensing and Safety Requirements," AECB-1059, 1972. Also, from Reactor Siting and Design Guide, 1964.

** For other organs use 1/10 ICRP occupational values.

*** For other organs use 5 times ICRP annual occupational dose (tentative).

The population reference dose limit for the dual failure situation was chosen to have a relatively small effect. On the basis of data available at the time, it was estimated that exposure to this dose might lead to about a 0.1% increase in the lifetime incidence of cancer in a population of one million people.

It was implicit in these criteria that applicants would be required to complete dual failure analyses of a complete range of serious process failures combined with a failure of the reactor shutdown system. As higher-power reactors were being designed in the early 1970s, it became increasingly difficult to predict the consequences of the more severe accidents involving failure to shut down. This led to the requirement for two shutdown systems. These were required to be conceptually different and sufficiently separate and independent of each other so that it could be assumed with reasonable certainty that they would not fail simultaneously. Also, each of the two systems must be shown to be effective for the complete range of design basis serious process failures. Analyses aimed at showing the effectiveness of one of the two shutdown systems may not claim any credit for action of the other shutdown system (except where necessary to show that a partial actuation of the other system could not impair the effectiveness of the system being assessed).

The various potential dual failures define the performance requirements for the special safety systems. For example, a LOCA accompanied by failure of the ECCS will lead to the release of fission products from the fuel that must be accommodated by the containment. Similarly, a LOCA with impaired containment sets the effectiveness required of the ECCS.

Although the single/dual failure approach adequately defined the required effectiveness of the special safety systems, some problems in coverage became evident. These included the following:

- failure to recognize the great variation in rates of occurrence and in the consequences of different single and dual failure scenarios;

- failure to deal effectively with the problem of safety support systems, such as electrical supply, compressed air, or service water, whose failure could in some cases result in simultaneous failure of a process system and a safety system;

- failure to deal effectively with the need for continuing availability of safety systems after an accident;

- failure to adequately address the need to design for, and analyse, the consequences of potential common-cause events, such as earthquakes and aircraft crashes, which might result in damage to both process and safety systems.

Such concerns pointed to a need for a more comprehensive approach to safety evaluation. This was identified not only by staff of the AECB and the utilities, but also by independent advisory groups set up by the AECB. The concerns were addressed as follows:

- The safety analysis rules were modified to recognize five accident classes (this is implemented in Consultative Document C-6 addressed in Article 7.2.4, with a copy provided in Attachment 7.10).

- A more probabilistic approach, the "safety design matrix," was introduced to assess support systems and human failures.

- The safety design matrix approach was also used to assess post-accident reliability of systems.

- A "two-group separation" approach was introduced to provide protection from common-cause failures. For localized failures (such as fires and internal floods), this involved separation by distance or barriers between the two groups of safety and safety support systems. For site-wide events (such as earthquakes, tornados and site floods), it involved qualification or protection of at least one of the two groups. In addition, the new safety analysis rules, in C-6, required explicit analyses of the consequences of some common cause events.

THE CONCEPT OF “DEFENCE-IN-DEPTH”

The concept of “defence-in-depth” is a key element of the Canadian nuclear reactor safety philosophy. This principle is required to be applied to all aspects of nuclear power station design, construction and operation. There are three elements to the defence-in-depth approach:

- accident prevention
- accident mitigation
- accident management

In the Canadian approach, these elements were addressed by requiring:

- measures to prevent accident initiators (for example, to make sure of an acceptably small frequency of serious process failures);
- redundant and diverse systems to prevent fuel failures in the event of serious process failures and measures to contain radioactive releases in the event of fuel failures (these include the special safety systems);
- provisions for both on-site and off-site emergency plans and procedures designed to cope with a wide range of equipment failures, including some that are considered beyond the design basis for the facility.

The first and most important principle of defence-in-depth is accident prevention. Accidents are prevented by making sure that the probability that a system or component in the plant will fail during operation is low. This is accomplished by:

- applying sound engineering practices during the siting, design, construction and operation of a power plant;
- using proven technologies;
- designing, building and maintaining the plant according to recognized codes and standards;
- making sure plant staff are appropriately trained;
- employing appropriate quality control and quality assurance methods in all phases of design, construction, and operation;
- monitoring events in other similar facilities to anticipate problems before they occur.

In the Canadian approach, the criterion selected was a limit of no more than one serious process failure every three reactor-years. A serious process failure is one that, given the complete absence of any protective system action, could lead to fuel failure. The limit of one serious process failure per three reactor-years meant that it would be apparent, in a relatively short operating period, if the limit was not being met. However, it was recognized that for certain more important events (such as loss of coolant and loss of reactivity control accidents) much lower frequencies of occurrence were required. Canadian nuclear power stations typically include additional protective equipment (which is separate from and independent of the special safety systems) to make sure there is an acceptably low frequency of challenges to the safety systems. Examples of such “process-protective” equipment include the setback and stepback functions of the reactor regulating system. These are designed to cope with some reactor control failures without requiring action by the safety shutdown systems.

The defence-in-depth approach also requires that provisions and procedures are in place to mitigate the consequences of accidents. Accident mitigation is achieved in the following ways:

- incorporating into the design reliable and effective special safety systems (SDS1, SDS2 ECCS, and containment) that are capable of limiting the consequences of accidents;

- incorporating multiple barriers to the release of radioactive material from the plant and measures to protect these barriers from damage due to accidents. These barriers include:

  - the fuel matrix,
  - the fuel sheath,
  - the heat transport system,
  - the containment system,
  - the exclusion zone;

- building in redundancy so that there are alternate ways of achieving the safety objective. Examples of this include:

  - the auxiliary boiler feed pumps, the shutdown cooling system, and the emergency water systems; all of which are capable of removing heat from the reactor;
  - a secondary control room for use should the main control room be unavailable for any reason;
  - redundant electrical power supplies and service water supplies to essential equipment.

The Canadian approach also requires that, in the event of a serious process failure combined with a complete failure of one of the special safety systems, the remaining safety systems must be sufficiently effective to prevent a large off-site release. That is, safety analyses must show that the consequences of the postulated event combination will not exceed the reference off-site dose limit for dual failures.

Finally, it is recognized that the consequences of reactor accidents can also be minimized by sound accident management on-site and off-site. Therefore, the last element of an effective defence-in-depth program is effective emergency planning. This is achieved by:

- developing operating procedures in advance to assist and guide operators in responding to accidents. These procedures include:

  - generic emergency operating procedures should the operators have problems diagnosing the accident;
  - training the operators in the use of these procedures by means of simulators and other techniques.

- developing effective off-site plans to minimize the consequences of a release of radioactive material to the environment. Off-site actions might include:

  - sheltering,
  - food and water interdiction,
  - distribution of potassium iodide pills,
  - evacuation.

THE ANALYTICAL TECHNIQUE

As was noted earlier in Historic Developments, the basic single/dual failure approach required development to deal with issues such as post-accident reliability of safety and safety support systems and human actions under accident conditions. In 1975, AECL's designers proposed the safety design matrix analytical technique to deal with:

- matters of interdependency
- post-accident operation
- actions requiring operator intervention

The safety design matrix contained a combination of fault trees and event trees. The analyst selected an event that is a potential safety concern, and the possible causes of this event were identified by a fault-tree analysis. Various postulated consequences were represented by diagrams of the sequences of events that could occur, accompanied by a narrative. This technique was used to assess whether the defences were sufficiently independent of the fault and whether the safety functions could be maintained with a sufficiently high degree of confidence.

Safety design matrices contributed to a better understanding of system behaviour and interactions under abnormal operating conditions and were used to identify required operator actions and desirable design modifications. The success of the technique depended upon the thoroughness of the analyst in identifying interdependencies between systems. Also, systematic validation of the analysis was problematic. Nevertheless, the safety design matrix was a major tool for assessing the reactors which were being designed and constructed in the late 1970s. This technique helped to identify design requirements such as redundancy and separation within safety-related systems. It also provided a method of identifying the operator actions required under fault or accident conditions.

The use of the safety design matrix has now been largely superseded by PSA techniques. For new Canadian power reactor designs (such as CANDU 9), a preliminary PSA is used to define system reliability requirements and subsequently a detailed PSA will be used to confirm the adequacy of the design.

Modifications have also been made to the basic single/dual failure safety analysis requirement, to reflect Canadian experience in applying the approach. In June 1980, the Consultative Document C-6, providing the requirements for CANDU safety analysis, was issued for comment. In 1983, agreement was reached to apply C-6, on a trial basis, to the licensing of Darlington nuclear power station.

The safety analysis requirements proposed in C-6 differed from previous practice in several respects:

- a requirement for a systematic review to identify postulated initiating events;
- five event classes (replacing the two categories of single and dual failure);
- more explicit consideration of combinations of postulated initiating events with failures of mitigating systems (not just the classical dual failures);
- more focus on reliability and testing requirements for mitigating systems;
- more consideration of station states, operating modes and operation at different power levels;
- more sensitivity and error analysis;
- more detailed information requirements for analytical codes;
- better identification of operator actions and of the available annunciations.

Although the Darlington trial application was, on the whole, a success, it did require lengthy discussions between AECB staff and the licensee in order to reach agreement on the interpretation of specific clauses. It was evident that a rewrite of C-6 was required. C-6, Rev. 1 which takes into account the lessons learned is currently at the final draft stage.

Table 1 of C-6, contains a list of single and dual failures whose consequences are expected to be analysed by the licensee, for example predicted, in order to demonstrate the safety of a nuclear power station. The list is not exhaustive, but it is intended to include the major failures of concern in a CANDU nuclear power station, as the list was based on more than 20 years of CANDU licensing experience. It is recognized, however, that variations in station design may either introduce new potential for failures of concern or eliminate some failures in the existing table. Therefore, the licensee is required to perform a systematic station review to confirm that all major failures of concern for the proposed station design have been identified and critiqued by AECB staff. In the event that a new failure mode is identified, the failure sequences in Table 1 of C-6 would serve as a frame of reference for the placement of the new failure, in terms of the relative likelihood of occurrence.

The requirements contained in C-6, Rev. 1, will represent an increase in both the scope and the rigour of design basis accidents, which operating power plants licensed prior to the Darlington station would have to consider. These requirements have not been readily accepted by the licensees who have proposed an incremental approach to address the scope of the design basis accidents.

THE DESIGN CRITERIA AND REQUIREMENTS

Table 1.2 lists the Power Reactor Safety Criteria and Principles published by the AECB in 1972. Since then, the treatment of single and dual-failures has been modified (see C-6) to reflect the range of probabilities within each of these two classes of accident. In addition, the current application of these criteria places more emphasis on related programmatic requirements, for example:

- requirements for quality assurance programs
- requirements for human factors engineering program plans

Apart from this, the basic criteria remain the same. However, it should be noted that application of the As Low As Reasonably Achievable (ALARA) principle has led to off-site doses during normal operation that are less than 1% of the maximum permissible doses set out in item 5 of Table 1.2.

It is the licensee's responsibility to translate this set of high-level criteria into detailed design rules, codes and standards for the process and safety systems. The licence application must identify the station systems important to safety and must contain documents that define the design requirements for such systems. When accepted by the AECB, these design requirement documents become part of the station's licensing basis and form the basis for future regulatory activities (such as inspections and change approvals). However, unlike regulations, they are plant-specific and would not necessarily be imposed on subsequent stations (although the trend in Canada has been towards increasing standardization). The intention is to set basic performance standards but leave the designer and operator some flexibility to develop the optimum way of meeting the basic safety requirements.

For example, designers recognized that, to provide the required defence-in-depth, it is necessary to protect systems that perform essential safety functions from common-cause failures. The "two-group separation" approach was proposed by the designer to separate by distance or barriers the two groups of systems, and to protect at least one of the two groups. The requirements for grouping and separation of systems were not prescribed by regulation but were defined in licensee documents (such as safety design guides) that were accepted by the AECB as part of the licensing process.

To supplement the general requirements, the AECB has issued regulatory policy statements on the requirements for the special safety systems. The AECB has also issued policy statements on overpressure protection requirements for reactor primary coolant systems, decommissioning, and reporting requirements. The regulatory laws and requirements governing the safety of nuclear power stations are described in detail in Article 7.2.

TABLE 1.2
POWER REACTOR SAFETY CRITERIA AND PRINCIPLES

1. Design and construction of all components, systems and structures essential to or associated with the reactor shall follow the best applicable code, standard or practice and be confirmed by a system of independent audit.

2. The quality and nature of the process systems essential to the reactor shall be such that the total of all serious process failures shall not exceed one per three years. A serious process failure is one that in the absence of protective action would lead to serious fuel failure.

3. Safety systems shall be physically and functionally separate from the process systems and from each other.

4. Each safety system shall be readily testable, as a system, and shall be tested at a frequency to demonstrate that its (time) unreliability is less than 10^{-3}.

5. Radioactive effluents due to normal operation including process failures other than serious failures (see 2 above), shall be such that the dose to any individual member of the public affected by the effluents from all sources, shall not exceed 1/10 of the allowable dose to atomic energy workers and the total dose to the population shall not exceed 100 person-Sv/year.

6. The effectiveness of the safety systems shall be such that for any serious process failure the exposure of any individual of the population shall not exceed 5 mSv and of the population at risk, 100 person-Sv.

7. For any postulated combination of a (single) process failure and failure of a safety system, the predicted dose to any individual shall not exceed: 250 mSv whole body, 2.5 Sv thyroid, and to the population, 10^4 person-Sv.

8. In computing doses in 6 and 7 the following assumptions shall be made unless otherwise agreed to:

   (i) meteorological dispersion that is equivalent to Pasquill category F as modified by Bryant[1]
   (ii) conversion factors as given by Beattie[2]

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1. Bryant, P.M., UKAEA report AHSB(RP)R42, 1964.
2. Beattie, J.R., UKAEA report AHSB(S)R64, 1963.

To summarize, the Canadian approach makes sure that basic safety requirements are met while allowing flexibility to deal with changing conditions. The safety characteristics of the CANDU have had, inevitably, a significant influence on the safety philosophy developed by the AECB, although the latter has, in turn, influenced the design.

5. A SURVEY OF THE MAIN THEMES AND MAIN SAFETY ISSUES CONTAINED IN THE REPORT

5.1 Main Themes in this Report

The main themes in this report are summarized as follows:

- The Canadian regulatory approach combined with the practices of the Canadian utilities provide a safety review process that gives ongoing assurance of nuclear power station safety. This process meets the Convention's requirements for a station safety review.

- Canadian reactor safety philosophy and reactor safety requirements, applied through the regulatory process, make sure that the risk to the workers, the public and the environment associated with nuclear power station operation is as low as reasonably achievable. The resultant risk is comparable to that of modern reactor designs.

- The Canadian regulatory agency has sufficient independence, legislative authority, and resources to make sure there is compliance and enforcement of regulatory safety requirements pertaining to nuclear power stations.

- Canadian legislative requirements are currently under revision (a new Act has already been passed by Parliament, new regulations are scheduled to be approved early in 1999). The new Act and regulations will provide more explicit and effective regulation of nuclear energy in Canada (see Article 7.2.1).

- The first responsibility for safety rests with the licensees. All licensees of nuclear power stations in Canada are publicly-owned electric utilities who report to a government body. All take their responsibility for safety seriously and are able to raise adequate revenue to support safe station operation.

- Safety reviews are performed periodically and for special events by the designer, utilities, and the regulator. This ongoing process is one of the fundamentals of nuclear power station licensing in Canada.

- Nuclear emergency response planning in Canada is well developed, including agreements with other countries or international organizations (see Article 16).

5.2 The Main Safety Issues in this Report

The main safety issues contained in this report are summarized as follows:

- The management of nuclear power stations, for some utilities, has not been given adequate attention by the owners over the last decade. This lack of management has resulted in declining standards of operation and maintenance to the extent that operation and maintenance are now only marginally acceptable. Configuration control has become poor. To date, programs to compensate for the effects of aging degradation have not been fully successful. Large remedial programs are being put in place to correct these deficiencies and to achieve the necessary standards of excellence (see main safety assessments performed in Articles 6.2 and 6.3).

- A limited number of outstanding safety issues remain to be resolved (see Article 6.2.3 and Annex 6.1). In some cases, these issues limit the power output of the nuclear power station.

6. A LIST OF NUCLEAR POWER STATIONS IN CANADA

A total of 25 nuclear power stations have been built in Canada. These include 14 reactors that are now in operation. The reactors that are shut down include:

- four (Pickering A, units 1, 2, 3 and 4) placed in the guaranteed shutdown state at the end of 1997 because they did not meet the regulatory requirement for an enhanced shutdown system. This was followed by a business decision taken by the licensee in 1997 to defuel and place these units in a lay up state.

- three (Bruce A, units 1, 3 and 4) that will be defuelled and placed in a lay up state during 1998 as a result of a business decision taken by the licensee in 1997.

At present, no power reactors are planned or under construction in Canada.

A list of the nuclear power stations in Canada and their operational status is provided in Table 1.3. Figure 1.7 provides a map for Canada with identification of the nuclear power stations sites.

TABLE 1.3
LIST AND STATUS OF NUCLEAR POWER STATIONS IN CANADA

Reactor	Type	Gross Capacity MW(e)	Construction Start	First Criticality	Operating Status
NPD	PHWR[1]	25	Jan. 1, 1958	Apr. 11, 1962	Shutdown: Aug. 1, 1987
Douglas Point	PHWR	218	Feb. 1, 1960	Nov. 15, 1966	Shutdown: May 4, 1984
Gentilly-1	HWLWR[2]	266	Sep. 1, 1966	Nov. 12, 1970	Shutdown: April 1, 1979
Bruce A, unit 2	PHWR	904	Dec. 1, 1970	July 27, 1976	Shutdown: Oct. 8, 1995
Pickering A, unit 1	PHWR	542	June 1, 1966	Feb. 25, 1971	Shutdown[3]: Dec. 31, 1997
Pickering A, unit 2	PHWR	542	Sep. 1, 1966	Sep. 15, 1971	Shutdown[3]: Dec. 31, 1997
Pickering A, unit 3	PHWR	542	Dec. 1, 1967	Apr. 24, 1972	Shutdown[3]: Dec. 31, 1997
Pickering A, unit 4	PHWR	542	May 1, 1968	May 16, 1973	Shutdown[3]: Dec. 31, 1997
Bruce A, unit 1	PHWR	904	June 1, 1971	Dec. 17, 1976	Shutdown[4]: Apr. 4, 1997
Bruce A, unit 3	PHWR	904	July 1, 1972	Nov. 28, 1977	Shutdown[4]: April 9, 1998
Bruce A, unit 4	PHWR	904	Sep. 1, 1972	Dec. 10, 1978	Shutdown[4]: Jan. 19, 1998
Gentilly-2	PHWR	675	Apr. 1, 1974	Sep. 11, 1982	Operating
Pickering B, unit 5	PHWR	540	Nov. 1, 1974	Oct. 23, 1982	Operating
Pickering B, unit 6	PHWR	540	Oct. 1, 1975	Oct. 15, 1983	Operating
Pickering B, unit 7	PHWR	540	Mar. 1, 1976	Oct. 22, 1984	Operating
Pickering B, unit 8	PHWR	540	Sep. 1, 1976	Dec. 17, 1985	Operating
Point Lepreau	PHWR	680	May 1, 1975	July 25, 1982	Operating
Bruce B, unit 5	PHWR	915	July 1, 1978	Nov. 15, 1984	Operating
Bruce B, unit 6	PHWR	915	Jan. 1, 1978	May 29, 1984	Operating
Bruce B, unit 7	PHWR	915	May 1, 1979	Jan. 7, 1988	Operating
Bruce B, unit 8	PHWR	915	Aug. 1, 1979	Feb. 15, 1987	Operating
Darlington, unit 1	PHWR	935	Apr. 1, 1982	Oct. 29, 1990	Operating
Darlington, unit 2	PHWR	935	Sep. 1, 1981	Nov. 5, 1989	Operating
Darlington, unit 3	PHWR	935	Sep. 1, 1984	Nov. 9, 1992	Operating
Darlington, unit 4	PHWR	935	July 1, 1985	Mar. 13, 1993	Operating

Notes:
1. Pressurized Heavy Water Reactor.
2. Heavy Water/Light Water Reactor.
3. Placed in a guaranteed shutdown state because the reactor did not meet the regulatory requirement for an enhanced shutdown system. This was followed by a business decision taken by the licensee in 1997 to place the unit in a lay up state.
4. Placed in a lay up state as a result of a business decision taken by the licensee in 1997.

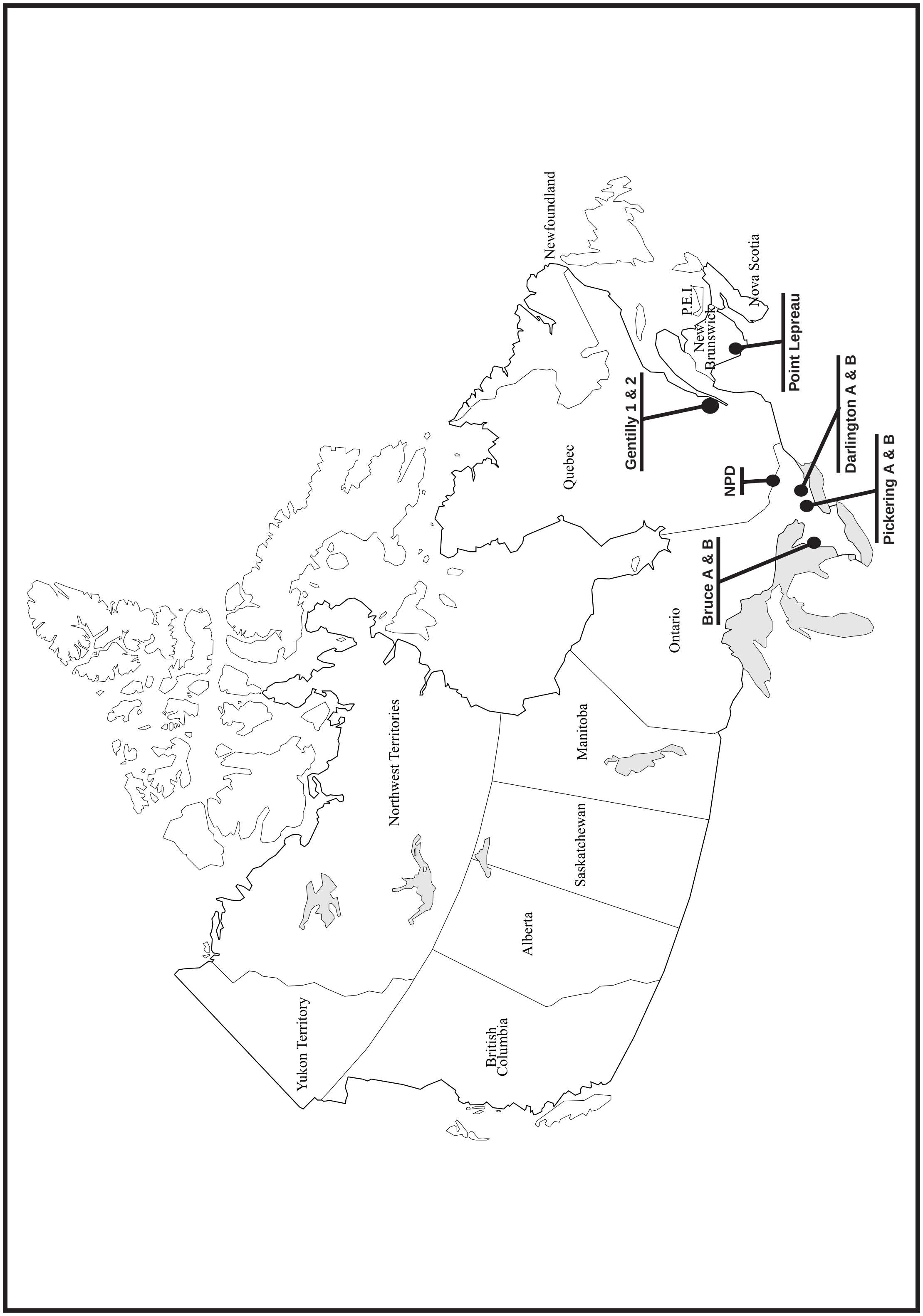

Figure 1.7: Nuclear Power Stations in Canada

ARTICLE 6
EXISTING NUCLEAR POWER STATIONS

6.1 LIST OF EXISTING NUCLEAR POWER STATIONS

A listing of the nuclear power stations and their operational status is provided in the Introduction, part 6, Table 1.3. All nuclear power stations built in Canada are of the CANDU type. The basic characteristics of the CANDU reactor, with the exception of NPD and Gentilly-1, are explained in the Introduction, part 3.

6.2 SAFETY ASSESSMENTS PERFORMED AND MAJOR RESULTS

In Canada, safety is the number one priority for nuclear power stations. It is a Canadian practice to perform safety assessments of nuclear power stations in response to significant safety incidents and operating experience both nationally and internationally. The following sections describe some of the major safety assessments performed, their results, and details of corrective actions performed.

6.2.1 Safety Assessments Following Major Incidents

Some safety assessments have resulted from major nuclear power station accidents. These accidents often focus attention on particular aspects of nuclear power station design and operation, even though the accidents occurred in several reactor types in various jurisdictions.

NRX (NATIONAL RESEARCH EXPERIMENTAL)
In 1952, a loss-of-regulation accident occurred at the Chalk River NRX research reactor. Following the accident, an in-depth assessment of safety shutdown was performed. The single most important technical lesson was that a reactor should always have a fast shutdown capacity available and that this capacity should be independent of any control system.

This shutdown capacity must always be available. Operational manoeuvres must be separated from the ability to insert swiftly sufficient negative reactivity into the core to achieve unequivocal shutdown. The assessment also identified the need for the development of a formal operational safety philosophy and the institutional structures to support it.

THREE MILE ISLAND
The small break Loss Of Coolant Accident (LOCA) in 1979 at Three Mile Island (non-CANDU station outside Canada) resulted in several assessments to determine the implications for CANDU safety. Some of the major recommendations of these assessments were:

- review of emergency procedures
- review of control room design to make sure that proper emphasis was placed on human factors

- review of operating experience feedback
- consideration of requirement for use of simulators in operator training

Most of these recommendations have been implemented. For example, the AECB provides licences to control room operators and their supervisors through a system of written and simulator-based examinations.

CHERNOBYL
Following the 1986 Chernobyl accident (non-CANDU station outside Canada), safety assessments were conducted to ascertain its implications for the safety and regulation of CANDU nuclear reactors. One of the major lessons learned was the importance of flux tilt as an initial operating state.

The AECB subsequently requested all licensees to reassess the effectiveness of all CANDU shutdown systems under circumstances in which the neutron flux/power distribution was severely distorted from its nominal conditions. Also, Ontario Hydro Nuclear (OHN) was requested to reexamine the safety of Pickering A reactors especially about accidents involving failure of the reactor control system and loss of coolant accompanied by unavailability of the shutdown system. Each of the four reactors at Pickering A has only one shutdown system. They were licensed before the requirement for two independent shutdown systems. Eventually, the AECB mandated the installation of a shutdown system enhancement in each of the Pickering A stations (see Article 6.3.3).

6.2.2 Safety Assessments in Response to Operating Experience (OPEX)

Most safety assessments are the result of incidents that occur during the routine operation of nuclear power stations. The most significant are:

PICKERING LOSS OF REGULATION
During the early days of operation (1971-1975) of the 4-unit Pickering A station, there were several events involving failure of the reactor regulating system that resulted in the rapid increase of power which was terminated by successful shutdown system action. Although these events had no immediate consequences, the station failed to meet requirements on limiting the frequency of serious process failures. Urgent action was necessary to improve the reliability of the regulating system. A detailed assessment resulted in a range of recommended improvements (mainly design, equipment, and operational changes) which were implemented both at Pickering A and subsequent stations. The frequency of such events has been much lower in recent years.

PRESSURE TUBE RUPTURE AT PICKERING A, UNIT 2
A 1983 pressure tube rupture event in Pickering Unit 2 led to the recognition of deuterium ingress into pressure tube material (hydriding) and the importance of avoiding contact between calandria and pressure tubes.

The pressure tube failure was caused by a combination of hydriding of the Zirconium-2 pressure tube and contact between the pressure tube and associated calandria tube that surrounds the pressure tube. This led to concentration of hydride at the cooler location of contact, blistering of the pressure tube, and the resulting failure. Although the calandria tube was not formally designed to withstand the full heat transport system pressure, it did in fact survive. The operators successfully shutdown the reactor and put it in a safe state. Emergency core cooling injection was not required.

This event revealed extensive hydriding and blistering that necessitated replacement of all pressure tubes at Pickering-A station. An alloy of Zirconium containing 2.5% Niobium is used as pressure tube material in all CANDU reactors following this event.

This incident also resulted in:

- garter spring (spacers between the pressure and calandria tube) relocation programs for all reactors (except Darlington)
- new garter spring design for Darlington

AECB fitness-for-service requirements preclude operation of reactors under conditions where pressure tubes could form blisters. This led to engineering programs at all stations to inspect pressure tubes and make sure that garter springs are properly positioned.

BRUCE LOSS OF REGULATION FROM LOW POWER

The AECB set up an investigation after two successive failures of the reactor control system during the restart of Unit 2 of Bruce A in 1992. The significance of these events was that they resulted in a need for shutdown system action to prevent fuel failure. They are therefore termed “serious process failures”, and should be limited to less than once per three years.

The investigation led to significant operational and procedural changes to the reactor regulating systems at all Bruce units.

DARLINGTON SHUTDOWN SYSTEM SOFTWARE

The Darlington shutdown systems make extensive use of computers in their control logic. Each of Shutdown System One and Shutdown System Two uses nine computers arranged in three channels. The trip computers, which receive analog signals from the field, compare these signals with appropriate setpoints and execute the trip function if required. Additionally, the trip computers transmit field signals, processed data and trip status to the Display/Test computer for display to the operator on two dedicated CRTs. The Display/Test computer is also used to control field devices used to test the system during routine availability testing. The third computer in each channel is the monitor which:

- carries out spread checks on system input signals to detect failing instrumentation;
- acts as an interface with the operator when system testing is carried out;
- passes test programs to the Display/Test computer for execution;
- acts as the operator interface for the transmission of neutron flux detector calibration data to the trip computer.

AECB staff conducted an extensive review of the hardware and software design of these systems. The most critical area of the design is the software used to determine the need for and to execute a trip when required.

Extensive verification of the trip computer software was required by the AECB. The review concluded that the programs could not be verified as providing safe behaviour under all possible circumstances. This is primarily due to the inadequate quality of documentation, software structure, and software testing. A program of software redesign and restructure, and documentation improvement, was introduced by OHN.

PICKERING A UNIT 2 SMALL LOSS OF COOLANT ACCIDENT (LOCA)
In December 1994, a small LOCA occurred at the Pickering A Unit 2 station. This event was caused in part by an incorrect design of pressure relief arrangements that led to chattering of the relief valve and subsequent failure of an adjacent pipe. Maintenance shortcomings also contributed to the event. The accident was terminated by successful operation of the Emergency Core Cooling System (ECCS), and there were no failures of fuel or significant release of radioactive material.

The AECB decided that all units remain shut down until extensive corrective actions were completed. These included extensive analysis and testing of the modified system. As a result, design changes (improved bleed condenser overpressure relief system) and improved maintenance (replacement of existing primary heat transport liquid relief valve actuator diaphragms) were required.

The safety assessment also revealed similar deficiencies in the primary heat transport overpressure relief systems at all CANDU reactors (except Pickering B) that resulted in both design and procedural changes.

POINT LEPREAU: WOOD COVER LEFT IN PRIMARY HEAT TRANSPORT SYSTEM
In 1995, a wood cover left in the heat transport system after planned maintenance at the Point Lepreau reactor resulted in the failure of a heat transport system pump shaft and put debris of wood and metal screws in the reactor cooling system. The major finding of the subsequent safety assessment was that there were shortcomings in the management of maintenance work and in the verification that followed maintenance.

New Brunswick (NB) Power undertook a major effort to remove the resultant debris. Robotic cameras were used to determine the nature and location of debris. Techniques were then developed and tested to remove larger debris items and to back flush each affected channel to remove wood and screws. Subsequent inspections revealed these efforts to be successful, thus allowing the AECB to approve restart of the reactor.

NB Power instituted inspections of fuel discharged from the affected quadrant of the reactor. These inspections were intended as a check in case debris or remaining screws enter a fuel channel. Only minor marks have been found on a small number of fuel bundles. Some of the channels that had fuel with marks on discharged fuel were inspected to confirm that the pressure tubes were in satisfactory condition. However, one pressure tube had to be replaced in 1998. It had a flaw caused by a screw that scored the tube during the recovery of debris.

NB Power also put improved procedures into place to make sure of foreign materials exclusion. These procedures are applicable to all station systems. Station management has also acknowledged that although they have been actively involved in promoting safety culture, day to day production schedule pressures may have led to an implicit and unrecognised contradiction of the safety culture message.

GENTILLY-2: HEAT TRANSPORT SYSTEM FLOW REDUCTION, AND ITS IMPACT ON REGIONAL OVERPOWER PROTECTION TRIP (ROPT) SETPOINTS
The monthly heat balance performed at Gentilly-2, had indicated that a reduction of the total reactor flow, of the order of 3 to 4%, occurred in the mid-90's. The consequences of that flow reduction have been assessed with particular emphasis on a reassessment of the ROPT setpoints. This led Gentilly-2 to self-impose a reduction of about 7% in their ROPT detector trip setpoints, which consequently led to a reduced reactor power of 97% during the period from the 1997 annual outage to the 1998 annual outage. Various options are considered to restore full power operation, including reduction in secondary side pressure and steam generator cleaning.

In addition to the ROPT reassessment, various loss of flow analyses have been reassessed with the observed flow reduction taken into account.

6.2.3 Other Safety Assessments

AECB GENERIC ACTION ITEMS (GAI)
AECB uses GAI as one method of pursuing concerns that arise through its continuing safety assessment, and may affect more than one station. The GAI have the following titles:

- Hydrogen Behaviour in Containment
- Core Cooling in the Absence of Forced Flow
- Assurance of Continued Nuclear Station Safety
- Post-Accident Filter Effectiveness
- Reactor Operation with a Flux Tilt
- Best Effort Analysis of ECCS Effectiveness
- Impact of Fuel Bundle Condition on Reactor Safety
- Molten Fuel-Moderator Interaction
- Pressure Tube Failure with Consequential Loss of Moderator Inventory
- Compliance with Bundle and Channel Power Limits
- Void Reactivity Uncertainty Allowance in LOCA Analyses

- Moderator Temperature Predictions
- Fire Protection for CANDU Nuclear Power Plants
- Feeder Pipe Fitness for Service

See Annex 6.1 for technical descriptions of the above GAI.

OHN SAFETY REVIEW

Between 1986 and 1988, a government-appointed commission conducted a safety review of the nuclear power stations of OHN.

Some of the key recommendations of the commission were:

- maximum and effective priority be given to finding a solution to the pressure tube problem, and to improved in-reactor monitoring;
- commission a study of factors affecting human performance throughout the utility for the purpose of achieving optimum efficiency and the maintenance of high standards of operation;
- large scale upgrading of process and safety systems at Pickering A to make sure of future safe operation;
- consistent policy be established by AECB governing the back-fitting of existing reactors.

PICKERING ENHANCED ASSESSMENT

In November 1996, AECB staff recommended that the licences for Pickering stations A and B be renewed for a period of six months instead of the normal licence period of two years.

AECB staff stated that it was too early to judge the effectiveness of many of the actions taken by OHN to improve operational safety. AECB staff also stated that an evaluation period was required to confirm that the commitments made by OHN were fulfilled and that the implementation of improvement was on track. To evaluate this, AECB staff carried out an enhanced assessment over the period of December 1996 to March 1997.

Eight assessment areas were chosen:

- Significant Event Report Follow-up
- Maintenance Program
- Procedural Compliance
- Radiation Protection Program
- Self-Assessment
- Training
- Surveillance Testing and Monitoring
- Reactor Safety During Outages

The results were presented to the Board (see footnote 2 in article 7.1) on May 15, 1997 for initial consideration of Pickering licence renewals. The assessment results were grouped into three categories:

Category 1: The results of these assessments were very encouraging. Significant progress had been made in implementing new programs that will be both effective and sustainable. The assessment areas considered to be in Category 1 were:

- radiation protection
- significant event report follow-up
- nuclear safety during outages

Category 2: In this category, the results of these assessments were less encouraging. Although considerable effort had gone into improving performance in these areas, the sustainability and effectiveness of the programs being implemented was questionable unless changes were made. The assessment areas considered to be in Category 2 were:

- self-assessment
- training
- surveillance testing and monitoring

Category 3: In this category, major improvements were required, or else success in either the effectiveness or sustainability of the programs being implemented could not be expected. The assessment areas considered to be in Category 3 were:

- maintenance program
- procedural compliance

OHN INDEPENDENT INTEGRATED PERFORMANCE ASSESSMENTS (IIPA)
In January 1997, OHN announced that a team of nuclear industry experts from the United States, called the Nuclear Performance Advisory Group (NPAG), had been employed to help manage its nuclear program and to implement needed improvements in OHN operations. In the spring of 1997, NPAG initiated a series of detailed reviews of OHN's operations at its Pickering, Bruce, and Darlington nuclear power stations and in OHN's Head Office groups. These reviews, called IIPA and "Safety System Functional Inspections" (SSFI), were carried out in April and May of 1997 with the objective of developing "an integrated, accurate, and comprehensive understanding of the performance of OHN".

The IIPAs applied performance objectives and criteria that have been developed according to criteria of the World Association of Nuclear Operators (WANO), the U.S. Nuclear Regulatory Commission (USNRC) and OHN. Their results were judged against industry standards (defined by OHN as the key parameters used by the Institute of Nuclear Power Operations (INPO)/WANO in their performance monitoring program). These key parameters include the essential nuclear safety and environmental aspects of operation. In addition, OHN applied criteria that were related to cost competitiveness such as the efficiency of station operations, the use of resources and long term viability of corporate assets. OHN did not generally judge the IIPA results against Canadian regulatory requirements or standards.

The performance areas assessed in the nuclear generating station IIPA included:

- operations
- maintenance
- training
- quality
- engineering
- radiation protection
- chemistry
- emergency preparedness

The Corporate Support Services IIPA covered:

- engineering
- radiation protection
- chemistry
- emergency preparedness

The Regulatory Affairs and Nuclear Assurance IIPA covered:

- safety focus/culture
- problem solving and self-evaluation
- organization/management, programs/processes and regulatory interface

The Organizational Effectiveness IIPA covered:

- standards setting and performance monitoring; processes and procedures
- work organization
- resource utilization
- information technology; materials management
- configuration management
- organizational systems

The purpose of an SSFI is to confirm that the system or function being reviewed will fulfill its safety mission and to make sure that inadequacies do not exist that may cause challenges to safety systems. SSFI are modelled after the "vertical slice" audit methodology developed by the USNRC, and reviews the design and maintenance of a system by investigating the following among other areas:

- change control
- operations
- maintenance
- training

OHN's Board of Directors received the results of the IIPA and SSFI and an OHN report entitled "Basis for Continued Operation" on August 12, 1997. The conclusions of these studies are critical of the management of OHN. They identify a number of shortcomings in the operation and maintenance of the nuclear generating stations.

OHN states in the IIPA and SSFI reports that the reports were, by design, negative in slant and emphasize the weaknesses in performance rather than the strengths. The reports conclude that the stations can continue to operate safely at the same time as the short- and long-term improvements are implemented.

As part of this review, OHN also carried out SSFI on the following systems:

- Darlington Compressed Air Systems
- Fire Protection (all OHN sites)
- In-service Environmental Qualification Program (all OHN sites)
- Bruce A Emergency Coolant Injection System
- Pickering Electrical Distribution System
- Bruce B Service Water Systems

OHN's SSFI and IIPA reports concluded that in the majority of cases there was a failure to control change and a lack of configuration management. OHN found that this has led to the absence of accurate up-to-date records of changes in addition to incomplete evaluations of the impact of these changes on risk to the station. AECB staff audits, assessments, and routine regulatory work have had similar findings and reached similar conclusions as the IIPA and SSFI (see above: Pickering Enhanced Assessment).

Environmental Qualification (EQ) was the subject of an OHN SSFI. Its conclusions state that the EQ program at OHN suffers from inaction and management neglect.

Many IIPA and SSFI findings deal primarily with maintenance, testing, inspection, and surveillance. These findings identify wide-ranging problems, including:

- deficient procedures
- poor planning and resourcing
- poor foreign material exclusion
- poor quality of work
- deficient post maintenance testing

Some problems are equipment-specific and others are programmatic. Poor maintenance may have been a significant contributor to the problems that OHN now faces. OHN had begun to use, with limited success, modern maintenance techniques such as:

- reliability-centred maintenance to identify safety important equipment for preventive maintenance
- integrated operational planning for managing maintenance resources
- foreign material exclusion practices

Some of the findings comment on the lack of standardization between OHN stations. Other findings comment on the lack of resources for various activities associated with station operation. A percentage of the findings comment on poor management and leadership. It should be noted that in 1996, OHN established a Leadership and Management Training Department dedicated to nuclear power station personnel and began upgrade training for managers and supervisors. AECB staff is currently working to develop tools and techniques to assess licensee organizational and management effectiveness. IIPA and SSFI findings also comment on the lack of appropriate safety focus. AECB staff members, in their routine regulatory work, have seen examples of managers acting conservatively with regards to nuclear safety.

A number of IIPA and SSFI findings deal with procedures and compliance to procedures. Ontario Hydro concluded that procedures are deficient and that there is a lack of compliance to those procedures. IIPA and SSFI also indicated weaknesses in the content and clarity of the Operating Manuals and the Abnormal Incident Manual.

A large number of findings relate to deficiencies in self-evaluation of performance, problem identification, corrective action process and quality assurance programs in general. OHN is reasonably efficient in identifying specific technical deficiencies in its nuclear power stations. However, OHN has not always identified human performance deficiencies or completed the necessary root cause determination for specific technical deficiencies. Consequently, OHN could not take the appropriate corrective actions. Even where OHN had identified the appropriate corrective actions, in many cases those actions were not taken in a timely manner.

Almost 10% of the IIPA/SSFI findings deal with training issues, and comment on:

- lack of training policies
- inadequate training manuals
- inadequate training resources
- poor continuation training
- lack of on-the-job training

AECB staff evaluations of OHN training to date have noted that OHN does have a training policy. However, there are gaps in supporting documentation. More significant, however, is that OHN training programs do not adhere to its own training policy.

AECB staff received formal assurance that OHN considered it safe to continue operation of the reactors based on the findings of the IIPA. OHN used a process called Technical Operability Evaluation (TOE) as the key to determine the acceptability of continued operation as a result of current and future review findings. Probabilistic risk assessments were used to provide additional support for the results of the TOE. AECB staff found that the TOE process was formal and rigorous. The process has good quality assurance aspects. The findings were dispositioned in a timely manner and decisions documented and supported with hard evidence. The process also adds the benefit of an external review to the decision. The TOE provides a formal process for documenting and tracking deficiencies from identification through closure. Ultimately relying on the decision-making capability of plant personnel, the process increases the involvement of the station’s senior personnel in station operations.

Neither the AECB staff’s detailed reviews of individual IIPA and SSFI findings nor the integration of these individual findings revealed an issue that required immediate licensee action to put OHN stations into a safer state. AECB staff concurred with the OHN conclusion that systems important to plant safety can be expected to perform their safety function successfully when called upon to do so. They believe that station operating experience supports this expectation. AECB staff followed up with its own vertical-slice inspection of a shutdown system at Pickering B (see below). The inspection results also support this expectation.

Although no single IIPA/SSFI observation or finding alone posed a serious safety concern, the number and variety of these findings indicated some reduction of barriers that constitute the “defence-in-depth” approach to safe nuclear power station operation. In response to these findings, OHN has undertaken an aggressive program of improvement efforts to:

- restore originally designed safety margins;
- update system design information;
- strengthen the design control process.

PICKERING ENHANCED COMPLIANCE INSPECTION

In December 1997, an AECB staff team carried out an enhanced compliance inspection of shutdown system one at Pickering B. The objectives of this inspection were to determine if Shutdown System One is able to perform its intended safety functions and if OHN is providing adequate technical and operational support to the system. The inspection was modelled after the SSFI developed by the USNRC. The team witnessed several scheduled routine tests, and observed activities such as:

- system walkdown by the responsible engineers
- maintenance work performed by control technicians
- activities performed by AECB authorized staff in the control room

The team also conducted interviews with OHN staff and reviewed the documents and drawings related to system design and operation. The team observed strengths such as:

- on-line testing to verify availability
- conservative decision making
- records of high actual past availability and predicted future availability

AECB staff team also found weaknesses indicating that some safety margins built in by the application of defence-in-depth principles are being eroded. Some examples are:

- deficiencies in the calibration program
- inadequacies in the design of human-machine interface
- impairment limits of some trip parameters are not well defined or supported by safety analysis for current plant equipment conditions

AECB staff concluded that Shutdown System One in Pickering B has been designed to perform its intended safety function and that OHN is providing adequate technical and operational support to the system.

POINT LEPREAU PERFORMANCE IMPROVEMENT PROGRAM (PIP)
As a result of the decreasing safety performance observed at Point Lepreau during 1996, the AECB added an extra condition to the current operating licence for NB Power to report to the Board (see footnote 2 in Article 7.1) every six months on actions it is taking to improve safety performance. In response to this condition, NB Power has initiated the PIP and updated the Board on the progress with their program at the April 1997, October 1997, and April 1998 Board Meetings.

NB Power initiated a number of independent assessments of the station through WANO and other external consultants. The results of these assessments, along with internal assessments, were used to identify areas where improvement was required. By April 1997, NB Power had come up with 43 projects to be carried out under PIP. It also determined that the root cause of the problems at Point Lepreau was a corporate failure to understand the lifetime management of the station and failure to develop strategic planning and appropriate resources.

In the summer of 1997, NB Power decided to focus on a limited number of PIP projects related to improving human performance and safety culture. Failures in these areas were identified as the root causes of the unplanned events that occurred at the station in 1996. These focus projects included:

- management effectiveness
- conduct of work
- improved event investigation

Most of these focus projects have now been completed.

Recently, NB Power has added three new projects to their PIP. These projects address:

- station communications
- self-assessment
- simplifying the work control process

These new projects were initiated in response to feedback from the initial PIP projects, as well as in response to issues raised by the AECB Quality Assurance audit.

For PIP projects aimed at improving safety culture, AECB staff determined that the training and processes being developed, such as the conduct of work station instructions, were an important step in developing a good safety culture. Some improvement has been observed at the management level and production pressures are no longer apparent during outages. However, cultural problems still appear to exist at the station. Event analyses conducted by NB Power show that the most prevalent factors to events are still related to human performance.

GENTILLY-2 SAFETY CULTURE SELF-ASSESSMENT
In September 1997, a self-assessment of safety culture was initiated at Gentilly-2 nuclear power station. This assessment was scheduled in three stages as follows:

- In the first stage, a survey was done using a questionnaire distributed to all personnel on site. This survey consisted of fifty sentences describing attributes of a good safety culture. Personnel were asked to state whether they agree that the situation on site satisfies such attributes. They also were asked to add their comments on the survey to express other concerns that they may have.

- In the second stage, the managers were met in interviews with the employees where they would elaborate on the same topics, and also discuss the strengths and weaknesses of their unit according to different organizational factors. The results of the survey for their unit, from the first stage, were also discussed to insure consistency. This part of the assessment is in progress.

- In the third stage, the results of the exercise will be used in producing a report to senior management in order to develop a strategy to correct any weakness to be found in the process. Also, the type of training that may be needed for personnel and managers will be determined.

The self-assessment is expected to be completed in the fall of 1998.

6.3 STATIONS THAT REQUIRED CORRECTIVE ACTIONS/ PROGRAMS FOR SAFETY UPGRADING

The following are corrective actions/programs for safety upgrading applied to specific power stations as results of safety assessments. Such actions/programs have either been completed or are in the process of being completed.

6.3.1 Darlington Nuclear Power Station

DARLINGTON LOSS OF FLOW
Under normal operating conditions, the reactor coolant is circulated by the primary heat transport pumps to cool the fuel inside the reactor. If one or all of these pumps stop, drying out of the fuel sheath and overheating of the fuel could occur. To protect the reactor against such accidents, the reactor shutdown systems monitor the flow and the pressure of the coolant and automatically shutdown the reactor if these are not within certain limits. Safety analysis is done to show that these limits cover the entire range of allowed operation and, that the chance of any fuel heat-up is very low.

An analysis in 1997 of the loss of flow revealed that, for a range of reactor power above 60%, the shutdown systems would not act as effectively as originally calculated, and an increase in fuel temperature could result. This finding led to significant power derating for several months. Procedural and hardware changes were implemented before the reactors returned to high power operation.

6.3.2 Bruce Nuclear Generating Station

BRUCE A/B POSITIVE REACTIVITY DUE TO FUEL RELOCATION FOLLOWING A LOSS OF COOLANT ACCIDENT
OHN assessment in 1993 of the reactivity effects for the movement of fuel bundles during a postulated LOCA resulted in the Bruce reactors being derated to 60% full power operation. An elongation of fuel channels that resulted from creep produced a gap at the end of the channels. Therefore, a break at one end would result in fuel movement which would increase the reactivity in the core. Since this assessment, OHN implemented a number of equipment and operational changes and provided better technical support and analysis before raising power. These reactors are still derated to 90% of full power operation.

BRUCE A EMERGENCY CORE COOLING SYSTEM (ECCS)
Following the recognition that ECCS could not prevent fuel failures for Large Loss Of Coolant Accidents (LLOCA), AECB staff requested OHN to conduct a system-by-system review of the impact, which resulted in design changes. AECB also requested that major shielding be added to the Bruce A ECCS. This request also led to a redesign of the Bruce B ECCS, which was under construction at the time.

During the 1970s, experimental research showed that the gravity-fed ECCS which was the original design at Bruce A station was incapable of meeting the original design requirements. At the instruction of the AECB, the reactors were back-fitted with a high pressure ECCS and heat exchangers. High pressure ECCS and heat exchangers were subsequently incorporated into all subsequent reactors.

BRUCE A CONTAINMENT
Tests requested by the AECB at Bruce A revealed that the design of dousing system headers was inadequate and required a major redesign. These tests also indicated that changes were necessary at Pickering and these were back-fitted. The AECB also required major improvements to the emergency filtered air discharge systems at Bruce.

BRUCE A/B BOILER TUBE DEGRADATION
Boiler tube degradation as a result of stress corrosion cracking and fretting has been observed at Bruce A and B nuclear power stations. After a review of the relevant information, the AECB required a significant expansion of OHN’s inspection and technical support activities.

OHN is committed to a boiler tube life management program that involves inspection and plugging to maintain the number of tubes at risk of failure within tolerable limits.

6.3.3 Pickering Nuclear Generating Station

PICKERING A REACTOR BUILDING LEAKAGE
In 1992, improvements of the devices isolating the Pickering A units from the main containment duct were required to meet AECB instruction to increase the time at which venting the containment becomes necessary following an accident. The AECB also required extensive repairs to the dome of the Unit 1 reactor building to make sure safety margins were maintained.

SHUTDOWN SYSTEM ENHANCEMENT (SDSE) AT PICKERING A
Pickering Nuclear Generating Station A (PNGS-A) reactors were licensed for operation by the AECB before the introduction of the regulatory requirement for two independent, diverse and fully capable shutdown systems. Therefore, the PNGS-A reactors were designed and built with only one fast-acting shutdown system and were judged acceptable on the basis of analysis presented at the time. Dual failure analysis that involved the loss of a shutdown event had been a licensing issue with the AECB since 1975.

Since the early 1980's, the shutdown system has been upgraded to improve its reliability and effectiveness so that the probability of shutdown failure would be extremely low. Upgrades include:

- increasing the number of shutoff rods from 11 to 21;
- upgrading the boiler room high pressure trip parameter;
- adding a boiler low level, a heat transport low pressure, and a boiler feedline low pressure trip parameters.

Following the 1986 Chernobyl accident, AECB staff requested OHN to reassess the safety of the PNGS-A reactors under the dual failure assumptions that involve the failure to shut down. In 1987, OHN submitted a revised analysis of the consequences of a LLOCA combined with failure to shut down. This analysis concluded that the structural integrity of containment would be maintained and that the dual failure reference dose limits would be met. AECB staff found this analysis to be speculative and concluded that the consequences could not be quantified with confidence. Following discussions between AECB and OHN staff, and because verifying results of loss of shutdown analysis to the satisfaction AECB would require expensive and time-consuming research programs which in themselves would not increase reactor safety, OHN decided to investigate the SDSE. This would reduce the probability of failure of shut down so that loss of shut down analysis would no longer be required.

The results of the investigation, and the enhancement proposed, were documented in several submissions to the AECB and include such improvements as detailed below:

- The final SDSE design that was approved provides a new set of triplicated trip sensors and trip logics augmented with new moderator dump logic.

- The SDSE trip parameters are neutron overpower, high log neutron rate, heat transport high pressure and low pressure and manual trip.

- The enhancement also includes the addition of two more shutoff rods.

The existing shutdown system and the SDSE are independent of each other from trip sensing to the final relay contacts in the shutoff rod drop logic and the moderator dump logic. Both the new and the existing logic trains will actuate all the shutoff rods. If power rundown characteristics are not satisfactory after a reactor trip, a dump signal is generated by the existing shutdown system and/or SDSE, either of which causes the moderator dump valves to open and shut down the reactor.

OHN committed to installing this enhanced shutdown system on all PNGS-A operating reactors by the end of 1997. On this basis, PNGS-A operating licence contains a condition mandating this installation.

SDSE was installed in Unit 4 of PNGS-A, the installation commissioning was successfully completed and the system was readied for future on-power testing. Some minor installation was also done on the other units. AECB staff was satisfied with the progress OHN made on the installation of SDSE. However, in August 1997, OHN announced that the PNGS-A reactors would be shut down at the end of 1997 and that work on the installation of SDSE would be suspended. In accordance with the requirements of the operating licence, OHN shut down all the four PNGS-A reactors by December 31, 1997, and made a business decision to lay them up. The restart of these units will require AECB approval.

6.3.4 Gentilly-2 and Point Lepreau Nuclear Generating Stations

AECB staff requested NB Power and Hydro-Québec to consider the potential effects of secondary side pipe failures. One of the main concerns was the protection of the main control room. The corrective actions required in 1992 were:

- identification of all practicable design changes
- enhanced protection of the main control room by a variety of means such as highly reliable in-service inspection and steam leak detection
- definition and demonstration of procedures for the secondary control room

NB Power and Hydro-Québec have followed up on these requirements. They have also put in place a risk reduction program including enhanced inspections programs.

6.3.5 All Nuclear Power Stations

POWERHOUSE DESIGN & ENVIRONMENTAL QUALIFICATION (EQ)
AECB questions on the effects of high pressure piping failures in the powerhouse at Bruce A and B resulted in:

- a major program of work on all stations, including the installation of large pressure relieving devices in Bruce, Pickering and Point Lepreau generating stations
- the back-fitting of a qualified electrical power supply at Bruce A
- a major environmental qualification program for all OHN stations
- design changes to protect safety equipment against harsh environment at all stations

EQ provides assurance that essential equipment will function in a harsh environment under accident conditions.

TWELVE-HOUR SHIFTS
Since 1989, the AECB has expressed concern about shift schedules for nuclear power station staff who work twelve-hour shifts. In 1989, only OHN power station shift

crews worked twelve-hour shifts. Beginning in 1992, AECB staff authorized twelve-hour shifts for Point Lepreau shift crews. In 1997, AECB staff authorized a one-year trial period for twelve-hour shifts at Gentilly-2 beginning April 1, 1997. The development of AECB policy on twelve-hour shifts is described below.

In response to Hydro-Québec's request to adopt twelve-hour shifts and to better manage and deal with future requests for changes to shift schedules, AECB staff issued a contract with the objective of assessing the proposed shift schedule for Gentilly-2 and consolidating the AECB's set of criteria. These criteria deal with six major categories:

- limits to hours of work,
- tracking and reporting hours of work,
- overtime policy,
- rest and recovery provisions,
- work organization,
- transition from 8- and 9-hour shifts to 12-hour shifts.

AECB staff expects licensees to:
- track employees' hours of work;
- report violations of the policies;
- have a comprehensive overtime policy;
- maintain a minimum shift complement;
- effectively manage the transition from 8- and 9-hour to 12-hour shifts.

AECB staff has identified a number of basic criteria for limits to hours of work, including:
- a normal shift length of 12 hours, excluding travel and turnover time;
- a regular schedule designed to average no more than 48 hours per week;
- limits on the amount of overtime in one year;
- a maximum of 60 hours in a seven-day period, including appropriate time off before and after;
- a maximum of five consecutive day shifts or four consecutive night shifts.

For rest and recovery provisions, AECB criteria include:
- after three or more consecutive night shifts, a minimum of 72 hours off;
- after three day shifts or two night shifts, a minimum of 48 hours off;
- after two day shifts, a minimum of 24 hours off.

All Canadian nuclear power station shift schedules are now in compliance with these criteria.

6.4 CANADIAN POSITION FOR CONTINUED OPERATION OF NUCLEAR POWER STATIONS

Although there has been some reduction in safety margins, standards of operation, and in material conditions at Canadian nuclear power stations since they were originally licensed, the level of defence-in-depth at these nuclear power stations is still acceptable and Canadian regulatory requirements are still being met or exceeded.

Although there have been improvements in both operation and design, it is generally recognized that further improvements are required in the standards of nuclear power station operation and maintenance in Canada to arrest further deterioration in the material conditions, restore safety margins, and enhance defence-in-depth.

The industry has therefore committed to plans and programs to improve the standards of nuclear power station operations in a timely manner.

The AECB monitors the execution of these plans to verify that:

- commitments are being kept
- the desired improvements are achieved within a reasonable period of time
- the improvements are subsequently sustained

Longer term operation of nuclear power stations may be made conditional upon timely and satisfactory completion of some safety-related improvements.

ARTICLE 7
LEGISLATIVE AND REGULATORY FRAMEWORK

7.1 A COMPREHENSIVE DESCRIPTION OF THE CANADIAN LEGISLATIVE AND REGULATORY FRAMEWORK

Canada is a confederation, with ten provinces and two territories administered by the central or federal government. The Canadian Constitution is expressed in the Constitution Acts of 1867 and 1982.

The provinces are self-governing in the areas of legislative power assigned to them by the Constitution Acts. These areas include:

- local commerce
- working conditions
- education
- direct health care
- resources in general

The Constitution Acts give the Parliament of Canada legislative power over works declared by it to be for the general advantage of Canada. The Parliament of Canada used this declaratory power when, in the Atomic Energy Control (AEC) Act of 1946, it declared works and undertakings constructed for the following purposes to be works for the general advantage of Canada and therefore subject to federal legislative control:

- the production, use and application of atomic energy;
- the research, or investigation, with respect to atomic energy;
- the production, refinement or treatment of prescribed substances (includes deuterium, fissile and radioactive materials).

The Supreme Court of Canada, in recent decisions, has affirmed federal legislative control based on the declaratory power and the power to legislate for the peace, order, and good government of Canada in matters having a national dimension.

The AEC Act also created the Atomic Energy Control Board (AECB) as the agency responsible for administering the Act. The Act, which was subsequently amended in 1954, was a short document that authorized and defined the powers of the Board[2]. The Board is composed of five people, one of whom is appointed President and Chief Executive Officer of AECB. Under section 9 of the Act, the Board was empowered to make regulations governing all aspects of the development and application of nuclear energy.

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2. The term “Board” is used in this document to refer to the five members appointed by the Govenor-in-Council. The initials “AECB” is used to refer to the agency and its staff.

The 1954 amendment to the Act transferred the responsibility for research and the exploitation of nuclear energy from the AECB to a minister designated by the government. As a result of this transfer of responsibility, Atomic Energy of Canada Limited (AECL), a government-owned company established in 1952, was made directly responsible to the designated minister. The AECB was clearly established as the regulatory agency.

On March 20, 1997, Bill C-23, the Nuclear Safety and Control (NSC) Act (see Article 7.2.1) received Royal Assent. This enactment will replace the AEC Act with a modern statute to provide more explicit and effective regulation of nuclear energy.

The only other legislation enacted by Parliament specifically about nuclear energy is the Nuclear Liability Act. This Act, which entered into force in October 1976, places total responsibility for nuclear damage on the operator of a nuclear installation.

The AEC Act imposed federal jurisdiction on areas involving nuclear energy applications that would otherwise have been under provincial jurisdiction. Examples of such areas include:

- occupational health and safety
- regulation of boilers and pressure vessels
- off-site emergency preparedness
- environmental protection

Under the Canadian Constitution, provincial laws may also apply in these areas if they are not directly related to nuclear energy and do not conflict with federal law. Because both federal and provincial laws may apply in some regulated areas, the approach taken has been to try and avoid duplication by seeking cooperative arrangements between the federal and provincial departments and agencies having responsibilities or expertise in these areas. Although these arrangements have been successful in achieving industry compliance, there has been a need to give them a firmer legal basis. The new NSC Act deals explicitly with the issues of incorporating provincial laws by reference and delegation of powers to the provinces for administration and enforcement of those laws.

Using its authority under the AEC Act and its regulations, the AECB maintains regulatory control over:

- power and research reactors
- nuclear research and test establishments
- uranium mines and mills
- uranium refining and conversion facilities
- fuel fabrication facilities
- heavy water production facilities
- particle accelerators
- radioactive waste management facilities
- prescribed substances and items
- radioisotopes

The AECB regulatory regime also includes the control of nuclear materials and other nuclear items for purposes of non-proliferation safeguards. This control provides assurance that Canada's national policies and international commitments about non-proliferation of nuclear weapons and other nuclear explosive devices are met. This is carried out in cooperation with other federal government departments and the International Atomic Energy Agency.

The AECB regulates nuclear power stations and nuclear materials through a comprehensive system that issues licences containing conditions that must be met by licensees. Regulatory control is also achieved by setting standards that licensees must meet. Some standards are prepared within the AECB, such as the requirements for special safety systems at nuclear power stations. Others are set by provincial authorities or national standards associations.

The AECB's licensing system is administered with the cooperation of federal and provincial government departments and agencies in such areas as health, environment, transport, and labour. The concerns and responsibilities of these departments and agencies are taken into account before licences are issued by the AECB if there is no conflict with the provisions of the AEC Act and its regulations.

After a licence is issued, the AECB carries out compliance inspections to ensure that its requirements are continually met. If the compliance inspection and assessment program identifies a noncompliance or an adverse trend that may eventually lead to a noncompliance, there is a range of possible actions that the AECB can take. These range from a recommendation for licensee action (for situations that should be improved but do not involve noncompliance) to prosecutions (for the more serious instances of noncompliance).

7.2 A SUMMARY OF THE LAWS, REGULATIONS AND REQUIREMENTS GOVERNING THE SAFETY OF NUCLEAR POWER STATIONS IN CANADA

At present, pending the coming into force of the new NSC Act, regulatory control over nuclear power stations in Canada comes primarily from the authority of these statutes and regulations:

- the AEC Act (see Attachment 7.1);
- regulations made pursuant to the AEC Act, for example:
    - Atomic Energy Control Regulations (see Attachment 7.2)
    - Physical Security Regulations (see Attachment 7.3)
    - Transport Packaging of Radioactive Materials Regulations (see Attachment 7.4)
    - Cost Recovery Fees Regulations (see Attachment 7.5)
- the Nuclear Liability Act (see Attachment 7.6).

The last major revision to the AEC Regulations came into effect in June 1974. These regulations describe the general conditions and requirements for the licensing of prescribed substances and nuclear power stations. Further amendments have been issued since that date, but they do not relate to the licensing of nuclear power stations.

Legal requirements are imposed on licensees by the AEC Act, the regulations made pursuant to that Act, and the licence conditions which the regulations empower the AECB to include in specific licences. Contravention of the act, regulations, or binding licence conditions constitutes an offence under the Act. Sanctions for offences include prosecution and revocation, suspension or amendment of the licence by the AECB.

For nuclear power stations, the regulations require that:

- A licence to operate be acquired from the Board (s. 9), with a prerequisite of Board approval to construct or acquire the facility (s. 10).
- General prescribed requirements are met when applying for and obtaining a licence (s. 9).
- Records are kept and occurrences are reported (s. 11).
- Radiation exposures are kept below the limits are set out in Schedule II.

As with most other countries, the radiation protection regulations are based upon the recommendations of the International Commission on Radiological Protection.

7.2.1 The New Act

On March 20, 1997, Bill C-23, the NSC Act (see Attachment 7.7) received Royal Assent. The new Act will replace the current AEC Act with a modern statute to provide for more explicit and effective regulation of nuclear energy.

The existing Act encompasses both the regulatory and developmental aspects of nuclear activities. However, the new Act will disconnect the two functions and provide a distinct identity to the regulatory agency. The new Act will replace the AECB with the Canadian Nuclear Safety Commission (CNSC). It clearly defines the CNSC’s separate role from that of AECL, which is the federal research, development, and marketing organization for nuclear energy.

Since the AEC Act was first adopted in 1946, the mandate of the regulatory agency has evolved from one chiefly concerned with national security to one that focuses primarily on:

- health
- safety
- environmental consequences

The new Act will provide the CNSC with a mandate to establish and enforce national standards in these areas. It will also provide a more explicit basis for implementing Canadian policy and fulfilling Canada’s obligations about non-proliferation of nuclear weapons.

The new Act will:

- permit an increase in the number of members of the Commission from five to seven to provide a broader range of expertise and will permit them to sit in panels;

- make the Commission a court of record with powers to hear witnesses, take evidence and control its proceedings, and at the same time, maintaining its flexibility to hold informal hearings;

- set out a formal system for review and appeal of decisions and orders made by the Commission, designated officers and inspectors;

- empower the Commission to require financial guarantees, to order remedial action in hazardous situations and to require responsible parties to bear the costs of decontamination and other remedial measures;

- bring the enforcement powers of compliance inspectors and the penalties for infractions into line with current Canadian legislative practices;

- bind the Crown, both federal and provincial, and the private sector;

- provide authority to incorporate provincial laws by reference and to delegate powers to the provinces in areas better regulated by them or where licensees would otherwise be subject to overlapping regulatory provisions;

- explicitly provide for the recovery of the costs of regulation from persons licensed under the Act.

The new Act will come into force when a suitable set of new regulations has been prepared. The AECB has consulted widely with the public, interest groups, licensees, and other stakeholders on the development of the new regulations. The AECB expects to have the new legislation proclaimed and all supporting documentation, including regulations, in place by the end of 1998 or by early 1999.

7.2.2 Nuclear Liability Act

The other substantive legislation enacted by Parliament specifically about nuclear energy is the Nuclear Liability Act (see Attachment 7.6). This Act, which entered into force in October 1976, places total responsibility for nuclear damage on the operator of a nuclear installation. It requires the operator to carry insurance in the amount of $75 million. It also provides for the establishment of a Nuclear Damage Claims Commission. This commission will deal with claims for compensation when the federal government deems that a special tribunal is necessary; for example, if the claims are likely to exceed $75 million. The Act recognizes that Canada may enter into international arrangements that carry nuclear liability, but at present, Canada is not a party to any such arrangement.

7.2.3 Canadian Environmental Assessment Act (CEAA)

Before 1984, the AECB was not directly concerned with environmental protection during site preparation and construction of nuclear power stations. Instead, the AECB would maintain contact with the provincial agencies responsible for environmental protection. This allowed the provincial agencies to identify any environmental issues that needed to be resolved as a prerequisite to AECB's approval. The AECB policy has been that it would not approve any site preparation or construction work until all these environmental issues were resolved.

In 1984, the federal government introduced an environmental review process that applied to proposals where:

- a federal government agency was the initiating department;
- there was an environmental effect on an area of federal responsibility;
- the federal government had a financial commitment;
- the proposal made use of lands administered by the Government of Canada.

Federal regulatory bodies such as the AECB were obliged to observe the process.

In 1995, this process was succeeded by the CEAA (see Attachment 7.8), which sets out responsibilities and procedures for the environmental assessment of projects involving the federal government. The Act applies to projects for which the federal government holds decision-making authority - whether as proponent, land administrator, source of funding, or regulator.

The majority of federal projects requiring an environment assessment will undergo either a screening or a comprehensive study. Both can be considered self-directed environment assessments in the sense that the Responsible Authority (RA) determines the scope of the environment assessment and the scope of the factors to be considered, directly manages the environment assessment process, and ensures the environment assessment report is prepared. The RA is the federal decision maker having responsibility under the Act. The AECB is a RA for projects that it regulates.

In practice, the project proponent may conduct the environment assessment, prepare the report, and design and implement mitigation measures and a follow-up program. The RA alone, however, remains directly responsible for ensuring that the screening or comprehensive study is carried out in compliance with the Act, and for deciding on the course of action with respect to the project following the screening or comprehensive study.

The CEAA requires that early on in the project a proponent carry out an integrated environment assessment of the possible impact of all licensing stages, before any irrevocable decisions are made.

7.2.4 Regulatory Documents

In addition to the various legally binding regulations issued pursuant to the AEC Act, the AECB issues documents on matters related to its regulatory mandate. These set out in AECB policies and standards on various matters or provide guidance to licensees on acceptable ways of complying with regulatory requirements. The guidance documents may be incorporated into the regulations of nuclear power stations as binding licence conditions. However, in most cases, licensees use guidance documents to develop their design and/or operating documents, and it is these licensee-produced documents that are incorporated into binding licence conditions.

The regulatory guidance documents include:

- R-7: Requirements for Containment Systems for CANDU Nuclear Power Plants (1991)
- R-8: Requirements for Shutdown Systems for CANDU Nuclear Power Plants (1991)
- R-9: Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants (1991)
- R-10: The Use of Two Shutdown Systems in Reactors (1977)
- R-77: Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems (1987)
- R-90: Policy on the Decommissioning of Nuclear Facilities (1988)
- R-99: Reporting Requirements for Operating Nuclear Power Facilities (1995)

See Attachment 7.9 for copies of these regulatory documents.

Regulatory documents R-7, R-8, R-9 and R-10 contain the principal safety standards that the AECB advises its licensees to meet for the special safety systems:

- the containment system
- the shutdown systems
- the emergency core cooling system

Regulatory document R-77 contains the standards for overpressure protection of the primary coolant system in CANDU reactors having two shutdown systems. It recognizes that the effectiveness of overpressure protection depends on the operation of the two shutdown systems and of the system's overpressure relief valves.

Regulatory document R-90 contains the AECB's policy on decommissioning. For new reactors, it advices that a conceptual decommissioning plan be prepared and submitted with the application for construction approval. This is to ensure that the need for eventual decommissioning of the facility was considered in the design. It is also intended to provide a basis for determining the amount of any financial guarantee that may be required to assure adequate financing for decommissioning of the facility.

Regulatory document R-99 contains the general reporting requirements for nuclear power stations. It is referred to in the operating licences, and it represents a minimum set of reporting requirements that all power reactor licensees must comply with. Additional reporting requirements may be imposed on individual licensees through specific licence conditions.

Consultative Document C-6 (Attachment 7.10), and C-6 Rev. 1 currently under preparation, contain the standards for safety analysis of CANDU nuclear power stations. Among other things, they set out the standards for identifying the initiating events to be analysed. For example C-6 Rev. 1 will:

- identify approximately 200 potential initiating events considered to be pertinent to the safety of CANDU nuclear power station;
- recommend a systematic review of the proposed plant by the licensee to identify any additional failures not contained in the general list. The output of the systematic plant review is a complete list of the postulated initiating events that must be analysed for the proposed design.

More about C-6, and C-6 Rev. 1, can be found in the Introduction section, part 4, under the title “The Analytical Technique”.

7.3 A DESCRIPTION OF THE LICENSING SYSTEM FOR NUCLEAR POWER STATIONS IN CANADA

The current AEC Regulations stipulate two formal licensing steps for nuclear power stations:

- construction approval (s. 10)
- operating licence (s. 9)

In practice, formal approval is also given for the site. Proposed regulations to be made under the new NSC Act would require a Site Preparation Licence.

The AECB’s licensing system is administered with the cooperation of federal and provincial government departments in areas such as:

- health
- environment
- transport
- labour

Regulatory control is also achieved by setting standards and guidelines for the licensees. Some are prepared within the AECB, and others are set by provincial authorities or national standards associations.

For all nuclear power stations, the five-member Board actually makes the decision to grant or not to grant a licence (or to authorize any conditions attached to a licence). A decision to issue or renew an operating licence normally requires at least two Board meetings to provide an opportunity for public input. The first meeting is for initial consideration of the application, and the second is for the decision. In making its decision, the Board considers the applicant's request, recommendations from the staff of the AECB, and any written or oral presentations from the public.

The specific stages of the licensing process are:

- site acceptance
- construction approval
- commissioning
- operating licence

7.3.1 Site Acceptance

At the site acceptance stage, the AECB must be assured that it is feasible to design, construct, and operate the facility on the proposed site so as to meet all safety and environmental protection requirements.

The AECB will not issue a site approval or Site Preparation Licence unless an environment assessment has been completed as required by the CEAA. If the environment assessment concludes that further investigation is needed, or if public concerns about the project warrant, the RA refers the project to the Minister of the Environment for a referral to mediation or a panel review. In the case of a comprehensive study, the Minister determines whether the project can be referred back to the RA for action or whether further investigation is required.

The AECB will also need to be assured that the site meets all safety requirements. The site affects safety in two ways:

- Site characteristics could affect the impact that radioactive releases have on the surrounding inhabitants. These can affect the expected dilution of any releases as well as the potential for concentration of radioactive materials in the food chain.

- Site characteristics that define the risk of external events that can affect the safe operation of the plant. These are events such as earthquakes, tornadoes, or external floods; as well as industrial and transportation accidents that may cause explosions, missiles, or toxic gas releases near the plant site.

Before approving a proposed site, the AECB requires the applicant to submit a Site Evaluation Report that includes a description of the design of the plant, and identifies and assesses the site characteristics that may be important to the safety of the proposed plant. These include:

- information on land use
- present population and predicted population expansion
- principal sources and movement of water
- water usage

- meteorological conditions
- seismology
- local geology

During this phase, the AECB requires that the applicant publicly announce its intention to construct the facility and to hold public information meetings where the public can express its views and question applicant officials.

Although a particular site may have some unfavourable characteristics, such as an unusually high population density or a higher-than-average risk of earthquakes, this does not necessarily make the site unacceptable. The site may be acceptable if the plant is designed to an appropriate standard. For example, the proximity of a railway line to the Darlington site was judged acceptable because the proponent undertook to design the plant to cope with the consequences of postulated railway accidents.

The main goal of the AECB at the site acceptance stage is to ensure that the site characteristics important to safety have been identified, and that the proponent recognizes that these characteristics must be accounted for in the design of the plant.

7.3.2 Construction Approval

Before it grants construction approval, the AECB must be assured that the site design will meet AECB safety requirements, and that the plant will be built to appropriate quality standards. Therefore, the design must be sufficiently advanced to enable safety analyses to be performed and their results assessed.

The first step is to identify the initiating events and event combinations that place the most severe demands on the safety systems. Generally this involved a combination of judgement, knowledge of the results of analyses of previous plants and the selected scoping analyses. The selected initiating events are then analysed in detail. These analyses are used to define the design requirements for safety systems.

The primary documentation required at the construction licence stage includes:

- a Preliminary Safety Analysis Report (PSAR) that combines the site information of the Site Evaluation Report, a description of the reference design including its major safety features, and the preliminary safety analyses showing the effectiveness of the proposed safety features;
- the reliability analyses of the special safety systems and other systems important to safety;
- a comprehensive commissioning program;
- a description of an overall quality assurance program for the project together with specific quality assurance programs for:
    - design,
    - procurement,
    - manufacture,
    - construction and installation,

- commissioning,
- preliminary plans for operation,
- conceptual plan for decommissioning the plant.

Construction will only be authorized after the design and safety analysis programs have progressed to the point that, in the judgement of the AECB, no major design changes will be required after the construction licence is issued. For systems not yet designed, the emphasis is on defining the major safety design requirements.

The AECB reviews the analysis of those postulated accidents that define the major design requirements for the plant's safety features. At the construction licence stage, the AECB demands analyses of enough postulated accidents in adequate detail in order to:

- ensure that all major safety design requirements have been identified;
- show that the reference dose limits can be met.

In particular, the applicant must be able to show that the standards for the special safety systems (shutdown systems, emergency core cooling systems, and containment system) will be met under all normal and upset conditions. These standards are defined in AEC Regulatory Documents R-7, R-8, R-9. In practice, this requires that the applicant considers most, if not all, of the postulated accidents identified in Consultative Document C-6. The applicant may be able to present reasons why a particular postulated event needs not be analysed in detail before the construction licence is approved. This could be because other analysed events place more stringent demands on the design of safety systems.

The AECB reviews the information in the PSAR and in supporting documents. The review concentrates on selected topics judged particularly important to safety to confirm that it forms an adequate basis for construction approval. In making this judgement, the AECB relies on experience from previous licensing reviews. For example, staff members rely on their experience to judge which accident cases are likely to define the major safety requirements for the plant and therefore require detailed analysis.

The AECB also takes account of any unusual or novel design features in deciding the topics that require in-depth examination. For example, in the licensing of Darlington, the AECB reviewed in detail the methods proposed to protect safety equipment from damage that might be caused by the breaking of large pipes. This was because the methods were different from those accepted in previous plants. The methods put less emphasis on physical pipe restraints. The applicant relied more on the argument that by careful design, material selection, and fabrication, pipes would crack and begin leaking long before a violent break would occur, if they were to fail. On completion of its review, the AECB agreed that the leak-before-break argument could be accepted. This decision applied to Darlington and would likely apply to any future nuclear plant if the same conditions were met.

In addition to reviewing the design and safety analysis information included in the application, the AECB also checks on the applicant's progress towards resolution of items outstanding from the site acceptance stage. The staff conclusions and recommendations from all of these reviews are documented in reports submitted to the Board who makes the final decision on approval of construction.

During construction of the plant, the AECB periodically audits activities important to safety. These audits are primarily intended to confirm that the licensee is complying with the quality assurance standards and procedures defined in the licence application. Such audits have concentrated on systems such as:

- primary coolant system
- special safety systems that are designed to prevent or mitigate the effects of serious accidents.

The reason for this emphasis is the particular importance of these systems to the defence-in-depth philosophy. The results of these audits are recorded in AECB assessment reports. The AECB has a formal documentation system to track the licensee's response and the final disposition of directives and actions arising from these audits.

7.3.3 Commissioning

Before commissioning takes place, at least one staff member of the AECB is located at the station to observe and report on the commissioning and start-up processes.

The AECB does not attempt to participate in all aspects of the licensee's commissioning program. Reliance is placed on the licensee's internal review process, which is mandated by the commissioning quality assurance plan. The AECB's direct involvement in commissioning concentrates on a few major tests that are considered particularly important to safety. A detailed description of the commissioning process is provided under Article 19.2.1.

7.3.4 Operating Licence

Before it issues an operating licence, the AECB must be assured that the construction of the plant conforms to the design submitted and approved, and that the plans for operation are satisfactory. The requirements include:

- submission of a Final Safety Report
- completion of a previously approved commissioning program
- AECB examination and authorization of the control room operators and shift supervisors
- AECB approval of the candidate to be appointed to the position of station manager, production manager, and senior health physicist
- AECB approval of operating policies and principles
- preparation of plans and procedures for dealing with radiation
- preparation of a specific program for quality assurance in operations

TABLE 7.1
PARTIAL LIST OF PREREQUISITES FOR AN OPERATING LICENCE

Subject	Prerequisites
Safety Analysis	▸ Final Safety Report ▸ Safety Analaysis Basis Documents ▸ Probabilistic Safety Analysis / Safety Design Matrices ▸ Reliability analysis of special safety systems ▸ Computer codes used in safety analysis
Safety Related Research	▸ Research results and reports
Design	▸ Design Manuals and Design Guides
Operation	▸ Operator Training Manuals and Procedures ▸ Operating Policies and Principles ▸ Operating manuals and system operating flowsheets ▸ Abnormal incident response manuals ▸ Safety system testing program ▸ Commissioning program
Quality Assurance	▸ Station Quality Assurance Program Commissioning and Operation ▸ Corporate Quality Assurance Program
Pressure-retaining Components	▸ System and component classification list ▸ Overpressure Protection Report ▸ Baseline and in-service inspection program
Radiation Protection	▸ Radiation Protection Regulations ▸ Radiation Protection Procedures ▸ Emission limits for radioactive materials ▸ Environmental Monitoring Program
Emergency Measures	▸ On-site emergency procedures ▸ Provincial Emergency Plan
Physical Security and Safeguards	▸ Station Security Plan and Procedures ▸ Safeguards Implementation Plan ▸ Facility Attachment
Decommissioning and Waste Management	▸ Waste and hazardous substance management and disposal procedures
Nuclear Liability Insurance	▸ Proof of coverage required by the Nuclear Liability Act

Table 7.1 provides a partial list of the prerequisites that a licensee must complete for an operating licence.

A provisional licence is issued to permit startup, to operate at low power levels and then to increase the power up to the design rating, subject to AECB approval. Provided all has proceeded satisfactorily, a full operating licence is then issued, usually for a term of two years. Among the terms of an operating licence is the requirement that the licensee informs the AECB promptly of any occurrence or situation that could alter the safety of the plant. Regulatory document R-99 (see Attachment 7.9) contains the standard reporting requirements included in the operating licences of Canadian nuclear power stations. The Board retains the right, by regulation, to impose additional conditions at any time.

Annex 7.1 provides a sample power reactor operating licence.

7.4 A DESCRIPTION OF THE SYSTEM OF REGULATORY INSPECTION AND ASSESSMENT OF NUCLEAR POWER STATIONS TO ASCERTAIN COMPLIANCE WITH APPLICABLE REGULATIONS AND LICENCES

Although the primary responsibility for the safe operation of the plant remains with the licensee, there is continued surveillance by the resident AECB inspectors. Pursuant to section 12 of the AEC Regulation, inspectors have the right not only to perform inspections, but also to direct the licensee to take certain remedial actions. The AECB also includes in its operating licences a condition requiring that operations, reports, tests, inspections, analyses, modifications, or procedural changes requested by the board are to be completed expeditiously.

In 1993, the AECB established a formal compliance inspection program. Its objective is to obtain information and evidence to demonstrate that reactor operation complies with the requirements of the AEC Regulations and the facility operating licence. Although the program is still under development, a core set of inspection activities is now being routinely carried out by the resident inspectors.

These inspections can be categorized into four broad types, listed in increasing depth and detail of inspection:

- Rounds: Performed on a routine basis by the resident AECB inspectors. This type of inspection examines system and equipment status on a plant area basis, for example:
    - reactor building at specific location
    - turbine hall
    - battery rooms
    - control equipment rooms
    - main control room

The objective is to gain an overall perspective of the status of the plant in the area examined, noting any obvious deficiencies or abnormalities. Rounds are conducted according to written check sheets which allow for the recording of the inspector's observations and recommendations for follow-up action. The sheets are dated, signed and retained on file at the project offices.

- Operating Practice Assessments: Examination of a particular aspect of station operation, for example:

  - security practices
  - effluent monitoring practices
  - fuel handling operation
  - unit startup
  - configuration control during outages

  The staff members who conduct the inspection are chosen based on the area being assessed. They could involve specialists from head office, project officers from the site office, or a combination of the two. Operating practice assessments are usually done according to preplanned inspection guides prepared for the specific occasion. Results are normally recorded in an AECB report which is sent to the licensee for follow-up action as necessary and retained on file.

- Audits: Formal, in-depth and detailed examination of one or more topics relating to a specific aspect of plant operation, for example:

  - quality assurance programs
  - periodic inspection programs
  - health physics programs
  - change control programs

  These inspections are always preplanned to a high degree of detail with the acceptance criteria spelled out in advance. The licensee is notified in advance of the audit and its subject area. Entrance meetings, daily briefings of audit results, and exit meetings are included in audit plans. The results are recorded in an AECB report to the licensee and follow-up actions are recorded and assigned target dates for completion.

- System Inspections: In-depth and detailed examination of the status of a chosen plant system, for example:

  - shutdown system one
  - reactor regulating system
  - class III power
  - containment

These inspections are normally done by the resident AECB inspectors according to preplanned written check sheets. Results are generally transmitted formally to the licensee by letter, and if necessary, follow-up actions with target dates are spelled out.

The inspections, evaluations and audits performed by the AECB draw on the best international and domestic practices and standards. To strengthen these reviews, over the last several years, the AECB has been developing a set of safety performance indicators to be used to benchmark acceptable levels of operational safety. The indicators are devised to allow tracking of operational trends important to safety and to compare the performance of stations with each other. The performance indicators are used to assess, summarize and report on the performance of licensees about safety. These indicators have been in use, for a one-year trial period, since January 1998. Following this trial period, the safety performance indicators will be codified in a regulatory standard.

The performance indicators will be used in conjunction with other information gathered by the AECB. The overall regulatory safety assessment process includes the conclusions drawn from the performance indicators, from event analysis, and from inspections/investigations. Conclusions drawn from these three elements, taken alone or in combination, may result in additional regulatory inspections.

The AECB's compliance program is one of the regulatory activities through which the safety of licensees' operations is assessed. It is the part of the regulatory program that determines whether or not a licensee is complying with AECB requirements through:

- observation
- examination and auditing of physical plant status
- plant operating practices and managed programs for its operation

Compliance program activities also inform the licensees of the requirements for the safe operation of the nuclear power facilities. Finally, through the program, the AECB can request from, or impose upon, a licensee corrective action when those requirements are not met.

The AECB compliance program policy requires written programs that:

- promote compliance through consultation, communication, advice and technical guidance;
- verify compliance through monitoring, inspections, audits and investigations;
- enforce compliance through graduated deterrent measures such as licensing actions, directives, and prosecutions.

To verify compliance with the regulatory requirements, the AECB:

- evaluates the licensee's operations and activities;
- reviews, verifies and evaluates information supplied by licensees;
- ensures that administrative controls are in place;

- evaluates the licensee’s remedial action and the action taken to avoid incidents in the future;
- examines licence conditions for evidence that suitable licensing action could avert similar incident.

The principal objective is to enforce compliance effectively by ensuring the correction of violations and deterring future violations. The program also aims at preventing violations by detecting and responding to adverse trends that, if uncorrected, might eventually lead to violations. The actions of the AECB to meet this objective are designed to help protect the environment, and the health and safety of workers and the public.

7.5 A DESCRIPTION OF THE PROCESS OF ENFORCEMENT OF REGULATIONS AND CONDITIONS OF LICENCES USED IN CANADA

There is a range of possible actions that the AECB can take in the event of noncompliance including:

- recommendations for licensee action
- action notice
- directive
- licence amendments
- restricted reactor operation or reactor shutdown (In most circumstances, if a shutdown is appropriate, the AECB expects that a licensee would initiate the shutdown on its own before it became necessary for the AECB to order it to do so.)
- revocation or suspension of the licence
- prosecutions

See Annex 7.2 for some of the major design and operational changes resulting from AECB actions.

A licensee who is subject to enforcement action is entitled to request a hearing before the Board to contest the action taken by AECB staff. For an amendment, suspension or revocation of the licence, the licensee would normally receive advance notice and have an opportunity to be heard by the Board before the action is taken. The AEC Regulations give the Board the authority to suspend a licence without prior notice, where it is necessary to do so in the interests of health, safety, and security. In this case, the licensee may request the Board to hold an inquiry into the reasons for the suspension.

The following are some examples of specific instances of noncompliance, the severity of which would normally lead to prosecution:

- exposures to the public or workers in excess of the dose or exposure limits
- failure to take all reasonable measures to comply with an inspector’s directive

Licence sanctions recommended by an inspector can be imposed by the AECB without a court proceeding. It is important to note that sanctions are intended as a deterrent to prevent future violations. Licence sanctions may include:

- Short Term Licence or Extension: If the AECB is not satisfied that a licensee has the required commitment to safety, as indicated by the current violation and compliance history, the staff may recommend to the Board that a licence be granted for a shorter term. Alternatively, a short term extension may be granted to allow sufficient time for the licensee to make the required improvements before the licence is considered for renewal.

- Licence Amendment: AECB staff may recommend a licence amendment to the Board. The licensee is notified in writing of the proposed action and is given an opportunity to be heard by the Board. Licence amendments cover a wide range of possibilities and are decided on a case-by-case basis. Examples of licence amendments include:

    - limitations to power operation
    - a requirement to obtain AECB approval before reactor startup
    - a requirement to appear before the Board on a regular basis to provide status reports on progress in improvements to operation and maintenance programs

- Licence Suspension, Revocation, or Non-renewal: AECB staff may recommend a licence suspension, revocation, or non-renewal to the Board. The licensee is notified in writing of the proposed action and is given an opportunity to be heard by the Board. This course of action can be taken in any of the following circumstances:

    - licensee is in serious noncompliance
    - licensee has been successfully prosecuted
    - licensee has a history of noncompliance
    - AECB has lost confidence in the licensee's ability to comply with the regulatory requirements

In addition to the AECB actions described in Annex 7.2, the AECB has restricted Bruce A to a one-year licence period in 1988 and 1989. It has also restricted Pickering to six-month and nine-month licence periods in 1996 and 1997. Both stations were required to report on the progress they have made for improving standards. In response, the licensee provided additional resources and manpower, and instituted a quality improvement program.

ARTICLE 8
REGULATORY BODY

Canada has established a regulatory body about fifty years ago. The history and development of the Canadian legislative and regulatory framework, and the creation of the Atomic Energy Control Board (AECB), are described in Article 7.1.

8.1 A DESCRIPTION OF THE MANDATE AND DUTIES OF THE AECB

The mandate of the AECB is to regulate the development, production, and use of nuclear energy in Canada in a manner that prevents unreasonable risk to:

- health
- safety
- security
- the environment

Over the years, the AECB has used its powers to supplement the legislative and regulatory framework. The main AECB regulatory activities for operating power stations include:

- operating licence renewal
- compliance activities
- change approvals

This is in addition to the licensing activities for new power stations as described in Article 7.3.

8.1.1 Operating Licence Renewal

The AECB's safety review process focuses on obtaining assurance that the risk to the public and employee health and safety, and to the environment remains within the bounds of the original licensing basis for the facility. This review process covers all areas of AECB regulatory requirements and accommodates a two-year licence renewal cycle. AECB staff, on-site and at head office, continuously monitor the operation of the reactors and the licensee's compliance with safety and licensing requirements.

The safety of operating nuclear power plants is reviewed for compliance with the following:

- requirements of the Atomic Energy Control (AEC) Regulations
- relevant regulatory documents
- industry codes and standards
- the facility operating licence
- pertinent station policies and procedures

The review process consists of the following:

- annual AECB staff review of station safety performance
- compliance inspection program
- review of significant events reported by the licensee
- review and approval of proposed temporary and permanent changes
- management of generic safety issues
- safety analysis reviews
- design change and equipment performance reviews
- reliability and risk assessment
- human factors assessment
- pressure-retaining component surveillance/assessment
- quality assurance reviews
- radiation protection program and environmental impact reviews
- operator certification and training program assessments

Although the AECB continuously reviews and monitors reactor operations, there is a need for a formal periodic review of plant operation. The AECB conducts such a review annually. This includes a review of the information contained in the licensee's annual report as well as that in various event reports submitted throughout the year. It also takes into consideration the licensee's responses to the AECB's requests concerning generic safety issues, and concerning the results of AECB's monitoring of station.

The AECB's review of each station's operation is summarized in an annual report prepared for the Board and for subsequent release to the public. This report is an important input to the two-year licence renewal process. The areas that it covers include:

- employee and public radiation safety
- safety system performance
- operations and maintenance
- station management
- training
- emergency preparedness
- safety analysis
- quality assurance
- nonproliferation and safeguards

A feature of the reports is the compilation of a number of assessments in the above areas. Performance indicators give the AECB early warning of any problems in station operation that might lead to a deterioration in the level of safety. AECB assessments have led to providing restricted licences to Bruce A and Pickering A and B stations in 1988 and 1996 respectively.

In addition to the AECB's annual assessment reports, the AECB staff prepares a Board Member Document before licence renewal. This document addresses in detail all issues of licensing significance and assesses the performance of the licensee in those areas over the previous licensing period. The AECB's licensing process encourages public review and comment on licensing issues during the period between the Board's initial consideration of an application for a licence and the final Board decision, which is approximately two months.

Each year, a formal review meeting is held between the AECB and licensee management. This meeting is in addition to the many formal and informal meetings held with licensee staff on specific topics. The review meeting's main purpose is to discuss with the licensee the AECB's "report card" on the nuclear power station and progress on high-profile safety issues.

8.1.2 Compliance Activities

The AECB maintains staff at each of the power reactor stations to monitor licensee compliance with the AEC Regulations and licences issued by the Board. A total of 27 engineers and scientists are posted on a full-time basis at reactor sites to inspect and ensure safety during:

- construction
- commissioning
- operation
- reactor maintenance

They also investigate unusual events at the reactors.

See Article 7.4 for a description of regulatory inspections and other functions of the AECB site staff.

The AECB, in addition, has a number of specialists at its headquarters in Ottawa. In cooperation with the site staff, these specialists review and verify the quality and reliability of key reactor components and provisions, such as:

- design
- construction
- commissioning
- safety analyses
- radiation protection
- performance
- adequate safety procedures
- management of the facilities

Head office staff also coordinates the review and resolution of generic safety issues, and codifies AECB regulatory requirements.

As stated in Article 7.4, in 1993, the AECB established a formal compliance inspection program. Its objective is to ensure that reactor operation complies with the requirements of the AEC Regulations and the facility operating licence. Although the program is still under development, a core set of thirty-one inspection activities is now routinely carried out. Procedures have been defined for each type of inspection, including periodic reviews of findings, and inspection checklists have been standardized. Resource requirements from resident site offices and from head office specialists have been estimated. A policy that governs implementation of the program has been adopted.

Although the project was aimed at the establishment of a pro-active inspection program to verify a consistent depth and coverage of each station's compliance with regulatory requirements, it also recognized the need for reactive inspections. These inspections are prompted by such things as core inspection findings or events. They are discretionary but important, and complement the core program. Whereas the core inspections are broad in nature, reactive inspections usually have a deeper focus.

Over the last few years, the AECB has moved to perform more integrated assessments of licensee performance. In January 1998, a Power Reactor Evaluation Division was created with responsibility to develop the standards and capabilities required to carry out this work.

8.1.3 Change Approvals

The safety of a reactor depends on the nuclear power station meeting its safety design objectives on a continuing basis. To ensure that this is achieved, the AECB requires that a licensee has controls in place for temporary and permanent changes to equipment, procedures, and documentation. These controls must make sure that all changes are subject to an appropriate level of safety review, and are authorized by a responsible member of the licensee's staff. Important changes require specific approvals by the AECB, and these requirements are enforced by the following licence conditions:

- Primary licence documentation: Specific AECB approval is required for changes to licence documents (for example, Operating Policies and Principles, and radiation emergency programs). This is contained in the licence conditions specifying the requirement for each document.

- Changes to special safety system equipment: Specific AECB approval is required for changes to the shutdown systems, containment, or the emergency core cooling system; and the licensee is required to maintain the trip setpoints of the shutdown systems at values approved by the AECB.

- Changes to equipment or procedures that could affect safety of the nuclear power station: Specific AECB approval is required for any change to the reactor equipment or to the utility's procedures that could cause a hazard different from those considered in the nuclear power station's licensing basis.

Most AECB approvals for changes are given by the resident inspectors. Temporary changes are frequently applied to the special safety systems to allow maintenance work to be carried out, particularly during unit outages. Most permanent design changes and major procedural changes are reviewed by specialist at AECB head office before approval is given.

The safety case for a reactor is not static. Advances in the techniques available for safety analysis, results of research, and changes to the plant equipment and operating procedures make an ongoing process of safety evaluation necessary. For this reason, the AECB requires licensees to review and update their facility's Safety Report at least once every three years.

8.2 A DESCRIPTION OF THE AUTHORITY AND RESPONSIBILITIES OF THE AECB

The authority of the AECB under the AEC Act and its regulations is described in Article 7.1. The act and regulations are described in Article 7.2.

The AECB's control of radioactive materials extends to the import and export of nuclear items, and covers domestic and international security of nuclear materials, equipment, and technology. Accordingly, the AECB participates in activities of the International Atomic Energy Agency to ensure that Canada complies with the requirements of the Treaty on the Non-Proliferation of Nuclear Weapons (N.P.T.), and with other bilateral and multilateral agreements.

In the near future, the Canadian Nuclear Safety Commission (CNSC) will be a court of record with powers to hear witnesses, to take evidence, and to control its proceedings. At the same time, it reserves the right to hold informal hearings. It will be empowered to require financial guarantees from licensees, to order remedial action in hazardous situations, and to require responsible parties to bear the costs of decontamination and other remedial measures. The authority and responsibilities of CNSC are specified under section 9 of the new Nuclear Safety and Control (NSC) Act as follows:

- to regulate the development, production and use of nuclear energy and the production, possession, and use of nuclear substances, prescribed equipment, and prescribed information in order to:
  - prevent unreasonable risk to the environment and to the health and safety of persons associated with that development, production, possession, or use;
  - prevent unreasonable risk to national security associated with that development, production, possession, or use;
  - achieve conformity with measures of control and international obligations to which Canada has agreed.
- to disseminate objective, scientific, technical, and regulatory information to the public concerning the activities of the Commission and the effects, on the environment and on the health and safety of persons, of the development, production, possession, and use referred to in the first paragraph.

The new Act provides the CNSC with the authority to:

- set and enforce national standards in the areas of health, safety, and environmental protection related to nuclear energy;
- establish a clearer basis for implementing Canada's policies and obligations about the non-proliferation of nuclear weapons.

More about the new Act can be found in Article 7.2.1.

Despite these desirable legislative changes, the AECB does not intend to change its nuclear regulatory philosophy and approach when the NSC Act and regulations come into effect.

8.3 STRUCTURE OF THE AECB, AND ITS HUMAN AND FINANCIAL RESOURCES

The AECB consists of a President, a federally appointed Board, and staff hired by the Board. This general structure is defined by the following existing legislation:

- The AEC Act establishes a five-member Board, consisting of the President of the National Research Council of Canada and four members appointed by the federal government through the Governor-in-Council (Cabinet).

- The AEC Act further stipulates that one of these members shall be appointed by the Governor-in-Council to be the President and chief executive officer of the Board. Sub-section 5(2) states that the President "has supervision over and direction of the work of the Board, and of the officers, technical and otherwise, employed for the purpose of carrying on the work of the Board".

The Board functions as a regulatory and quasi-judicial decision-making body. It makes licensing decisions for major nuclear facilities and sets policy direction on health, safety, security, and environmental issues that concern the nuclear industry and the public.

The Board usually meets nine or ten times a year to deal with matters not delegated to its staff. Board meetings are held at AECB headquarters in Ottawa, or at locations convenient to the site of AECB-licensed facilities or activities.

AECB has approximately 400 employees including:

- administrators
- financial officers
- auditors
- scientists
- engineers
- chemists
- biologists
- mathematicians
- health and nuclear physicists
- accountants
- technicians
- electronic data processing experts
- safeguards experts
- security experts
- information processing and management specialists
- support staff
- maintenance personnel

This is in addition to other specialists in a wide variety of fields and disciplines essential to effective discharge of the Board's responsibilities and daily operation.

See Annex 8.1 for an AECB staff organization chart.

AECB also obtains services from external sources where it requires special expertise, mainly through the Research and Support program. The program provides access to independent advice, expertise, experience, information, and other resources via contracts placed in the private sector and with other agencies and organizations in Canada as well as in other countries. During Fiscal Year 1997/98, a total of $2.109M was spent on research and support work. A total of 117 projects were active during the year covering the areas of:

- environmental impact assessment and management;
- safety-related control and electrical systems;
- seismologic studies;
- pressure boundary integrity;
- integrity of containment and safety-related structures;
- human factors;
- internal dosimetry;
- health effects in human populations;
- physics and fuel studies;
- probabilistic safety assessment;
- emergency preparedness;
- radio-biology and external dosimetry.

Subject to federal policies and applicable legislation, AECB employees are hired by the Board to perform assigned duties. Under the supervision of the President, they perform various tasks that are essential to the functioning of the AECB, and to the effective discharge of the Board's responsibilities.

The tasks of AECB staff are to:

- evaluate and process applications for AECB licences;
- develop and prepare licensing recommendations;
- administer AECB policies and procedures;
- maintain records;
- monitor, audit, and inspect nuclear facilities and activities;
- draft and administer licences;
- evaluate the qualifications and performance of licensees and their staff;
- prepare documents and reports;
- review reports and records;
- develop and enforce regulatory standards and requirements;
- assist the AECB in discharging its mandate to disseminate objective information regarding nuclear energy.

The job-related performance of all AECB staff is formally evaluated each year in accordance with AECB administrative policies and procedures. In return for continuing employment, AECB staff must maintain and demonstrate required skills and satisfactory performance.

In addition to the support provided by AECB staff and external sources, the President and the Board receive specialist advice on radiological protection, nuclear safety, and medical matters from Advisory Committees. Legal advice is provided by counsel employed by the federal Department of Justice. In addition, federal policies on regulatory fairness and public consultation, as well as provisions of nuclear legislation and the Board's policies on appeals and representations, allow licensees and the public the right and opportunity to be heard on nuclear matters of concern.

The advisors to the President and the Board include:

- Audit and Evaluation Group: Helps management ensure that the AECB functions efficiently and effectively. The group is directly accountable to the President. It evaluates management and regulatory programs, frameworks, and activities. It also audits performances and results, and reports upon its findings and conclusions.

- Legal Services Unit: Assigned by the federal Department of Justice to provide legal support and advice to the AECB.

- Advisory Committee on Radiological Protection (ACRP): Advises the President and the Board on generic matters within their terms of reference.

- Advisory Committee on Nuclear Safety (ACNS): Assisted by support staff who is retained by the AECB.

- Group of Medical Advisors (GMA): The GMA is made up of medical professionals who have been appointed Medical Advisers to the Board under the AEC Regulations. These appointees are nominees of the provinces, Atomic Energy of Canada Limited (AECL), the Department of National Defence (DND), and Health Canada. GMA is assisted by support staff who is retained by the AECB.

The AECB is funded by Parliamentary appropriations. Its total expenditure for the fiscal year ending March 31, 1997 was $49,774,188. During this period, the AECB collected $30,072,647, or approximately 60% of its total expenditure, through fees charged for licences and permits. See Attachment 7.5 for a copy of the "Cost Recovery Fees Regulations". The funds recovered are credited directly to the Consolidated Revenue Fund (treasury) of the federal government.

8.4 POSITION OF THE AECB IN THE GOVERNMENT STRUCTURE

The AECB is a departmental corporation, named in Schedule II of the Financial Administration Act. The AEC Act stipulates that the AECB shall report to the Parliament of Canada through a member of the Privy Council for Canada (Cabinet) designated by the Governor-in-Council as the Minister for purposes of the Act. Currently, this designate is the Minister of Natural Resources Canada.

The AEC Act requires the Board to "comply with any general or special direction given by the Minister with reference to the carrying out of its purposes". However, it is an accepted constitutional convention in Canada that any political directives given to agencies such as the AECB are of a general nature and cannot interfere with Board decisions in specific cases.

In practice, the AECB functions as an independent, quasi-judicial decision-making body on a day-to-day basis. It makes licensing decisions for nuclear facilities and sets policy direction on health, safety, security, and environmental issues that concern the nuclear industry and the public.

The Board requires the involvement and support of the Minister for special initiatives, such as the introduction of new legislation. For example, the NSC Act was sponsored by the former Minister of Natural Resources on behalf of the Board.

The AECB is not part of Natural Resources Canada (NRCan) nor is it accountable to staff or executives of the department other than the Minister. However AECB staff routinely interacts with management and staff of NRCan in areas of mutual interest. NRCan has a general interest in various matters relating to nuclear energy and natural resources. For example, the department funds the cleanup of certain low-level radioactive wastes on behalf of the Government of Canada, and consequently has an interest in related AECB policies and licensing matters.

In keeping with federal policies on public consultation and regulatory fairness, the AECB routinely consults with parties and organizations that have an interest in its regulatory activities. These include:

- AECB licensees;
- the nuclear industry;
- federal, provincial and municipal departments and agencies;
- special interest groups;
- individual members of the public.

As required by federal policies on Access to Information, formal consultations are conducted in an open and transparent manner.

AECB licensees include several publicly-funded institutions or agents of the federal and provincial governments. These include:

- Atomic Energy of Canada Limited (the federal nuclear research and development company);
- the following nuclear operations of provincially owned electrical utilities: Ontario Hydro Nuclear, New Brunswick Power, and Hydro-Québec;
- Canadian universities;
- hospitals and research institutions.

The AECB licenses and regulates the nuclear activities of these organizations in the same manner and to the same standards required from privately-owned companies or operations.

8.5 RELATIONSHIP OF AECB TO BODIES RESPONSIBLE FOR PROMOTION AND UTILIZATION OF NUCLEAR ENERGY

The Minister of Natural Resources is responsible for both of AECB and AECL. The AEC Act specifies that certain federal "companies", including AECL, are bound by the act and regulations thereunder. Consequently, there is a legal separation between AECB's regulatory role and AECL's promotional role, and the latter is subject to the former. The new NSC Act will give the national regulatory agency a more distinct identity, emphasizing the difference between its mandate and that of AECL (see Article 7.2.1 for more information on the new NSC Act).

Two Canadian organizations associated with the utilization of nuclear energy are the Canadian Nuclear Association (CNA) and the Canadian Nuclear Society (CNS). The AECB has no formal connections of any kind to these organizations, whose objectives and mandate are described below.

8.5.1 The Canadian Nuclear Association (CNA)

Established in 1960, the CNA is a nonprofit voluntary membership body that promotes the orderly and sound development of nuclear energy in Canada and abroad for purposes other than those related to armament. The Association represents many industries and enterprises sharing a common interest in the development and application of nuclear energy for peaceful purposes. Members include:

- uranium producers
- reactor manufacturers
- electrical utilities
- engineering companies
- banks
- employee unions
- departments of federal and provincial governments
- educational establishments

The CNA was established to:

- create and foster an environment favourable to the healthy growth of sound applications for nuclear energy and radioisotopes;
- encourage cooperation between various industries' utilities, educational institutions, government departments and agencies, and other authoritative bodies that have a common interest in the development of economic uses for nuclear power and radioisotopes;
- provide a forum for the discussion and resolution of problems that are of concern to the members, the industry, or the Canadian public in general;
- stimulate cooperation with other associations with similar objectives and purposes.

The main services offered by the Association to its members are:

- gathering and circulating authoritative and timely information about all aspects of nuclear energy;
- presenting its members' viewpoints to public inquires, regulations, governments, etc.;
- helping foster an accurate public understanding of the uses of nuclear energy for peaceful purposes;
- providing analysis of opportunities and problems relating to the development and use of nuclear energy;
- sponsoring activities about the successful marketing of nuclear-related products and services in Canada and around the world;
- sponsoring conferences, seminars, and courses on various subjects about the development and use of nuclear energy, uranium production, etc.;
- establishing and maintaining the close liaison with other associations having related interests.

8.5.2 The Canadian Nuclear Society (CNS)

The CNS is primarily a technical society of individuals involved in, or associated with, the Canadian nuclear program. It was originally established in 1979 and became incorporated under federal law in 1998. The major activities of the CNS are directed at the exchange of information related to nuclear science and technology through conferences, specialized meetings, and publications. Subject matter goes beyond nuclear power to include areas such as waste management, use of radioisotopes, nuclear education, etc.

The primary category of CNS membership is that of an individual directly involved in the use or development of a nuclear technology in any of the above areas or an individual who is simply interested in nuclear technology. Another category of CNS membership is that of an educational institution, such as school, university, or public library, that has an interest in providing timely information on nuclear science and technology to a student body or to the public at large. This type of membership has all the privileges of an individual membership with the exception of voting rights.

The objectives of the CNS are:

- to act as a forum for the exchange of information about nuclear science and technology;
- to foster the development and beneficial utilization of nuclear science and technology for peaceful uses;
- to encourage education in, and knowledge about, nuclear science and technology;
- to enhance the professional and technical capabilities of those involved in nuclear science and technology in the Canadian context.

ARTICLE 9
RESPONSIBILITY OF THE LICENCE HOLDER

As stated in earlier articles, the prime responsibility for safety of a Canadian nuclear power station rests with the licensee. The following sections will identify the main responsibilities and activities of the licence holder related to safety, and the mechanism by which the Atomic Energy Control Board (AECB) makes sure that the licensee meets its responsibility.

9.1 DESCRIPTION OF THE MAIN RESPONSIBILITIES AND ACTIVITIES OF THE LICENCE HOLDER RELATED TO SAFETY ENHANCEMENT

9.1.1 Main Responsibilities of the Licence Holder Related to Safety Enhancement

The main responsibilities of a Licence Holder related to the safety enhancement of the nuclear power station can be expressed as follows:

- ensuring that the responsibilities of the operating organization are defined and implemented;
- establishing and nurturing a safety culture as an integral part of a comprehensive management framework.

DEFINING AND IMPLEMENTING RESPONSIBILITIES
The overall responsibilities for nuclear safety are defined in policies and implemented by procedures. Adherence to these procedures is demanded by the line organization and verified by internal assessments and reviews.

The day-to-day responsibility for nuclear safety rests with the organization operating the facility. During the operational phase, the operating organization is in complete charge of the plant, with full accountability and authority for approved activities in the safe production of electric power. Because such activities directly affect nuclear safety, the operating organization is responsible for:

- establishing the policy for adherence to safety requirements
- establishing procedures for safe control of the plant under all conditions
- making sure that adequate resources and facilities are available at all times for both planned activities and contingencies

The above fundamental responsibilities are implemented through a specific, and explicit, set of responsibilities including those provided below. The associated activities are usually implemented through a hierarchical system of procedures, worker training, and the corporate safety culture. Each operating organization is responsible for:

- making sure that the licensed activity is carried out according to the requirements of relevant acts, regulations, and licences;
- making sure that any nuclear power station, nuclear substance, prescribed equipment, or prescribed information encompassed by the licence meets the provisions of relevant laws, regulations, and requirements;
- making sure there is a sufficient number of qualified workers to carry on the licensed activity safely and in accordance with relevant laws, regulations, and requirements;
- taking all reasonable precautions to protect the environment, the health and safety of persons, and the maintenance of security;
- providing and maintaining, within the manufacturer's specifications, the devices (as required under relevant laws, regulations, and requirements or the licence) for detecting and measuring radiation, radioactive nuclear substances, or hazardous substances encompassed by the licence at the site of the licensed activity;
- requiring that any person on the site of the licensed activity use the devices, clothing, equipment, and procedures in accordance with relevant laws, regulations, and requirements of the licence;
- taking all reasonable precautions to control the release of radioactive nuclear substances or hazardous substances from the site of the licensed activity into the environment;
- implementing measures to alert it to the illegal use or removal of a nuclear substance, prescribed equipment or prescribed information, or illegal use of a nuclear power station;
- implementing measures to alert it to an act of sabotage or attempted sabotage anywhere at the site of the licensed activity;
- making sure that there is compliance with all applicable safeguard agreements;
- instructing the workers on the physical security program at the site of the licensed activity and on their obligations under that program;
- ensuring that radiation protection of the public and employees shall be according to the Radiation Protection Regulations. Radioactive emission shall be kept below the levels stated therein and consistent with the principle of As Low As Reasonably Achievable (ALARA);
- minimizing the probability and potential consequences of nuclear power station accidents through conservative operation and decision making in the operation of the facility. Conservative operation and decision-making means prudent decisions in the direction of improved safety, and are accomplished by maintaining station operation within the bounds defined by the reactor licensing documents (see safe operating envelope in Article 19.2.2).

NURTURING SAFETY CULTURE
The term “safety culture” usually refers to the dedication and accountability of all individuals engaged in activities that affect nuclear safety. This culture must originate at the highest level of the organization and be implemented by:

- clearly defining the duties arising from the responsibility of operating a nuclear power station, and making sure that they are exercised by an unbroken delegation from the highest governing level of the licensee throughout the managerial chain to the individual employee at each facility;

- providing and maintaining the resources necessary to safeguard the health and safety of the workers, the public, and the environment from radiological hazards.

See Article 10.2.2 for more about the establishment of a safety culture.

9.1.2 Main Activities of the Licence Holder Related to Safety Enhancement

The main activities of the licence holder related to safety enhancement include the following:

DESIGN CONFIGURATION CONTROL
A configuration control process for the existing design includes:

- the design audit process (vertical/horizontal slice);
- the process of periodically making sure that the system is designed to meet relevant standards and has not been inadvertently modified outside its specifications (for example, by addition, removal, or replacement of components; or other changes to its operating environment);
- the process of periodically making sure that the system is being maintained, inspected, tested and operated to make sure it stays within the required specifications.

To make sure design and safety requirements are identified and considered in the design process, procedures that govern design control and modification include provisions to identify and document design interfaces for each design modification. Also, design changes are assigned a Design Team Leader who is responsible for making sure that design interfaces, design requirements, and nuclear safety requirements are documented, addressed, and reviewed; and that interfaces between design disciplines are maintained.

To ensure the timeliness of response to design issues that involve safety improvements, engineering change requests are screened for safety significance. A priority is also placed on them commensurate with their importance to nuclear safety. Engineering changes involving nuclear safety significant design improvements will be handled in a prompt conservative manner that ensures protection of station personnel, equipment, and the general public; and maintenance of defence-in-depth.

The design control and modification process fully complies with Canadian Standards Association (CSA) N286.2, “Design Quality Assurance for Nuclear Power Plants.” Periodic audits of governing documents for the design control and modification process are performed to verify that processes contained within these documents comply with CSA N286.2. Audit programs have also been established at the generating stations to make sure design activities, functions, and deliverables comply with CSA N286.2.4.

Design control and modification activities are governed by a common series of corporate level documents to make sure nuclear safety and acceptable standards of quality are achieved. These documents have been developed to incorporate the best industry practices for nuclear design. At OHN, procedures directing the design change and modification process fully comply with the recommendations of:

- Electric Power Research Institute (EPRI) TR103586, “Guidelines for Optimizing the Engineering Change Process for Nuclear Power Plants”
- NP6406, “Guidelines for the Technical Evaluation of Replacement Items in Nuclear Power Plants”

These procedures will also incorporate the following Institute for Nuclear Power Operations (INPO) recommendations:

- INPO 90009, “Guidelines for the Conduct of Design Engineering”
- AP905, “Configuration Change Process Description”
- AP906, “Design Change Process Description”
- TS402, “Plant Modification Control Program”
- TS412, “Temporary Modification Control”

ANALYSIS REVIEWS

Nuclear Safety Analysis reviews are undertaken periodically to:

- account for utility operating experience;
- account for improved analytical techniques;
- incorporate new information arising from research findings.

Canadian practice requires that the safety analysis for each operating station be reviewed and updated, and the Safety Report resubmitted to the regulator once every three years, or at another agreed frequency.

Safety analysis activities are governed by utility procedures, that are different for different utilities, to make sure that:

- the likelihood of errors and/or omissions in safety analyses is minimal and continually reduced;
- analysis work is subject to review and verification;
- analysis results are prepared in a manner that can be reviewed by an independent reviewer;
- analysis methods and results can be repeated independently.

Procedures controlling the nuclear safety analysis program include provisions for peer review and verification. Peer reviews are independent reviews performed by knowledgeable reviewers who are able to apply appropriate experience and technical judgment to the analysis evaluation. Peer reviewers can be outside consultants, or members of the same organization working in groups different from the group that performed the analysis. Peer review is required for all nuclear safety analysis work and includes an objective assessment of the overall correctness and technical standard of the work.

Safety analysis results are also subject to verification. Depending upon the complexity of an analysis, the verification may consider different aspects such as:

- application of specified methodology and assumptions
- assessment of results by comparison with hand calculations and other known bench-mark solutions
- examining repeatability of results using different software (computer codes)

The following elements of analysis are also verified:

- correct use of input data defining initial and boundary conditions
- correct use of data transferred from other sources
- correct documentation of results

RELIABILITY STUDIES

Safety systems must be able to reliably meet the performance requirements set by the safety analyses. For special safety systems (shutdown systems, emergency core cooling, and containment), an analysis of each system's reliability is required. This analysis is to show that the system can exceed a licensing target of 0.999. Testing of the system must be performed with sufficient frequency to show that the design reliability is being maintained.

At the end of each year, the licensee of each power station is required to submit a reliability report. This report contains an evaluation of the system reliability of each special safety system and safety-related system that has a reliability requirement. This is part of the reporting requirements contained in the Regulatory Document R-99 (see Attachment 7.9, also see Article 9.2). The annual reliability report must document:

- the completion of all required tests;
- an assessment of the ability of the system to meet its reliability requirements during any impairments that have occurred during the year;
- a review of the system reliability performance indices;
- an assessment of the predicted reliability.

The annual assessment of predicted reliability must include a review of all differences between the actual system status and the information used in the current reliability analysis. The review must take into consideration:

- differences between the actual design and the model
- differences between the actual operating or maintenance procedures and those assumed in the analysis
- differences between the actual component and system performances and those assumed in the analysis
- any discovery of new failure modes or failure trends
- any discovered differences between actual component failure rates and those in the model

If the review indicates differences that would invalidate the results of the reliability analysis, the analysis must be updated.

If the assessment indicates that the predicted reliability of a special safety system or safety-related system is less than the target, it means that the system does not meet its design specification. However, it does not necessarily imply that the special safety system or safety-related system would be unable to perform its function. In this case, the reliability report must include:

- evaluation and discussion of the significance of the results
- actions required to increase the predicted system reliability to the target reliability
- schedule for implementation of the above actions

RISK MANAGEMENT AND CONTROL

Programs are established, maintained, and documented to effectively manage and control the risk arising from the radiological hazards associated with nuclear operations to workers, members of the public, and the environment. The objectives of these programs are to make sure that:

- incremental risks are knowingly incurred only if they are understood, justified, and controlled;
- risk to public is maintained low in comparison to other risks from industrial activity to which they are normally subjected;
- workers will only be subjected to radiological risks which are low, understood, and voluntarily accepted;
- risks will be lowered to levels that are reasonably achievable.

PEER EVALUATION PROCESS

The peer evaluation process is a method for reviewing all significant safety processes that are important to the operation and maintenance of nuclear power stations. This process also identifies important improvement areas which include:

- areas in which the reviewed station would benefit from improvement;
- areas in which the reviewed station has developed a successful innovative approach that would be beneficial for broader adoption in the industry.

This process has been developed by INPO, and adopted by the International Atomic Energy Agency (IAEA) and World Association of Nuclear Operators (WANO). Similar processes are used in the chemical industry. In Canada, the nuclear utilities have adapted this approach to their programs.

The objectives of a Peer Evaluation are to:

- assess the extent to which the station is operated safely and reliably;
- promote excellence in operation, maintenance, and support of the plant;
- evaluate:
  - the knowledge and performance of station personnel,
  - the condition and performance of systems and equipment,
  - the quality of programs and procedures,
  - the effectiveness of station management.

A peer evaluation team generally accomplishes this by reviewing the appropriate programs, policies, and procedures for adequacy; and by observing station personnel performing their day-to-day work. Usually, a team would spend the majority of its time in the field with minimum impact on station staff who are performing their duties.

The evaluation team is led by a team manager and consists of approximately 20 evaluators. The team usually has over 200 years of cumulative nuclear experience. The following areas are evaluated:

- Organization and Administration
- Operations
- Maintenance
- Engineering Support
- Radiological Protection
- Chemistry
- Training and Qualification

Evaluators review recent plant data and performance reports. They also become acquainted with the station policies and procedures. Each evaluation area has general standards by which the station is evaluated. These include:

- corporate policies
- industry guidelines
- industry data
- evaluator’s experience

At OHN, the document “Performance Objectives and Criteria for Ontario Hydro Nuclear Generating Stations” is used for the evaluation.

The evaluation team must conduct an exhaustive examination of the station’s performance within a short time (for example, two weeks), that is representative of a period of normal station operation. The way in which the station performs during the evaluation is considered to be a “snapshot” of typical station performance. Particular emphasis is placed on observing station personnel perform their activities.

One measure of the effectiveness of station procedures and policies is how well they are executed. During a peer evaluation, the team focuses on those aspects of the station organization that are important in achieving high standards in the final output. The commitment to meeting performance objectives and criteria require stations to perform in the best manner possible and strive for excellence in each performance area. The actual performance of staff, rather than the documented program plans, is considered to be the yardstick for measurement in achieving excellence.

EXTERNAL EVALUATIONS
In addition to peer evaluations that were performed to the end of 1997, the WANO organization has been contracted by the Canadian utilities to carry out external and independent performance assessments of the operating nuclear power stations.

The WANO evaluation process, like the peer evaluation process, is based on the INPO methodology. The WANO evaluation team excludes staff from the utility but includes international experts with experience in nuclear power station operations. This external expertise is considered vital to support an independent process. The Chief Executive Officer of the utility receives the results of the final evaluation.

In Canada, in addition to the WANO type evaluations, an extensive independent self-assessment process was recently carried out at OHN power stations. This was called the Independent Integrated Performance Assessment (IIPA). The IIPA examined various aspects of OHN performance including control of nuclear design activities. The results of this assessment and the actions taken by OHN are described in Article 6.2.3.

9.2. DESCRIPTION OF THE MECHANISM BY WHICH THE AECB MAKES SURE THAT THE LICENCE HOLDER MEETS ITS PRIMARY RESPONSIBILITY FOR SAFETY

The AECB requires a licensee to identify and demonstrate its commitment to safety in its station operating procedures. This requirement, and the fundamental rules governing safe operation of the reactor, are set out in an Operating Policies and Principles (OP&P) document. The document is prepared and submitted by the licensee with the application for an operating licence for AECB approval (see Article 10.1.2. for more information about OP&P). Any failure of licensee staff to follow the requirements contained in the OP&P represents a breach of the licence and must be reported to the AECB.

AECB reporting requirements are specified in Regulatory Document R-99. They include the following:

- Event reports, which include but are not limited to:
  - unplanned events,
  - violations of a licence condition,
  - excessive or unmonitored radioactive emissions or emission pathways,
  - reportable doses of radiation,

- process failures,
- the actuation of shutdown, emergency core cooling, or containment systems.

In each case, the licensee is required to make an oral event report to the AECB within one business day of the discovery of the reportable condition. In addition, a detailed written report must be subsequently submitted within the time-period specified in R-99.

- Quarterly reports that provide information about:
  - changes in station personnel, procedures, equipment, or emergency exercises;
  - changes that could invalidate information contained in the facility's Safety Report;
  - results of routine and non-routine radioactive effluent monitoring;
  - dose statistics.

- Safety Report updates that reflect design and procedural changes and new analyses. The new analyses are to reflect new tools, methodologies, or research findings. The Safety Report updates are to be submitted within three years from the previous update.

- Annual radiological environmental monitoring reports that include:
  - the results of the off-site radiological environmental monitoring program;
  - the individual doses that were calculated as doses to the critical group;
  - a review of the radiological environmental monitoring quality assurance program;
  - any unusual findings during the calendar year.

- Annual research and development reports that describe research and development programs being carried out, and those planned to address unresolved safety questions.

- Periodic inspection program reports that are required within 90 days of the completion of any inspection carried out in accordance with the periodic inspection program requirements of CSA Standards N285.4 and N285.5.

- Annual reliability reports that contain an evaluation of the reliability of each special safety system, or any other safety-related system, that has specific reliability requirements described in the licensing document.

- Fissionable and fertile substance reports that describe the inventory or the transfer of fissionable and fertile substances.

In addition to reviewing the above reports, AECB routinely audits the licensee's compliance with its OP&P document. AECB also carries out regulatory inspections to make sure there is adherence to station procedures. (See Articles 7.4 and 8.1 for more details about AECB staff duties.)

ARTICLE 10
PRIORITY TO SAFETY

10.1 PRINCIPLES EMPHASIZING THE OVERRIDING PRIORITY OF SAFETY, AND THEIR IMPLEMENTATION

Safety principles, and the procedures and mechanisms to enhance them and give them overriding priority, are adopted and followed by Atomic Energy of Canada Limited (AECL) and the Canadian utilities. AECL performs reactor design and project management for its clients, and accepts their safety and regulatory requirements. The utilities carry the prime responsibility for the safety of nuclear power stations. Each utility is committed to sound nuclear safety principles and practices as an essential and integral part of its nuclear operations.

10.1.1 Safety Procedures at the Designer (AECL)

AECL places health, safety, and protection of the environment amongst its highest priorities. It factors these priorities into activities at all levels. The following statements are extracted from the AECL Management Manual:

- to meet or exceed the letter and spirit of all applicable safety, health, and environmental laws and regulations and, where appropriate, international standards. This is achieved through:
  - a commitment to an enhanced safety culture and the personal health of all employees through the provision of pro-active safety and health programs;
  - striving to protect our employees from unsafe conditions in the workplace, and make sure that those that cannot be eliminated are controlled to keep the risk as low as reasonably achievable;
  - making sure that any radiation exposures and emissions of radioactive materials caused by our activities, products, and services, are significantly below allowable levels; and that they are As Low As Reasonably Achievable (ALARA), taking into account relevant social and economic factors.

- to maintain emissions from our facilities to the environment below the limits defined in applicable regulations, and strive to further reduce them, to be as low as reasonably achievable, taking into account relevant economic and social factors.

- to strive to continually improve our environmental performance and to contribute to improvement in the environmental performance of the nuclear industry.

- to perform independent reviews of the impacts of our activities, facilities, services, and products on health, safety, and environment to make sure compliance with requirements and that they are acceptable.

10.1.2 Safety Procedures at the Utilities

As required by the Atomic Energy Control Board (AECB), the licensee in Canada identifies and demonstrates its commitment to safety in its station operating procedures. This requirement, and the fundamental rules governing safe operation of the reactor, are set out in an Operating Policies and Principles (OP&P) document. The document, prepared by the licensee, must contain:

- a definition of the authority and responsibilities of managers and operating staff
- the specific numerical limits for operating parameters that must be maintained to make sure that the plant always operates within its analysed Safe Operating Envelope (SOE)
- the principles to be applied for the safe operation of each of the plant's systems. These principles cover:
  - control of reactor power at all times
  - fuel cooling at all times
  - containment of fission product radioactivity
  - maintaining design intent
  - adherence to the operating limits that affect public safety
  - maximization of the availability of safety systems
  - maintaining knowledge of unit/system status
  - maintaining defence-in-depth
  - establishing fall back actions/countermeasures
  - applying conservative decision making

The initial OP&P document, with the application for an operating licence, as well as all proposed changes at any future point in time, is approved by the AECB. Like other documents specifically referenced in the operating licence, any failure of licensee staff to follow the requirements contained in the OP&P represents a breach of the licence.

10.2 PRINCIPLES DIRECTLY RELATED TO SAFETY

The design, operation, and regulation of nuclear power stations in Canada are based on the following safety principles:

- defence-in-depth strategies
- As Low As Reasonably Achievable (ALARA) radiation exposure
- establishment and nurturing of safety culture
- design, operation and maintenance by qualified staff
- safe operation limits based on analysis, research and development results, and operation experience
- periodic assessment and reporting of performance
- commitment to quality assurance programs that meet national and international standards

10.2.1 Design Safety Principles

The design of the CANDU reactor is based on the principles of multiple barriers to radioactive releases and multiple ways for guaranteeing each of the following basic safety functions:

- accident prevention measures
- redundancies in equipments and procedures
- diversity in performing safety functions
- physical and functional separation of the safety systems

See Article 18 for more information about the CANDU design safety principles.

10.2.2 Operation Safety Principles

ESTABLISHMENT OF SAFETY CULTURE
The safety culture is established and maintained by factors such as:

- the responsibility for safety rests with the operating utility;
- the ultimate responsibility for nuclear safety rests with the Board of Directors of the utility and requires a clearly defined delegation of that responsibility throughout the organization;
- the Board of Directors of the utility is committed to sound nuclear safety practices and provides the necessary resources to implement nuclear safety;
- organizational structures and alignment of functions that clearly define and provide for all required functions placed at the requisite organizational levels;
- accountability and authority assigned both vertically and laterally as required;
- managers are held individually accountable for making sure that they, and all employees under their direction, are aware of and committed to achieving safe and reliable operations through quality performance of expected behaviours and practices;
- operation of nuclear power stations according to the best practices in the international nuclear community;
- providing and maintaining the resources necessary to safeguard the health and safety of employees, the public, and the environment from radiological hazards;
- establishing, maintaining, and documenting programs to effectively manage and control the risk arising from radiological hazards;
- developing and delivering systematic training to employees.

The following are recent examples of conservative decision-making toward achieving an improved overall safety at the Canadian utilities. These examples provide a strong message that safety margins and limits will not be compromised in favour of electrical output. They also provide the foundation for developing a safety culture at all levels within the organization.

- at Ontario Hydro Nuclear (OHN):
  - the Chief Nuclear Officer's directive to limit the reactor power level of eight units pending a thorough assessment of the safety analysis process and SOE;
  - the proactive, conservative and thorough investigation of a low flow trip issue and the subsequent corrective actions;
  - self-initiated restart prerequisites in response to issues identified during Safety System Functional Inspections (SSFI).
- at Hydro-Québec: a voluntary decision was made to reduce reactor power at Gentilly-2 station to 97% following an indication of reduction in total reactor flow (see Article 6.2.2 for details).

SAFETY POLICIES

The major administrative control for the implementation of the Nuclear Safety Policy is the OP&P for each station. The OP&P are approved by AECB and explain in detail how the licensee shall operate, maintain, and modify station systems to maximize nuclear safety and keep the consequential risk to the public acceptably low. See Article 10.1.2 for details of the contents of OP&P.

The governing principle promulgated by OP&P is to maintain station operation within the defined boundaries of the SOE. In addition, to prevent, mitigate and accommodate any potential nuclear incident or accident, OP&P requires that the principles of nuclear and reactor safety be adhered to. Under no circumstances is the intent of the governing principles to be intentionally violated. If conditions are found to exist that contravene OP&P, either the affected system will be placed in the normal configuration or other safe state, or the unit will be put in a safe state following procedures and practices approved by the Operations Manager.

SAFETY ASSESSMENTS

Safety assessments are performed regularly for the nuclear power stations, and upon the occurrence of accidents or unplanned events in Canada or abroad. Unplanned events are also tracked and analysed and the necessary corrective actions are taken. Safety assessments performed in Canada, their results, and the corrective actions taken are described in Articles 6.2 and 6.3. Programs to collect and analyse operating experience including unplanned events are described in Article 19.2.7.

VOLUNTARY PRACTICES AND GOOD PRACTICES RELATED TO SAFETY

In addition to the fulfilment of regulatory requirements, utilities, on a voluntary basis, implement numerous initiatives directly related to improving nuclear safety. These include, but are not limited to:

- training all managers in basic managerial leadership practices that encourage:
  - effective two-way feedback,
  - coaching,
  - promotion of communications between managers and their subordinates about task assignments, deadlines, quality, resources, and employee concerns.

- establishing corporate and site performance indicators that support the right behaviour for safe, reliable operations.
- setting radiation dose limits for workers that are lower than the regulatory limits.
- establishing the following at OHN:
  - cross-functional and cross-site advisory roles for collecting and sharing lessons learned and best practices,
  - Executive Vice President (EVP) directed self-assessment groups that will do in-depth assessments across all sites on a continual basis on behalf of the EVP,
  - an on-site and off-site safety review capability that reports to the Site Vice President.

10.2.3 Regulatory Control Safety Principles

The safety principles in the AECB regulatory control stem from its mandate to make sure that the use of nuclear energy does not pose undue risk to health, safety, security, and the environment. This is exemplified through several mechanisms including, but not limited to:

- establishing classes of licences authorizing the construction and operation of a nuclear facility. Each licence may contain such conditions that the AECB deems necessary in the interests of health, safety, and security.
- maintaining a Senior Project Officer and staff at each facility to inspect the premises, records, and activities to make sure there is compliance with the licence and governing regulations. A summary of each station's performance against legal requirements, including the conditions in the operating licence, is issued annually.
- certifying that personnel for authorized positions meet the applicable qualification requirements, and have successfully completed the applicable training program and examination.
- conducting special hearings, and inspections as necessary.
- establishing a two-year licence renewal practice as a mechanism to make sure that there is compliance and periodic safety review.

In the near future, under the new Nuclear Safety and Control (NSC) Act, the Canadian Nuclear Safety Commission will have a clear mandate to establish and enforce national standards in the areas of health, safety, and environmental consequences of nuclear activities. The Canadian Environmental Assessment Act (CEAA) will continue to apply. The current practice will continue with the Commission considering the recommendations of CEAA panels as part of its licensing decision process, and will regulate all stages in the development, construction, and operation of nuclear facilities in cooperation with other regulating agencies to make sure appropriate levels of nuclear safety are met. See Article 7.2.1 for more information about the new NSC Act.

ARTICLE 11
FINANCIAL AND HUMAN RESOURCES

11.1 THE FINANCIAL AND HUMAN RESOURCES OF THE LICENSEE THAT ARE AVAILABLE TO SUPPORT THE NUCLEAR POWER STATION THROUGHOUT ITS LIFE

Each licensee in Canada has the prime responsibility for the safety of its nuclear power station. This responsibility includes providing both adequate financial and human resources to support the safety of each nuclear power station throughout its life. Adequate human resources are defined as the employment of enough qualified staff to carry out all normal activities without undue stress or delay, including the supervision of work done by external contractors.

The licensees of nuclear power stations in Canada are all publicly-owned electric utilities and report to a government body. Each has the authority to raise revenue through:

- the sale of bonds in the financial market
- the sale of electricity in order to assure that adequate financial resources are available to support the safety of each nuclear power station

In Ontario, the Ontario Energy Board (OEB) reviews and approves proposed rate changes. Each utility currently has exclusive rights to the sale of electricity in its service area. By the year 2000, the sale of electricity in each service area will be open to competition.

Each utility assesses its human resource requirements and recruits qualified staff on the open job market within Canada. Each utility also supplements individual training as required by extensive in-house training programs. If human resource needs exceed the availability of qualified candidates on the open market in Canada, the utilities can temporarily supplement their human resources by:

- hiring qualified consultants both nationally and internationally
- expanding their recruitment horizons to seek qualified personnel internationally

If human resource demands exceed this extended supply of suitable candidates, then the utility may make sure of adequate operational human resources by reducing demand. This involves reducing the level of operational activity within the utility by laying up or closing units, and redeploying qualified staff internally.

A fundamental conclusion of the Independent Integrated Performance Assessment (IIPA) of Ontario Hydro Nuclear (OHN) (see Article 6.2.3 for details) was that the utility did not have sufficient qualified staff to operate its 20 units and implement its performance improvements programs at the same time. Therefore, OHN decided to lay-up, at least temporarily, some of its units and redirect the qualified personnel at those units to assist in the implementation of the improvement programs.

The financial implications associated with such programs under the Nuclear Asset Optimization Plan (NAOP) are significant. The NAOP is structured to determine what human and financial resources are necessary to implement the required performance improvements. The OHN Board of Directors has reviewed and endorsed the NAOP. The Board is committed to provide the necessary financial and human resources required to support safe operation of each nuclear power station throughout its life.

11.2 THE FINANCING OF SAFETY IMPROVEMENTS MADE TO THE NUCLEAR POWER STATION DURING ITS OPERATING LIFE

The Canadian utilities maintain two separate budgets:

- the Operating and Maintenance budget
- the Capital Improvement budget

The Operating and Maintenance budget is used to make safety improvements to the nuclear power station during its operating lifetime. Large scale improvements, however, are covered by the Capital Improvement budget.

For large scale improvements, an item is costed for financing over the estimated remaining effective lifetime of the station. If approval for the station to proceed is received, then the item would be factored into the overall cost of operating the plant for future years. This in turn is factored into establishing the future consumer electricity rate. Because the provincial regulatory agencies (the shareholder of Canadian power stations) have to review and approve increases in electricity rates, they may instead decide to finance a large scale expenditure from general tax revenues. In either case, a large scale expenditure will not:

- affect normal operating budgets
- result in a negative impact on the financial resources available for other purposes bearing on nuclear power station safety

For OHN, which is a self-regulated Corporation, expenditures are dictated by:

- the company’s financial position
- current and planned performance
- service obligations (load forecast)
- financial and business strategies

These inputs are used to develop the envelopes for ongoing operating expenditures and for capital investments. The general rule is that ongoing safety-related programs are financed from the operating envelope and that large-scale improvement projects, including safety-related projects, are financed from the capital envelope. In either case, the costs of safety improvement programs/projects are part of the rate base and are recovered usually through rates charged to customers.

Within each envelope, programs/projects are ranked in accordance with criteria that reflect these objectives of the corporation:

- operating
- business
- financial

OHN assigns a high priority to safety-related programs/projects which have a significant impact on nuclear safety. This high priority makes sure that adequate financial resources will be applied to safety improvement programs/projects throughout the life of each nuclear power station.

11.3 PROVISIONS FOR FINANCIAL AND HUMAN RESOURCES FOR DECOMMISSIONING THE NUCLEAR POWER STATION AND RADIOACTIVE WASTE MANAGEMENT

The Atomic Energy Control Board (AECB) currently has no legislative basis for requiring licensees of nuclear power stations to provide guarantees that adequate financing and human resources will be available for the:

- decommissioning of nuclear power stations
- management of the resulting radioactive wastes including spent fuel

The new Nuclear Safety and Control (NSC) Act (see Attachment 7.7) and the draft for General Regulations - Section G2.1(1) require that licensees provide financial guarantees acceptable to the Commission. This legislation is expected to come into effect by the end of 1998 or early 1999.

The following describes the general principles that will be used to develop a Regulatory Guidance Document for the provision of financial guarantees for decommissioning activities:

- Proponents and operators of nuclear power stations shall propose decommissioning plans and funding measures.

- Decommissioning plans must be sufficiently detailed in order to:
  - demonstrate that they will remediate all significant impacts and hazards to persons and the environment in a technically feasible fashion;
  - make sure that compliance with all applicable requirements and criteria established in Acts, Regulations, and other Regulatory Standards and Guides is achieved;
  - enable credible estimates of the amount of financial guarantees.

- Financial guarantees:
  - must be sufficient to fund all approved decommissioning activities; and,
  - should be at arms' length from the licensee, and the AECB must be assured that it or its agents can, upon demand, access adequate funding measures if a licensee is not available to fulfill its obligations for decommissioning.

- Measures to fund decommissioning may involve various types of financial guarantees. Acceptable guarantees include: cash, letters of credit, surety bonds, insurances and legally binding commitments for a government (either federal or provincial).

- The acceptability of any of the above measures will be ultimately determined by the AECB on the basis of the following general criteria:

  - Liquidity: The proposed funding measures should be such that the vehicle can be drawn upon only with the approval of the AECB, and that pay-out for decommissioning purposes is not prevented, unduly delayed, or compromised for any reason.

  - Certainty of Value: Licensees should select funding, security instruments, and arrangements that provide full assurance of their value.

  - Adequacy of Value: Funding measures should be sufficient, at all or predetermined points in time, to fund the decommissioning plans for which they are intended.

  - Continuity: The required funding measures for decommissioning should be maintained on a continuing basis. This may require periodic renewals, revisions and replacements of securities provided or issued for fixed term. Where necessary to make sure there is continuity of coverage, funding measures should include provisions for advance notice of termination or intent to not renew.

11.4 THE RULES, REGULATIONS AND RESOURCE ARRANGEMENTS CONCERNING THE QUALIFICATION, TRAINING, AND RETRAINING OF PERSONNEL, INCLUDING SIMULATOR TRAINING FOR ALL SAFETY-RELATED ACTIVITIES IN OR FOR EACH NUCLEAR POWER STATION

In Canada, there is a hierarchy of laws, regulations and utility practices that provide requirements about personnel who perform critical safety-related activities, such as:

- the number of staff
- their qualifications
- their training

First, the Atomic Energy Control (AEC) Act (see Attachment 7.1) provides the legislative basis for insuring nuclear safety.

Second, the AEC Regulations (see Attachment 7.2) specifically address requirements for personnel.

Third, specific provisions are included in each operating licence which requires specific numbers of personnel, with explicit qualifications and training.

Fourth, each licensee has practices that address these personnel issues.

Under this Act, the AECB grants licences that authorize the licensee to construct, operate, modify, and decommission a nuclear power station in accordance with defined safety objectives and performance requirements pursuant to conditions of the licence.

The AEC Regulations require that each applicant for a licence provide details about the qualifications, training, and experience of any person involved in the operation of the nuclear power station.

Each operating licence contains general requirements to:

- make sure that there is a sufficient number of qualified workers to carry on the licensed activity safely and in accordance with the governing regulations;
- train workers to conduct the licensed activity in accordance with the provisions of the governing regulations.

Each operating licence also contains specific requirements to:

- have enough qualified personnel (minimum shift complement) in attendance at all times to make sure there is safe operation of the nuclear power station. The minimum station complement is specified in administrative documents, which receive AECB approval.
- have the following staff in the four-unit nuclear power station at all times, except as otherwise approved in writing by the AECB:
  - four authorized nuclear operators,
  - one shift supervisor,
  - a second AECB authorized supervisor.
- have in the control room two persons who have been authorized in writing by the AECB and who are qualified to operate the controls, unless, in the judgement of the senior operations person at the nuclear power station, the hazard to personnel would be unwarranted.

- have any significant changes in staffing and organization submitted to the AECB at least 30 working days before they are implemented, and be approved in writing by the AECB before implementation.

- have written approval from the AECB before the appointment of individuals to specific positions such as the following positions at OHN (equivalent positions exist at New Brunswick Power and Hydro-Québec):
  - operation manager,
  - shift superintendent,
  - shift operating supervisor,
  - authorized nuclear operator.

These authorized positions have detailed qualification, training, and retraining requirements commensurate with their accountabilities and authorities. When a licensee requests authorization by the AECB to appoint a person to an operating position, the licensee is required to submit:

- evidence that the person has successfully completed the required training programs
- evidence of successful completion of examinations set by the AECB
- evidence of co-piloting periods relevant to the position
- a written recommendation from the Operations Manager

The qualification process for authorized nuclear operators involves comprehensive simulator-based training and testing that exposes the trainee to a wide range of normal, off-normal, and accident conditions. The simulator experience is designed to create an environment and plant situation that is as realistic as possible. To create this high degree of realism, extensive efforts are applied to make sure that the fidelity of the simulator response and environment is like that of the actual nuclear power station. Simulator-based training and testing is also provided on a regular basis for authorized operators as part of the ongoing maintenance of their authorized status.

The Shift Superintendents, Shift Operating Supervisors, and Authorized Nuclear Operators each are required to successfully complete a formal evaluation of their ability to deal with a simulated station transient as part of the normal duties of the position.

Finally, each licensee allows only highly qualified and competent staff to perform the following functions and tasks which are critical to nuclear safety:

- recognize when a layer of defence is threatened by proposed actions, or changes to equipment, procedures, or staffing;
- monitor, operate, and maintain safety related systems (for example, calibrate instrument loops, perform safety system tests, welding, etc.);
- identify incipient equipment failures, so that corrective action can be taken before catastrophic failures occur;
- properly execute emergency response procedures to mitigate and accommodate the consequences of potential accidents.

The individuals performing tasks critical to nuclear safety are not limited to operations personnel, but also include other personnel such as:

- Engineers
- Station Engineering Support Staff
- Station Health Physics Staff
- Maintenance Support Staff
- Modifications Staff

These positions also have detailed qualification, training, and retraining requirements commensurate with their accountabilities and authorities.

ARTICLE 12
HUMAN FACTORS

12.1 THE METHODS USED TO PREVENT, DETECT AND CORRECT HUMAN ERRORS, INCLUDING ANALYSIS OF HUMAN ERRORS, MAN-MACHINE INTERFACE, OPERATIONAL ASPECTS, AND EXPERIENCE FEEDBACK

Canadian nuclear utilities, design, and regulatory organizations recognize the important role that human performance has in the safe design and operation of nuclear power stations.

Canadian organizations use a variety of approaches to make sure that human factors expertise is available from the following sources:

- dedicated human factors engineers/scientists on staff
- external human factors consulting firms

The methods of resourcing vary from organization to organization depending on the level of expertise required, resource availability and specific project needs.

All organizations recognize that the knowledge, experience and proficiency of their staff are important requirements to safely operate their facilities and to carry out their functions. All are committed to identifying required qualifications and to providing appropriate training to satisfy job requirements. Human factors engineering training is provided to staff as required.

Design activities that have a significant human interface component or that could otherwise impact on human performance are subject to a structured and systematic human factors engineering program. This program is integrated into the overall design engineering process.

Human factors design is one of several complementary human factors roles. These complementary roles include:

- health and safety programs
- human error in probabilistic risk assessment
- procedure development
- training

The goal of the human factors designer is to reduce operations and maintenance errors by integrating human factor considerations throughout the design cycle.

12.1.1 Operations Activities

Programs are tailored to the needs of the “project”, which include:

- new station design
- minor station modifications
- enhancements to an existing station
- Ongoing experience review

For projects with significant human machine interface components (for example, new station or significant improvement to existing design), the program is defined in a project-specific document called the Human Factors Engineering Program Plan (HFEPP). For modifications judged to be of a more minor nature, changes may be executed without specific HFEPPs. However, these minor changes must be executed according to design guides and expert subject matter input. Alternatively, these changes may be covered by a generic HFEPP.

There are several guiding principles that shape the approach to the various human factors design tasks. These principles address issues such as:

- minimizing human error
- designing for human cognition
- designing for human physical characteristics
- designing for supervisory control and automation

Methods employed by the human factors staff includes:

- subject matter expert interviews (providing feedback from operational experience);
- system and task analysis (defining event sequence, task allocation and operating procedures);
- assessment of compliance with international standards;
- mock-ups;
- prototypes (physical and virtual);
- dynamic simulation, both for design and evaluation purposes.

Canadian nuclear utilities and design and regulatory organizations have ongoing programs to capture, assess, and disseminate information in the following areas:

- design
- supply
- construction
- commissioning
- operations (CANDU and international events)
- regulatory issues
- research and development

Human factors considerations are part of the assessment process. For example, Significant Event Reports (SERs) and other incident reports are gathered and catalogued in both a utility database and an industry-wide database. Reports with a human error/ performance component are identified. These reports are monitored and distributed on an ongoing basis for assimilation of lessons learned. During design, these event reports are reviewed for potential applicability to the system or equipment that is being modified.

For modifications of significant complexity, designers review operational experience on similar systems. This operational experience may span several stations and usually covers both operational and maintenance issues. Collection of this data may be through documented sources such as previous operational reviews or by interviewing subject matter experts in relevant areas.

Canadian nuclear organizations have active research and development programs to support both short and longer term design, operations and regulatory needs. Seminars, reports, and conference papers are used to disseminate the results. Recent topic areas include:

- development of methods for evaluating systems to be retrofit into existing designs
- development of new displays to enhance operator situation awareness
- human performance issues in fuel handling and refuelling
- development of a systematic method for regulatory assessment of licensees' organization and management

AECL also has a program that provides training for human factors engineering design to its process, control, and instrumentation design staff. It also encourages the establishment of human factors "champions" within each design unit within the organization.

12.1.2 Design Activities

Nuclear power plant design activities in Canada follow one of two general design philosophies. The first of these is design evolution, in which the design effort for a current build project is evolved from a previously built design (which, in turn, was based on an even earlier design). This is the approach that has been taken with the CANDU 6 product. AECL is also engaged in new product design that uses a proven existing plant design as a reference, and encompasses improvements based on operating experience. This is the approach that is being taken with the CANDU 9 product. Within either of these contexts, design activities that have a significant human interface component or that could otherwise affect human performance are subject to a structured and systematic human factors engineering program, which is integrated into the overall design engineering process. The exact scope of this program differs between the two design philosophies (because the fundamental design processes also differ).

The nature of the human factors engineering program for a design is described in a HFEPP. These programs are invoked and are structured in a manner consistent with the design processes called for in the following documents:

- NUREG 0711 (Human Factors Engineering Program Review Model)
- IEEE 1023 (Guide for the Application of Human Factors Engineering to Systems, Equipment and facilities of Nuclear Power Generating Stations)
- IEC 964 (Design of Control Rooms of Nuclear power stations)

For the evolutionary design philosophy, there is a generic HFEPP that describes how proposed design changes will be evaluated to determine the need for a detailed human factors effort. This effort is then described in a more detailed HFEPP that is produced for new design. The detailed HFEPP identifies the elements of the program and explains how the elements will be managed. Specifically, the HFEPP describes the organizational structure and responsibilities of the human factors design team and indicates the level of effort required for the various elements of the program. The HFEPP details the planned human factors design tasks, including:

- analysis, detail design, and test and evaluation activities to ensure that the resultant plant design reflects current knowledge about human cognitive and physical performance;
- the engineering control methodologies that will be applied to the human factors design effort;
- milestones and deliverables that are synchronized with the overall design process and provide opportunities to assess the human factors design work.

Of the various human factors design activities, it is the analysis tasks that provide the most significant opportunity to reduce the potential for human error. They also contribute to an optimum design from a human performance standpoint, as follows:

- Function analysis, carried out as early as practicable in the design process, is used to identify and evaluate the functions that must be performed by the various plant systems in achieving the overall objective of safe and reliable electricity production.

- Human factors engineering principles and criteria are applied to determine human/equipment/software performance requirements for system operation and maintenance, and to allocate system functions to personnel, automation and/or a combination of the two.

This function analysis requires an understanding of the information flow and processing needs during operation and maintenance, and an understanding of a baseline estimate of operator/maintainer skills and capabilities. Where warranted (for example, where human performance is considered critical), the function analysis leads to a more detailed analysis of tasks, the results of which are used as part of the detailed design activity.

For operations, human factors designers produce an operational basis document that:

- provides a description of the operational principles and practices that govern the operation of a CANDU station in all possible operating situations;
- is done for both the evolutionary design and new design philosophies;
- provides the framework for describing these operating situations and the operational activities to be supported;
- is a summary description of the operational intent and preferred practice that any design change proposal must integrate with and support;
- is a “reference map” of station operational practices to promote understanding amongst designers;
- facilitates the identification of the impact of proposed design changes;
- assists in documenting the operational relevance of design features;
- guides the implementation of design changes.

A separate operational basis document is produced for each CANDU product.

12.2 MANAGERIAL AND ORGANIZATIONAL ISSUES

Each licensee incorporates an organizational and management philosophy that accounts for the capabilities and limitations of human performance. This is accomplished in a hierarchical method as follows:

- The primary responsibility for human performance rests with each individual.
- First line managers are accountable for monitoring and correcting human. performance issues.
- Management provides the necessary expectations, facilities and tools to aid human performance.
- Non-line organizations provide independent oversight of human performance.

Details of each level of this hierarchy are provided below.

12.2.1 The Primary Responsibility for Human Performance of Each Individual

Clear lines of authority and communication are established so that individuals throughout the organization are aware of their accountabilities toward nuclear safety. One of the primary management responsibilities for the licensee of a nuclear power station is the establishment of a safety culture. This safety culture emphasizes the necessity for personal dedication and accountability for each individual engaged in an activity that has a bearing on the safety of the nuclear power station. An individual’s recognition and understanding of this accountability, as well as a questioning and self-checking attitude, are essential for the minimizing human errors.

12.2.2 First Line Managers and Their Responsibilities in Human Performance Issues

The primary method used to detect human error is direct involvement in the work performed by observing and verifying the performance of employees. The flow of information and the communication of problems both up and down the line are key to detecting errors. This includes encouraging the admission of human errors.

12.2.3 Management's Roles and Responsibilities

Management's roles and responsibilities to aid in human performance include:

- clearly communicating performance expectations through meetings, policies, and procedures;
- establishing an effective organization with well-defined and understood accountabilities and authorities, and with sufficient numbers of properly qualified workers;
- developing sound procedures to clearly define safety-related tasks and evolutions to further reduce the possibility of human errors. The procedures are continuously enhanced through the incorporation of lessons learned. These lessons come from both the employees using the procedures as well as from organizations charted with the review of internal and external operating experience;
- providing the necessary training and education to individual employees to emphasize the reasons behind the established safety practices and procedures, together with the consequences of safety shortfalls in personal performance;
- providing sufficient and proper facilities, tools and equipment, and support staff;
- conducting self assessments.

In addition, each level of management is vested with a specific level of authority as defined in the station Operating Policies and Principles (OP&Ps) or other documents. Each manager has a clear understanding of what they can approve, versus what they must refer to a higher authority. Errors in decision-making are minimized by requiring that any individual approving a document or activity makes sure there is consistency and compliance with:

- the limits of authority of the individual's position;
- the applicable external requirements (e.g., laws, regulations and licence) and internal boundaries (e.g., OP&Ps, Safety Reports, Radiation Protection Regulations, and Quality Assurance Manuals);
- operating and maintenance practices;
- design assumptions and intent.

12.2.4 NON-LINE ORGANIZATIONS PROVIDING INDEPENDENT OVERSIGHT OF HUMAN PERFORMANCE

Several levels of oversight that are independent of the line organization also review human performance.

The first level of oversight within Ontario Hydro Nuclear (OHN) is provided by the Manager of Performance Assurance who reports directly to each Site Vice-President.

The Manager of Performance Assurance develops and executes an audit plan consistent with OHN Performance Assurance documents, licence requirements and commitments to the Atomic Energy Control Board (AECB). Performance-based assessments and surveillance of site functions are accomplished by site Performance Assurance personnel. Results are documented, trended, evaluated, and reported to allow for early detection and correction of performance problems.

The second level of oversight within OHN is provided by the Director of Performance Assurance. The Director of Performance Assurance:

- reports directly to the Executive Vice President & Chief Nuclear Officer, and is responsible for assessing the performance of all OHN functions;
- initiates and coordinates internal and external nuclear safety reviews, Safety System Functional Inspections, audits and surveillance consistent with OHN performance objectives and criteria.

The third level of oversight within OHN is provided by the Nuclear Oversight Committee (NOC) which entails a broad, systematic, and independent overview of nuclear safety that ensures the requirements of OHN's Safety Policy are met.

The NOC accomplishes its objective by the review of:

- various safety activities
- organizations
- programs
- procedures
- requirements and results with respect to effectiveness
- significance of occurrences
- trends that may affect nuclear safety

The NOC reports directly to the Executive Vice President & Chief Nuclear Officer and to the President and Chief Executive Officer.

In addition to the independent safety oversight groups within OHN, the Technical Advisory Panel on Nuclear Safety (TAPNS) and the Nuclear Review Committee report directly to the OHN Board of Directors.

In summary, management implementation of a "defence-in-depth" strategy is fundamental to addressing human performance issues as they relate to the safety of nuclear power. A defence-in-depth strategy compensates for potential human errors (and mechanical failures) through the use of successive barriers that prevent the release of radioactive material to the environment. When properly applied, no single human error (or mechanical failure) has the potential for compromising the health and safety of the public.

12.3 THE ROLE OF THE REGULATORY BODY AND THE OPERATOR

12.3.1 The Role of the Regulatory Body

One of the roles of the AECB is to make sure that licensees include human factors in the design, assessment, and operation of nuclear power stations. This is done through direct interaction with licensees about:

- design and modifications
- audits and assessments
- research projects for both direct application and the development of future regulatory tools

The benefits of the application of human factors principles are far-reaching in that human error may be reduced, operations made more efficient, and unsafe conditions prevented.

For each major design project undertaken by licensees, the AECB monitors the progress of that effort. This may be done through:

- licensee submission of human factors plans
- design guides and other documentation
- update and progress meetings

This process continues through the commissioning phase.

During the operational phase of a project, the AECB addresses various issues related to the operation and maintenance of the equipment. In addition, personnel and organizational issues are addressed. These may include, but are not limited to:

- staffing levels
- organization and management
- operator performance
- maintenance
- work control
- event analysis feedback
- operating experience feedback
- training

12.3.2 Recent Human Factors Activities at AECB

Over the past year, the Human Factors Section of the AECB has focused efforts on several areas including:

FUEL HANDLING

- A research contract to identify the human factors issues associated with changing from Fuelling Against the Flow (FAF) to Fuelling With the Flow (FWF) at Bruce A was completed in March 1997. The study identified a number of human performance issues associated with the transition to FWF in these particular CANDU reactors. A number of recommendations for the transition period, and for the long term operation and maintenance of the Bruce A fuel handling systems, were made to the licensee.

- The Human Factors Section commissioned a follow-up study to a 1994 research project on human factors issues in fuel handling at Darlington. The overall finding of the follow-up assessment indicated that many positive improvements had been made since the original study, including:

  - improvements to operating manuals
  - better tracking of Significant Event Report (SER) actions

  Recommendations were made about areas for ongoing AECB monitoring, and a small number of outstanding issues to be addressed by OHN were identified.

DEVELOPMENTS IN SAFETY CULTURE INITIATIVES

A major goal of the AECB is to develop and implement a systematic method for the regulatory assessment of licensees’ organization and management before the year 2000. This includes both CANDU reactors and other nuclear facilities. Considerable advancements were made during 1997 towards achieving that goal.

Building on work from previous stages of the research project, a model of human organizational characteristics in a CANDU nuclear power station was developed. The basis for that development was Mintzberg’s Machine Bureaucracy model[3]. However, it was discovered that the model did not adequately describe the influences of corporate level and dynamic external processes on Canadian nuclear power stations. This problem was resolved by combining aspects of two other Mintzberg models, the Adhocracy and Professional Bureaucracy models, into the Machine Bureaucracy. The resultant hybrid is called the Canadian Adaptive Machine Model (CAMM).

3. Mintzberg, H.T., “ Structure in Fives: Designing effective organizations,” New Jersey, Prentice Hall, 1983.

The CAMM was used to develop a number of hypotheses that were tested in a field situation with the cooperation of a Canadian nuclear power station. A sufficient number of the hypotheses were corroborated to support the adoption of the CAMM as a basis for further development of the organization and management assessment method. Arrangements are being made to further refine the tools used in the prototype methodology and this will form part of the research effort for 1998/99.

A seminar was held in Ottawa in December 1997 at which the results of the work described above were presented. The final report of that phase of the project will be released in the near future.

SIGNIFICANT EVENT REPORTING PROGRAM EVALUATION
During August 1997, the Human Factors Section conducted an appraisal of the Darlington Operating Experience Root Cause Analysis process. This was part of an ongoing effort on the part of the AECB to evaluate these programs at each CANDU nuclear power station in Canada. The process was found to be in a state of flux due to recent OHN initiatives in this area. As a result, further examination will be made after OHN has declared the new event reporting and investigation process to be fully in place.

RESEARCH CONTRACT ON THE PSYCHOLOGICAL IMPACTS OF ORGANIZATIONAL CHANGE
Concerns have been received from project officers about the psychological impact on OHN employees as a result of the lay-up and decommissioning of certain OHN facilities. In response, a short-term contract was initiated to obtain state-of-the-art information about the key issues in such circumstances. This contract has identified several performance indicators that can be used by the AECB to monitor the degree of any stress-induced behaviour change at OHN.

12.3.3 The Role of the Operator

The operator plays a key role in the detection and correction of human errors. The following are a few of the methods used to provide the ability to detect and correct human errors before their occurrence:

- self-checking and peer reviews that are implemented to reduce the incidence of human errors which involve incorrect manipulation of equipment;
- independent and second verifications to provide another method of identifying human errors which involve incorrect equipment configuration;
- providing experience feedback; the operator has the ability to identify weaknesses in procedures that could potentially result in human errors. Procedures can then be improved to effectively provide input/feedback to the Operating Experience Program.

ARTICLE 13
QUALITY ASSURANCE

13.1 QUALITY ASSURANCE POLICIES

In 1976, the Atomic Energy Control Board (AECB) proposed that standards on nuclear quality assurance be developed. A first committee was formed with participants from the industry and chaired by a member of AECB staff. The Canadian Standards Association (CSA) N286 series of Quality Assurance (QA) standards were developed and published to cover activities in various phases of the life-cycle of a nuclear facility. This includes:

- procurement
- design
- construction
- commissioning
- operating
- decommissioning

Recently two additional standards on QA were also published for computer software programming. All of the standards in the CSA N286 series incorporate a set of management principles that were developed by the N286 committee and incorporated into the CSA N286 series of standards.

As a minimum, the QA policies and programs that licensees are required to develop and implement must satisfy these standards. Because these standards are consensus documents, users are reminded that additional requirements may be imposed by the nuclear regulator.

The licensee is required to develop policies for two levels of application:

- The first level policies apply to the owner’s oversight responsibilities that are applied in all of the phases from design to decommissioning.

- The second level policies apply in each of the individual phases. They result in a licensee “corporate” owner of the licence specifying the requirements and direction to its own organization and to external organizations.

These requirements and direction address activities in various phases of the life-cycle of a nuclear facility. The “corporate” owner is then responsible for ensuring that they are successfully completed. The organizations that are responsible for the work develop their own policies and practices for control purposes.

At Atomic Energy of Canada Limited (AECL), the QA program is also based on principles incorporated in the CSA QA standard CSA N286. These principles are consistent with those on which the ISO-9000 series of quality standards are based. Designs and associated design documents are verified to make sure that the design is correct and meets specified requirements. This design verification is done by reviews (supervisory reviews, peer review and comment, independent third party and panel design reviews) and by testing. The extent of design verification is determined by the following design characteristics:

- complexity
- novelty
- degree of standardization
- state-of-the-art status
- safety implications

Verification requirements are identified in the design verification and quality engineering plans. These plans identify:

- design activities to be verified
- nature and extent of verification
- persons performing verification activities
- methods of verification
- relative location of the verification activities in the design cycle

Design review methods and requirements are specified in AECL's Design Verification procedure. For proven designs used in a new application an appropriate program of design reviews and tests is applied, to confirm that all existing analyses are valid and that the application of the design is correct.

The design work that AECL subcontracts to outside consultants is verified in the same manner as described above, or in accordance with the consultant's QA program as approved by AECL. The design work for which the supplier (manufacturer) is responsible is verified by the supplier. AECL has the responsibility for ensuring that the contractor has performed such design verifications and that the designer has used correct design inputs.

If required, verification or certification of the following reports is carried out in accordance with applicable codes, standards, and procedures:

- design reports
- stress reports
- seismic reports
- environmental qualification reports
- reports prepared by suppliers

Testing that is necessary to validate the design of critical systems and components is identified and executed. This testing is performed within AECL or by a contractor. When tests are performed by AECL, test requirements, test procedures, test data, test assumptions, and test results are documented and filed. Test results are evaluated against the specified acceptance criteria. Conclusions of the test are recorded and filed. Preparation of test requirements, test procedures, and test reports is controlled through applicable procedures. When tests must be performed by a contractor, test requirements are specified in the procurement documents.

Computer software used for the following applications is verified, validated, and documented:

- design
- design support analysis
- plant and safety system control
- safety analysis
- computer-assisted design
- drafting

These verifications, validations, and documentation are controlled through applicable procedures and plans.

When a manufacturer's standard product is selected with or without minor design changes, the design will be subjected to review and/or testing to demonstrate the satisfactory performance of the item. Alternatively, the manufacturer's evidence of verification may be evaluated to ensure satisfactory performance of the item.

13.2 LIFE-CYCLE APPLICATION OF QA PROGRAMS

Canada's new Nuclear Safety and Control (NSC) Act will give the Canadian Nuclear Safety Commission (CNSC), the new name for the AECB, the authority to make regulations regarding the nuclear facility, or part of a nuclear facility, with respect to activities in various phases of its life-cycle. This includes the life-cycle phases given above in Article 13.1, in addition to the following:

- siting
- maintenance
- modification
- abandonment
- disposal

This new act and the resulting regulations will require licensees to prepare and implement quality assurance programs for the life of the nuclear facility.

The AECB requires licensees and other organizations involved in activities related to nuclear safety to establish and implement QA programs. These programs are applied during all phases of the facility life-cycle from its design until it is decommissioned. Their main objective is to facilitate, support, and preserve safety objectives during various phases of the life-cycle of the facility. QA programs should focus on performance and emphasize the full responsibility of those who do the work, such as:

- designers
- constructors
- manufacturers
- operators
- maintenance workers
- radiation protection personnel

The licensee and the other organizations involved must demonstrate the effective fulfillment of the QA requirements to the satisfaction of the AECB.

According to the present licensing practice, when a licensee prepares the preliminary safety report for the nuclear facility, the AECB reviews it to determine what commitments the licensee is making about QA during the life of the facility. The expectation is that the licensee will commit to meeting the requirements of the nuclear QA standards referred to in Article 13.1 for the work involved during each phase of the project. The Safety Report should identify if the licensee intends meeting these standards. In this way, all further work related to activities in various phases of the life-cycle of a nuclear facility will be governed by corresponding standard. The AECB ensures that these commitments are identified when its staff reviews the Preliminary Safety Assessment Report (PSAR). Later, the AECB staff carries out reviews and audits to ensure that the licensees meet these commitments during each of the phases of application.

Nuclear safety is the fundamental consideration for identifying the items, activities, and processes to which the QA programs are to be applied during each of the life-cycle phases. The QA standards define what safety-related means. The AECB requires licensees to identify the safety-related items, activities and processes in accord with the definition, and reviews them for acceptance.

The QA program includes the controls and details of how licensees will manage, perform, and assesses the work they do in each life-cycle phase. This is fundamental because the life of projects crosses generations, and makes dependence on systematic processes for decisions, actions, and results a necessity. The QA program informs everyone involved about the following:

- organization structure
- functional responsibilities
- levels of authority
- methods of communication and decision-making

Such information is to be used by those managing, performing, and assessing the adequacy of work. It also includes management methods of control such as:

- planning
- training
- resource allocations
- work instructions and practices

As the licensee progresses from one phase to another, its organization and the methods to be used to process and control the performance of work will change. The licensee would have to describe these variations and modify its management processes accordingly. The licensee in accordance with its QA program for the overall nuclear facility must perform oversight activities of various disciplines of the project to retain responsibility in all circumstances.

The QA program makes sure that during all of the life-cycle phases, work is planned, controlled, and carried out according to established:

- codes
- standards
- specifications
- instructions

It is binding on all personnel whose work on the nuclear project can affect nuclear safety. This includes the work performed by organizations that are not part of the licensee's organization. The QA program makes sure that the licensee's ultimate accountability for safety is passed down the chain of command through senior managers and line managers to the working level. Responsibilities at each level are developed, understood, and exercised so that each individual takes responsibility for the quality of the work he or she performs. These arrangements must be in place through all phases of the project from design to decommissioning.

13.3 METHODS USED FOR IMPLEMENTATION AND ASSESSMENT OF QA PROGRAMS

Separate from the internal reviews and audits carried out by the licensees, the AECB reviews in detail the documentation that communicates the requirements of the QA program to licensee personnel. When it is accepted, the AECB plans and carries out real-time audits to make sure that the licensee and other organizations are complying. These audits are performance-based. They assess the following activities by the licensee during each particular phase of work for the facility to make sure that safety is the highest priority:

- work methods
- management processes and results
- overall compliance

When deficiencies are detected, the licensee is notified and is required to correct them. The AECB produces detailed reports of the audit findings and forwards them to the licensee for action and reply. The AECB may decide an enforcement action is appropriate.

The licensee's QA programs are also subjected to two levels of audit by its own management. The first level is bounded by the organizational lines that describe the phase or discipline. For example, the part of the licensee's organization that is responsible for the design of the nuclear facility conducts audits to make sure that technical requirements are being met consistently, and that management processes are being followed. In addition, a second level of audit and review provides the corporate oversight needed to make sure overall QA policies are satisfied. This is necessary particularly when interfaces between technical disciplines need to be bridged, and responsibilities need to be defined and turned over from one organization to another as the work progresses from one phase, discipline, or organization to the next. When licensees detect a deficiency, they must determine the extent of the problem, and the effect on safety. They must identify the breakdown in the management process that was the underlying cause of the problem and correct it.

Similarly, when the licensee has to rely on other organizations to carry out work, the licensee makes sure that QA requirements are passed on to them and are met. The licensee determines that these organizations have an acceptable QA program before work is contracted to them. Then as the work progresses, the licensee conducts real-time reviews, audits, and inspections to make sure that the work being done meets requirements. Their frequency is determined by factors such as safety significance and the performance of the contractor.

13.4 REGULATORY CONTROL ACTIVITIES RELATED TO QA

AECB activities related to QA are described in Article 13.2.

The regulations that will accompany the promulgation of the new NSC Act will require the implementation of QA programs during the life cycle of the nuclear facility. An application to construct a nuclear facility must include the QA program for its design. The licensee must also submit QA programs for activities in various phases of the life-cycle of a nuclear facility before these phases begin. These submissions must be provided on a schedule that will allow detailed reviews by the nuclear regulator. It is likely that the nuclear regulator will require compliance with licence conditions that will correspond to each of these phases.

ARTICLE 14
ASSESSMENT AND VERIFICATION OF SAFETY

14.1 THE LICENSING PROCESS AND SAFETY ANALYSIS REPORTS FOR THE DIFFERENT STAGES OF A NUCLEAR POWER STATION

The Atomic Energy Control Board (AECB) licensing process, including reporting requirements, is described in Article 7.3 for the stages of site acceptance, construction approval, and operating licence. The operating licence renewal, and AECB staff compliance activities and change approvals are described in Article 8.1.

AECB reporting requirements are given in Regulatory Document R-99 (see Attachment 7.9), and summarized in Article 9.2. These include a preliminary Safety Report required for the construction approval and a final Safety Report required for the operating licence. The Safety Report includes:

- a description of the design and its major safety features
- the safety analysis required to demonstrate the effectiveness of the special safety systems under normal and abnormal conditions

Special safety systems standards are described in Regulatory Documents R-7, R-8, and R-9 (see Attachment 7.9). Most of the postulated accidents identified in Consultative Document C-6 (see Attachment 7.10) have to be analysed to demonstrate compliance with the requirements of the document. There may be reasons granted that would exempt some of C-6 postulated accidents from being analysed. These reasons are usually related to the identification of more stringent demands placed on the safety systems by other analysed accidents.

The main features of the Canadian licensing process include the following:

- The licensing process is initiated at an early stage in the life of a new project. Assessment of the design of the nuclear facility begins well in advance of receipt of the application for site approval.

- Comprehensive safety evaluation is carried out by the AECB technical staff for each stage of licensing.

- When construction approval is granted, an AECB project office is established, with full-time resident staff, at the reactor site. This enables surveillance to be maintained over safety-related activities throughout construction, commissioning, and operation.

- As part of the licensing process, the AECB requires that the public be adequately informed about any proposed licensing activity. For this reason, applicants are required to carry out public information programs that may involve:

  - publicly announcing the intent to seek a licence;
  - publishing reports for public review;
  - holding meetings to allow public participation in the review process.

- The staff of the AECB will not recommend approval for licensing until the requirements of all federal, provincial, and municipal agencies involved in the review of an application have been met. Such requirements are further elaborated on in Article 7.3.

- Review, evaluation and monitoring by the AECB is continuous to make sure that safety performance standards continue to be met. AECB staff activities are described in Articles 7.4 and 8.1.

14.2 A SUMMARY OF ESSENTIAL GENERIC RESULTS OF CONTINUED MONITORING AND PERIODIC SAFETY ASSESSMENTS OF NUCLEAR POWER STATIONS

Continued monitoring and periodic safety assessments of Canadian nuclear power stations are performed by station operators and the AECB. The safety assessments and their major results are discussed in Article 6.2. These assessments are divided into:

- assessments following major incidents
- assessments in response to operating experience
- other safety assessments that are part of the periodic safety assessment efforts

See Articles 6.2 and 6.3 for more information about assessments, results, and corrective actions for individual stations.

In addition, AECB site staff performs routine inspections as described in Article 7.4. Such inspections include:

- rounds
- assessments of operating practices
- audits
- system inspections

AECB staff also reviews station operation on a periodic basis as part of operating licence renewals. These renewals include the following assessments:

- reliability and risk assessment
- human factors assessment
- pressure-retaining component surveillance and assessment

More details about AECB duties related to monitoring and assessments are discussed in Articles 7.4 and 8.1.

Self-assessments performed by the utilities are essential for continuous improvement of the safe and reliable operation of nuclear power stations. The assessment process consists of the following:

- worker assessment
- management assessment
- independent assessment

Worker assessment is a continuous activity covering day-to-day, week-to-week experience. It provides the general basis, over a period of time, for the management assessment activity that is a periodic process and is the basis for continuous improvement. Management needs to know how effective their assessments are in practice. To accomplish this, they arrange to have independent assessments done from time-to-time for validation purposes.

The self-assessment program is an ongoing process that determines how well the station owners/operators are providing leadership to meet requirements and expectations. All levels of management are required to do these self-assessments with the emphasis on the allocation of human and other resources to achieve the organizational goals and objectives.

At the senior management level, it is necessary to do a self-assessment in order to determine whether the overall performance effectively focuses on meeting strategic goals including safety.

Line management reports, which summarize both assessment and regulatory feedback categories, are sources of information on the overall performance of the organization. They also provide the basis for targeting improvement actions. Line management will, in addition, rely heavily on surveillance and review of worker performance. This could include, but should not be limited to:

- surveillance of items, services, and processes;
- review of design documents and validation;
- reviews of procedures and records;
- observation of independent assessments;
- regular facility tours.

At the supervisory level, direct observation of the work, supported by inspection and testing, must be routinely carried out. The organization must have a firm foundation of:

- procedural adherence
- self-critical culture
- questioning attitudes
- willingness to self-identify mistakes

Without this basis, the ideas of performance-based assessments and self-assessment programs will not succeed.

Ontario Hydro Nuclear (OHN) has started developing self-assessment programs and is in the process of introducing an OHN Standard on the subject. OHN's programs have the following objectives:

- to identify and measure the extent to which important standards and programs are met;
- to measure the effectiveness of corrective actions;
- to identify areas requiring additional management attention.

The development and management of self-assessment guides are the responsibility of the Accountable Manager for the specific program. The effectiveness of the self-assessment guides shall be reviewed annually by the Management Audit Department and reported to the Director.

AECB review has found that OHN has yet to establish a self-critical culture that readily identifies problems and errors. An effective corrective action and trending capability is also needed to detect lower level precursors and guard against recurrence or escalation of problems.

New Brunswick Power and Hydro-Québec have also started developing self-assessment programs. This is part of the Point Lepreau Performance Improvement Program (PIP), and the Gentilly-2 Safety Culture Self-assessment, as described in Article 6.2.3.

14.3 SAFETY VERIFICATION PROGRAMS IN EFFECT SUCH AS PREVENTIVE MAINTENANCE, IN-SERVICE INSPECTION OF MAIN COMPONENTS, AGEING PROCESSES EVALUATION

14.3.1 Inspection and Maintenance Programs

Operating licences issued by the AECB require the operator to:

- regularly perform tests and inspections on systems, equipment, and components to confirm availability;

- perform maintenance of such a standard that, in the opinion of the AECB, the reliability and effectiveness of all equipment and systems as claimed in the Safety Report are assured. To satisfy this requirement, the licensee must make sure that the effects of ageing degradation of components are managed in such a way that the safety of the station is not compromised, and that the claims of the station's Safety Report are upheld throughout the station's operational life.

An effective maintenance program will include the following features at a minimum:

- an effectively planned maintenance schedule, and a system for making sure that maintenance is executed according to the schedule;

- a system for recording deficiencies or planned maintenance, and initiating and controlling the work;

- a system to record, control, and authorize temporary changes to equipment;

- a process of inspection for important systems which contain pressure-retaining components, for example:

  - heat transport system,
  - containment,
  - secondary-side systems.

  These inspections are governed by the standards of the Canadian Standards Association (CSA), and contain requirements for review and approval by the AECB of any inspection findings that may indicate a significant flaw in a pressure-retaining component (periodic inspection program and in-service inspection program).

Since the mid-1980s, the AECB has paid increasing attention to the standards of operation and maintenance at CANDU stations. Its field project offices developed and, for several years now, have routinely conducted a structured compliance inspection program that involves field tours of the station and detailed system inspections.

14.3.2 Evaluation and Management of Ageing

Recognizing that the effects of ageing degradation of critical systems, structures and components can result in design safety margins being diminished and safety analyses being invalidated, the AECB has embarked on the development of a regulatory position on requirements for the management of ageing.

In 1990, the AECB required that each of its licensees submit for its review a summary of the means by which it is assured of the continued safe operation of its nuclear power station as it ages. The notice requested the licensees to address the following:

- the continued validity of steady state and dynamic analyses of the station, where key characteristics, such as heat transfer rates and flow-rates, have changed;
- the scope of the review of degradation mechanisms that could impact significantly on safety, and which might therefore require changes to surveillance and testing programs;
- the continued validity of reliability assessments of special safety systems, safety-support, and safety-related systems in the light of known or anticipated changes in component failure rates;
- the adequacy of the planned maintenance program.

From subsequent communications with the licensees, the AECB concluded that none of them had the necessary managed processes in place to provide adequate assurance of continued safe operation of their stations in the long term.

In 1991-92, the AECB issued draft recommendations for a regulatory position on requirements for the management of ageing. The program's fundamental requirement is to ensure that degradation of nuclear power plant systems, structures, and components, due to ageing is managed such that their contribution to the risk to public, worker safety, and to the environment, from operation of the nuclear power plant, remains within the bounds claimed and accepted as the licensing basis for the facility.

It was recommended that the program should be auditable and provide for the effective management of:

- ageing degradation of any component that could increase the probability or consequences of process system failures;
- ageing degradation of any safety-support or other safety-related system that could render a special safety system less effective or less reliable;
- ageing degradation that causes key system parameters such as flow rates, heat transfer rates, and pressure drops, to change to the extent that they exceed the limits assumed in the Safety Report.

Canadian nuclear power station owners have made a good start to the development of ageing management programs. Some have many of the required managed processes already in place. However, their overall life management programs do not yet provide the focus on safety with regard to identification of critical components and required performance standards needed to meet the proposed regulatory requirements.

As nuclear reactors age and critical components start to deteriorate, there is an increasing need for the regulator to assess the basis of acceptability of continued operation. There is also the need to address the issue of changing standards for nuclear safety as technology advances and public expectations increase. With its non-prescriptive regulations and two-year licence renewal cycle, the AECB has flexibility to adjust to these changes. However, the AECB recognizes that the present safety review process does not provide the level of assurance needed in the continued safe operation of nuclear power stations. In addition to the changes introduced over the past few years, broader and systematic changes to the regulatory process have now begun. This includes the development of new regulatory documents to define requirements for Probabilistic Safety Analysis (PSA).

14.4 REGULATORY CONTROL ACTIVITIES RELATED TO THE ASSESSMENT AND VERIFICATION OF SAFETY

The regulatory activities and duties of AECB staff are addressed in Articles 7.4, 7.5, 8.1, and 14.3. Activities related to the assessment and verification of safety include the following:

- site staff compliance inspections that include the following activities as detailed in Article 7.4:

    - rounds,
    - assessment of operating practices,
    - audits,
    - system inspections.

- providing change approvals according to the licensee control program for temporary and permanent changes to:

    - equipment,
    - procedures,
    - documents.

- developing performance indicators to give early warning for signs of deterioration.

- annual review of station operation, and annual review meetings with station management.

- operating licence renewals and the related safety reviews and assessments, which include:

    - annual AECB review of station safety performance,
    - review of significant events reported by the licensee,
    - reliability and risk assessment,
    - human factors,
    - pressure-retaining component surveillance/assessment,
    - quality assurance,
    - radiation protection program,
    - environmental impact,
    - operator certification,
    - training program.

- head office staff reviews of documents related to the following activities, to assess the quality of station systems and procedures in order to assure safety:

    - design,
    - construction,
    - commissioning,
    - safety analyses,
    - radiation protection provisions.

Head office staff also coordinate the review and resolution of generic action items (see Annex 6.1).

- enforcement of regulations by a variety of actions ranging from recommendations to prosecutions (more details in Article 7.5).
- development of regulatory positions for the management of ageing and the use of PSA (as addressed in Article 14.3).

ARTICLE 15
RADIATION PROTECTION

15.1 A SUMMARY OF THE NATIONAL LAWS, REGULATIONS, AND REQUIREMENTS DEALING WITH RADIATION PROTECTION AS APPLIED TO A NUCLEAR POWER STATION

Nuclear power stations in Canada are regulated by the Atomic Energy Control Board (AECB) in accordance with the requirements stipulated in the Atomic Energy Control (AEC) Regulations. The AECB is given the power by the AEC Act to monitor and enforce the AEC Regulations.

Radiation protection requirements are included in a number of sections in the AEC Regulations. These requirements were adopted from the recommendations of the International Commission on Radiological Protection (ICRP) of the early 1950s. The radiation dose limits for workers and members of the public are expressed in terms of maximum permissible doses to different critical organs, for example:

- whole body
- gonad
- bone marrow
- thyroid
- lung
- skin

There is a special maximum permissible exposure for radon daughters. The annual maximum permissible dose limit for the worker and member of the public are 50 mSv and 5 mSv respectively. The concept of "As Low As Reasonably Achievable" (ALARA) is not explicitly included in these regulations.

A number of amendments were made to the AEC Regulations but no major change was made in the basic radiation dose limits. In 1978, an amendment was made to the regulation for radiation protection. This amendment changed the maximum permissible exposures to radon daughters for workers and any members of the public. In 1985, an amendment made clear that the maximum permissible doses include radiation doses from sources both inside and outside the body.

With the change of the AEC Act to the new Nuclear Safety and Control (NSC) Act in the near future (see Articles 7.1 and 8.2), there will be a number of new regulations under the new act including one on radiation protection. The new radiation protection regulations will include a majority of the ICRP-60 (1991) recommendations for dose limits. They will also include ICRP-65 (1994) recommendations for the dose limits for exposure to radon progeny. The critical organ concept will be replaced by the effective dose equivalent concept.

The dose limits to control stochastic effects will be expressed in terms of effective dose and units of mSv. The dose limits to control deterministic effects are expressed in terms of equivalent doses and units of mSv.

The proposed dose limits for workers are an effective dose of 100 mSv for five years with an allowance of a maximum dose of 50 mSv in any single year within that five-year dosimetry period. For members of the public, the dose limit will be an annual effective dose of 1 mSv.

15.2 THE IMPLEMENTATION OF NATIONAL LAWS, REGULATIONS, AND REQUIREMENTS RELATED TO RADIATION PROTECTION

15.2.1 Dose Limits

Many nuclear power stations in Canada have developed individual company policies, regulations, and procedures based on the requirements outlined in the AEC Regulations (see Articles 7.2 and 7.3 for more information). Station operators will modify their policies, regulations, and procedures in accordance with the new NSC Act and the accompanying regulations for radiation protection. The management and workers in these nuclear power stations are expected to meet the requirements in these company policies, regulations, and procedures. Companies also carry out internal and external peer reviews on a regular basis to make sure that their operations meet the national laws on radiation and environmental protection. The performance of the nuclear power stations is continuously monitored by AECB staff.

To make sure that no worker exceeds the national dose limit for workers, nuclear power stations have established upper limits for station operation based on a fraction of the dose limits in the AEC Regulations. They have also developed working processes for the control of doses to workers (for example, special work plan and work procedures for high hazard work). They carry out dose budgeting for each operation.

15.2.2 Fulfilment of Conditions for Radioactive Release

To make sure that the radioactive releases from the nuclear power station do not impose undue risk on the environment, nuclear power stations have established effective station control for effluent release. These controls keep effluent release below the 1% of their respective Derived Release Limits (DRL). A DRL is an effluent release limit for a particular station. If the station exceeds the limit, members of the public who live near a nuclear power station may exceed their dose limits.

15.2.3 Steps Taken to Make Sure that Radiation Exposure Is Kept As Low As Reasonably Achievable

To make sure that radiation exposure is kept as low as possible, the nuclear power stations have established various programs. These programs include but are not limited to:

- management control over work procedures;
- personnel qualification and training;
- control of occupational and public exposure to radiation;
- planning for unusual situations.

15.2.4 Environmental Radiological Surveillance

The nuclear power stations have established different programs to monitor the effect of station operations on the environment. A federal agency, Health Canada, also carries out programs around all nuclear power stations to monitor the environment.

Health Canada began monitoring for environmental radioactivity in 1959. The initial role was to monitor fallout from atmospheric nuclear weapons testing. The current program consists of 25 monitoring stations for external gamma, radioactive aerosols and radioactivity in precipitation (see Figure 15.1). These were augmented in a few locations with drinking water and milk sampling. The monitoring of these latter stations has been essentially ceased, although the capability has been maintained.

Health Canada operates a second network, the Reactor Monitoring Network. In 1962, with the advent of the Canadian nuclear power program, Health Canada implemented a reactor monitoring network at the Chalk River Laboratories and the Ontario Hydro Nuclear's (OHN) NPD reactor site in Rolphton. With the enlargement of the nuclear program, the reactor network expanded to 13 stations, measuring gamma doses (TLD), tritium in air and radioactivity in drinking water around the nuclear sites (see Figure 15.2, where one location on the map may indicate more than one monitoring station). The current staff complement to operate both networks is six person years with an annual operating budget of approximately $315,000.

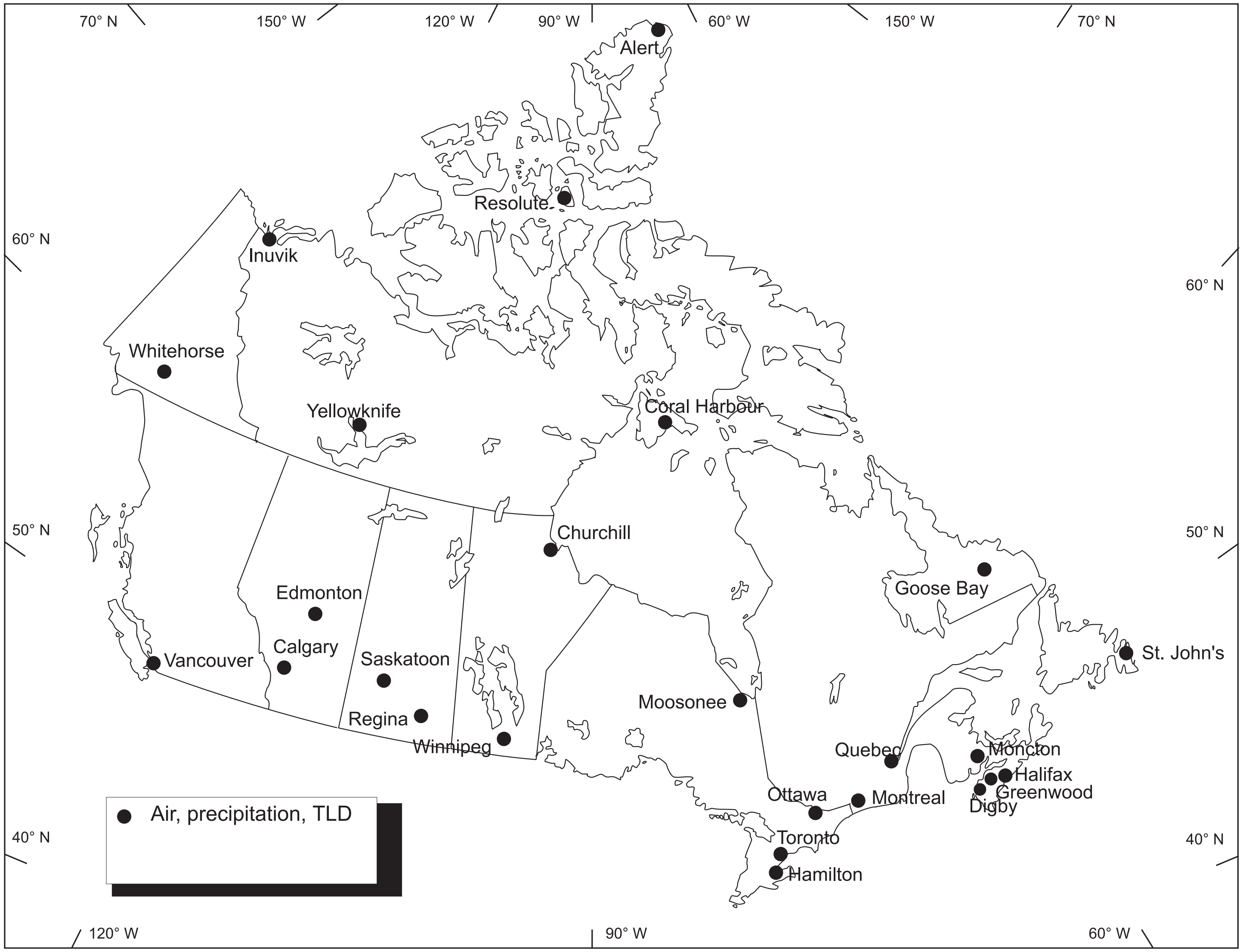

Figure 15.1: National Radioactivity Monitoring Network

Figure 15.2: Nuclear Reactors and Drinking Water Monitoring Sites

The present monitoring networks are maintained under the general enabling legislation of the Department of Health Act (1996). These networks make available specialist knowledge that includes the health effects of environmental radioactivity. In 1984, the federal cabinet appointed Health Canada the lead department for federal response to nuclear emergencies. The network also supports Canada's international obligations in providing early notification, information, and assistance in the event of a nuclear accident.

There have been a number of special projects to assess the impact of natural radioactivity, technologically-enhanced natural radioactivity, and releases from foreign nuclear facilities on Canadians. These projects include:

- the monitoring of Cs-137 in caribou and humans in northern Canada
- the across-Canada radon monitoring program
- the monitoring of drinking water in the uranium mining and processing communities of Port Hope and Elliot Lake, Ontario
- the monitor of food imports and exports

15.3 REGULATORY CONTROL ACTIVITIES RELATED TO RADIATION PROTECTION

AECB staff carries out many regulatory control activities related to radiation protection throughout the year. They develop and prepare regulatory guidance documents and programs for radiation protection, environmental protection, and emergency preparedness. These documents help nuclear power station licensees to interpret the requirements in the AEC Regulations (for example, Regulatory Standard, S-106 entitled: "Technical and Quality Assurance Standards for Dosimetry Services in Canada", issued in 1998. Copy of S-106 is given in Attachment 15.1).

To verify compliance with the requirements in the licences and the AEC Regulations, AECB staff:

- reviews documentation and operational reports submitted by licensees;
- conducts health physics appraisals;
- conducts appraisals of licensee radiation and environmental protection; programs, emergency preparedness programs, and other programs as required.

Staff also:

- monitors and evaluate radiation and environmental impacts of these licensed activities;
- reviews documentation and applications submitted by licensees and dosimetry service proponents;
- conducts on-site appraisals of dosimetry service applicants and prepares appraisal reports;
- recommends approval of licence applications.

If a significant incident occurs that involves radiation and environmental protection, staff carries out assessments and investigations.

Health Canada, under its legislative mandate, prepares guidelines for drinking water and food for normal and emergency conditions. Since drinking water guidelines are officially a provincial jurisdiction, the guidelines prepared by Health Canada can be adopted by the provinces with or without modifications.

ARTICLE 16
EMERGENCY PREPAREDNESS

16.1 A GENERAL DESCRIPTION OF LAWS, REGULATIONS, AND REQUIREMENTS FOR ON-SITE AND OFF-SITE EMERGENCY PREPAREDNESS

Canadian legislation empowers the Atomic Energy Control Board (AECB), using a comprehensive licensing and compliance system, to impose conditions on its licensees to assure health, safety, security, and protection of the environment. At the same time, it obliges the nuclear industry to protect its workers and the public from unacceptable levels of radiation. As a condition of licensing, the AECB requires applicants to assess the implications of their proposed activities, and to provide contingency plans to cope with potential accidents. After the plans have been reviewed and accepted by the AECB, they become binding upon the licensee. As discussed in Article 7.3, Table 7.1, the on-site emergency procedures and the provincial emergency plan are part of the regulatory requirements for the station operating licence.

Consequently, the nuclear industry in Canada, including non-power reactor licensees, retains the primary responsibility during both normal and abnormal or emergency situations for safeguarding its workers, the public, and the environment in accordance with national legislation and AECB licence requirements. Potentially, this collective responsibility encompasses a wide range of contingency and response measures to prevent, correct, or eliminate:

- accidents
- spills
- abnormal situations
- emergencies

The nuclear emergency planning includes on-site and off-site emergencies as described below:

- On-site nuclear emergencies are those that occur within the physical boundaries of a Canadian nuclear facility that is licensed by the AECB pursuant to the Atomic Energy Control (AEC) Act and regulations thereunder.

- Off-site nuclear emergencies are those that require support from the site and all levels of government, provided under the Federal Nuclear Emergency Plan (FNEP) of Health Canada to a Canadian province or territory as a consequence of a domestic, trans-boundary (for example, Canada and the United States (US)), or international incident.

The federal government is responsible for:

- the development, control, and regulation of peaceful uses of nuclear energy;
- managing nuclear liability;
- coordinating with, and providing support to, provinces in their response to a nuclear emergency;
- liaison with the international community;
- liaison with diplomatic missions in Canada;
- assisting Canadians abroad;
- coordinating the national response to a nuclear emergency occurring in a foreign country.

As a result, off-site planning, preparedness, and response to nuclear emergencies is a multi-jurisdictional responsibility shared by all orders of government.

To the extent possible, the federal government's emergency planning, preparedness, and response are based on the "all-hazards" approach. Because of the inherent technical nature and complexity associated with a nuclear emergency, hazard-specific planning, preparedness, and response arrangements are required. These special arrangements, which are one component of the larger federal emergency preparedness framework described in Part 1 of Annex D of the National Support Planning Framework, constituted the FNEP. The FNEP describes the federal government's preparedness and coordinates response to a nuclear emergency.

Under the common administrative framework of Canada's FNEP, the development and implementation of emergency preparedness and response plans to off-site nuclear emergencies, is primarily a provincial responsibility. However, there are direct inputs from the local government, the nuclear facility, and the federal government departments and agencies. This allows the various jurisdictions and organizations that have responsibilities for aspects of nuclear emergency preparedness to discharge their responsibilities in a cooperative, complementary, and coordinated manner.

16.1.1 Overview of the Federal Nuclear Emergency Plan in Relation to Emergency Preparedness Measures

Within the FNEP, a nuclear emergency is defined as an event that has led or could lead to a radiological threat to public health and safety, property, and the environment.

The FNEP contains:

- outlines of the federal government's aim, authority, emergency organization, and concept of operations for dealing specifically with the response phase of a nuclear emergency;
- a description of the framework of federal emergency preparedness policies, the planning principles on which the FNEP is based and the links with other specific documents of relevance to FNEP;

- a description of the specific roles and responsibilities of participating organizations that are involved in the planning, preparedness, or response phases of a nuclear emergency;
- specific Provincial Annexes that describe interfaces amongst federal and provincial emergency management organizations, and the arrangements for a coordinated response and the provision of federal support to provinces affected by a nuclear emergency.

The FNEP is intended to complement the relevant nuclear emergency plans of other jurisdictions inside and outside Canada. It describes the measures to be followed by the Government of Canada to manage and coordinate federal response activities to nuclear emergencies that could affect Canada. The FNEP would be activated if federal support to a Canadian province or territory was required as a consequence of any domestic, trans-boundary, or international incident.

The FNEP is issued under the authority of the national Minister of Health. This authority derives from a designation received from the Prime Minister of Canada, from the Emergency Preparedness Act, and from federal policies on emergencies and emergency preparedness.

16.1.2 Types of Nuclear Emergency Events

There are four types of nuclear emergency events covered by the FNEP. These categories are:

- an event at a nuclear power station in Canada or in the US along the Canada-United States border
- an event involving vessels visiting Canada or in transit through Canadian waters
- an event involving a nuclear power station in the southern US or in a foreign country
- other serious radiological events

In addition to the events listed above, the FNEP includes appendices that summarize the on-site emergency notification classifications adopted by Chalk River Laboratories, all nuclear power station installations in Canada and selected nuclear power stations in the US (Fermi-2, Washington, nuclear power plant,) for both airborne and liquid releases. Appendix 5 of the Plan is a facsimile of the International Nuclear Event Scale (INES).

The scope of the FNEP excludes:

- circumstances of war, such as the military use of nuclear weapons against North America;
- events that may pose a limited radiological threat, and consequently are not expected to exceed the response capabilities of regulatory, local, or provincial authorities;

- management and coordination of the Government of Canada's actions during the recovery phase. If federally assisted recovery actions are required as a consequence of a nuclear emergency, responsibility for these actions is to be assigned to a specific minister of the Federal Crown, during or immediately following the response phase of the nuclear emergency.

16.1.3 Dealing with Emergencies Under the Federal Nuclear Emergency Plan

Nuclear emergencies for which the FNEP could be activated cover a wide spectrum of emergency event probabilities and potential severities. Consequently, the owners/ operators of nuclear power stations in Canada are required by federal regulatory authorities and other jurisdictions to develop and maintain contingency plans to cope with all on-site emergencies.

The provinces, in cooperation with local jurisdictions, have established procedures to deal with any significant off-site nuclear impacts. Typically, these provincial plans provide for urgent protective actions if required. These procedures include:

- limiting access to the affected zone;
- providing temporary shelter to the affected population;
- blocking thyroid uptake of radiation;
- evacuating buildings or premises in areas near the nuclear power station.

The plans also recognize that ingestion control measures (for example, effecting a quarantine of farm animals, banning the sale of affected food, or restricting the use of affected drinking water) for a larger area could be necessary.

The provinces of Quebec, Ontario, Nova Scotia, New Brunswick, and British Columbia are the Canadian regions most likely to be affected by a nuclear emergency, as defined in the FNEP. This higher probability is due to their closer proximity to American and Canadian nuclear power stations, and the existence, in some cases, of nuclear power stations within their boundaries, or having ports which are visited by nuclear-powered vessels.

A nuclear emergency involving Nuclear Powered Vessels (NPVs) visiting Canada in transit through Canadian waters could create impacts similar to, but much less extensive than, those resulting from an emergency at a nuclear power station. In particular, naval reactors have much lower power ratings than nuclear power stations, contain less radioactive material, and operate at low or zero power when docking or approaching port. Consequently, for planning purposes, the Canadian Department of National Defence assumes that urgent protective actions could be required in a 1-5 km zone around Canadian ports of call visited by NPVs. Although an ingestion exposure emergency planning zone is not considered essential for potential emergencies involving NPVs, food and soil sampling and analysis could be required to make sure and plan for the protection of the population living in the immediate vicinity. Presently, NPVs visit ports in Nova Scotia and British Columbia, however no vessels are authorized to carry significant quantities of fissionable substances from irradiated fuel as cargo through Canadian waters. Consequently, for nuclear emergency events involving maritime vessels visiting Canada or in transit through Canadian waters, the FNEP, or parts thereof, might be implemented if the agency responsible for leading the response in the affected province requests assistance.

As the Chernobyl accident demonstrated, a severe nuclear emergency at a major nuclear power station that is distant from Canada would have an effect, as small quantities of radioactive material might reach Canada. Although these materials could exist in detectable amounts, they would be unlikely to pose a direct (for example, from exposure to fallout) threat to Canadian residents, property, or environment. Consequently, Canada’s response under the FNEP to a nuclear accident at a facility in the southern US or in another foreign country would likely focus on the following:

- controlling food imported from areas near the accident;
- assessing the impact on Canadians living or travelling near the accident site;
- assessing the impact on Canada and informing the public;
- coordinating responses or assistance to foreign jurisdictions and organizations, national or international.

The potential severity of other serious radiological events, as defined in the FNEP, will depend on case-specific factors. For fixed facilities and materials in transit, appropriate responses to possible emergencies can be planned in some detail. In other situations, emergency planning can be complicated by factors such as the potential magnitude and diversity of the radiation threat, the location of the source of the radiation, any impacts on essential infrastructures, and the speed at which related circumstances may evolve.

Some examples of potentially serious radiological events that would be covered by category 4 of the FNEP are:

- malevolent acts involving improvised nuclear or radiation dispersal devices
- use of conventional explosives on station known to use or store radiation sources
- the reentry of a nuclear powered satellite such as the 1978 crash of COSMOS 954
- a severe earthquake that could damage a radiation source, and lead to the release of radioactivity

16.2 THE IMPLEMENTATION OF EMERGENCY PREPAREDNESS MEASURES, INCLUDING THE ROLE OF THE REGULATORY BODY AND OTHER ENTITIES

Health Canada, the federal ministry of health, is the lead agency for all matters related to FNEP. In any nuclear emergency that affects Canada, the FNEP, or parts of it, may be implemented. In general, this activation will occur in support of a province, or in support of a federal department or agency that is leading the federal response to assure public health and safety.

There are 19 federal participants in FNEP. In keeping with the FNEP, federal policies, and Canadian legislation, these participants are also responsible for independently developing, maintaining and implementing their own nuclear emergency response plans. The AECB, in its role as national nuclear regulatory body, has its own nuclear emergency response plan. Similarly, Transport Canada administers the Canadian Transport of Dangerous Goods Regulations and operates the Canadian Transport Emergency Centre (CANUTEC) to make sure that hazardous substances are transported safely and to help emergency response personnel handle related emergencies, including those involving radioactive materials. The AECB and Transport Canada cooperate in emergencies and incidents involving radioactive materials in accordance with the FNEP, relevant federal legislation, and formal administrative arrangements.

16.2.1 Measures for Informing the Public During a National Nuclear Emergency

The FNEP describes how overall coordination is to occur in the event of a national nuclear emergency in Canada. Information is to be provided at the national level to members of the media and the public through a central point of contact; the Public Affairs Group (PAG). Under the direction of the National Coordinator, the PAG serves as the federal coordination point for the collection, generation, and distribution to the public and the news media of information concerning the emergency.

The PAG is made up of representatives of organizations that have defined responsibilities within the structure of the FNEP, in accordance with procedures set out in the National Emergency Arrangements for Public Information. The departments and agencies listed in the FNEP may be represented on the PAG, along with other organizations and governments involved in a specific nuclear emergency.

The PAG works closely with members of the Federal Regional Organization, other federal departments and agencies, and provincial authorities to coordinate and harmonize public information activities. For example, during a nuclear emergency, information centres may be established in strategic locations in response to local needs. In keeping with the cooperative spirit of FNEP and provincial and territorial emergency plans, these centres might be staffed by representatives from all levels of government and industry.

16.2.2 Provincial Emergency Plans that Cover Nuclear Power Station Installations

Governments of the Canadian provinces and territories have established their own individual plans, and customized them to address their specific needs. Typically, their administrative structure includes an Emergency Measures Organization (EMO) or equivalent to cope with a wide range of potential or actual emergencies in accordance with defined plans and procedures. In addition, those provinces that are hosts to major nuclear installations, such as nuclear power stations, have nuclear-specific emergency plans in place.

The provincial and territorial emergency preparedness plans provide for coordination with other relevant jurisdictions and organizations. They anticipate the involvement and support of the federal government at the national level, the involvement and support of both municipal and civic governments at the local level, and extensive participation by departments and agents of all levels of government.

Nuclear power station installations are located in the Canadian provinces of Ontario, Quebec, and New Brunswick, with the majority located in Ontario. Each of these provinces have off-site emergency preparedness plans in place that will deal with emergencies that may occur at any of their nuclear power station installations. See Annex 16.1 for a brief summary of the provincial plans.

16.2.3 Role of the Regulatory Body

The AECB participates in nuclear emergency planning, preparedness, and response activities as part of its responsibilities according to Canadian legislation.

During a nuclear emergency in Canada, the AECB would continue in its regulatory role, as anticipated in the FNEP and the AECB Emergency Response Plan. The AECB has clearly defined roles within the context of the FNEP. For example, it is a core member of each of the FNEP's four organizational groups (Coordination, Operations, Technical Advisory, and Public Affairs), and participates in emergency planning activities with other FNEP core agencies.

Since the AECB's regulatory obligations extend to a wide range of circumstances, stations, activities and materials, it must plan for its possible involvement in a similarly diverse range of emergency scenarios.

In keeping with national policy, and notwithstanding its participation in the FNEP, the AECB has developed the Emergency Response Plan, which will be fully implemented by the end of 1998. In support of this initiative, the AECB has constructed an Emergency Operations Centre (at its headquarters in Ottawa) to enhance its ability to respond to nuclear emergencies. This facility is now fully operational, and is being used during ongoing FNEP and AECB drills and training exercises to make sure and confirm nuclear emergency preparedness.

The AECB has established various technical and administrative arrangements in the interests of emergency preparedness. These arrangements form part of the AECB's emergency response plan. They include bilateral cooperation agreements with other national and international jurisdictions, as well as AECB operation of a Duty Officer Program whereby anyone can seek emergency information, advice, or assistance 24-hours a day for actual or potential incidents involving nuclear materials or radiation.

See Annex 16.1 for descriptions of the AECB emergency response plan, and its participation in tests and drills.

16.3 TRAINING AND EXERCISES

Canada conducts national-level, no-fault exercises, every three or four years to test and evaluate national contingency plans that are designed to deal with the effects of emergencies that it could face. These tests are part of the family of exercises termed Canadian National Exercises (CANATEX). The third exercise in this series, CANATEX-3, is scheduled for early 1999. It will test Canada's response to a (simulated) major accident at a nuclear power station installation.

Canada also participates in International Nuclear Exercises (INEX) which are organized and coordinated by the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD). The CANATEX-3 exercise will also be conducted as an INEX drill, in order to test the emergency response systems of participating countries.

16.4 INTERNATIONAL ARRANGEMENTS, INCLUDING THOSE WITH NEIGHBOURING COUNTRIES, AS NECESSARY

Canada is a signatory of the following three international emergency response agreements:

CANADA-US JOINT RADIOLOGICAL EMERGENCY RESPONSE PLAN (1996)
The Joint Plan focuses on emergency response measures of a radiological nature rather than generic civil emergency measures. It is the basis for cooperative measures to deal with peacetime radiological events involving Canada, the United States, or both countries. Cooperative measures contained in the FNEP are consistent with the Joint Plan.

CONVENTION ON ASSISTANCE IN THE CASE OF A NUCLEAR ACCIDENT OR RADIOLOGICAL EMERGENCY (1986)
Canada is a signatory of this international assistance agreement which was developed under the auspices of the International Atomic Energy Agency (IAEA). The purpose of the agreement is to provide for cooperation between signatories to facilitate prompt assistance in the event of a nuclear accident or radiological emergency to minimize its consequences and to protect life, property, and the environment from the effects of radioactive releases. The agreement sets out how assistance is requested, provided,

directed, controlled, and terminated. This Convention has yet to be ratified pending a review of domestic implementing legislation.

CONVENTION ON EARLY NOTIFICATION OF A NUCLEAR ACCIDENT (1987)
Canada is a signatory of this international notification agreement which was developed under the auspices of the IAEA. The Convention defines when and how the IAEA should be notified of an event with potential transboundary consequences, or when and how the IAEA would notify the signatories of an international event which could have an impact in their respective countries.

ARTICLE 17
SITING

17.1 A DESCRIPTION OF THE LICENSING PROCESS, INCLUDING A SUMMARY OF THE NATIONAL LAWS, REGULATIONS, AND REQUIREMENTS RELATING TO THE SITING OF NUCLEAR POWER STATIONS

The initial stage of the licensing process in Canada is the site acceptance as described in Article 7.3. The siting related regulatory requirements are summarized as follows:

- "Letter of intent" issued by the applicant to Atomic Energy Control Board (AECB) describing the following:
  - the type, size, and major characteristics of the proposed nuclear power station;
  - the site and its location;
  - the basic organization of personnel, including identifying contact persons with whom the AECB staff will communicate.

- Detailed assessment of the impact of the nuclear power station on the environment prepared by the applicant, and submitted to the federal and provincial environmental agencies for their review, as required by the Canadian Environmental Assessment Act (CEAA).

- "Site Evaluation Report" submitted to AECB for site acceptance. The report is to demonstrate that the site characteristics for the nuclear power station are suitable for the following activities related to the facility:
  - design
  - construction
  - commissioning
  - operation

  Although the emphasis of the report is on the identification and investigation of those site characteristics which bear on safety, the report must also contain sufficient information on the conceptual design and operation of the facility.

- Public information meetings to be held by the applicant to explain the safety, environmental, social, and economic impacts of the nuclear power station; and also to allow the public to express its views and receive answers to its questions.

17.1.1 Criteria for Evaluating All Site-related Factors Affecting Safety

The criteria described in this area falls under two categories.

The first category is related to demographics. This includes ease of access/egress from the site and populated areas, and site location with respect to electrical grid lines and the security of electrical connections. Easy access (availability of appropriate highways and bridges) is required to facilitate resources movement in the event of:

- a contingency
- shift crew rotation
- emergency generator fuel oil delivery
- fire and security response
- potential emergency response evacuation

The second category is related to the site impact on the safety of the nuclear power station. This includes the site susceptibility to:

- flooding (storm surge, dam burst, etc.)
- hurricanes
- tornados
- ice storms or other severe weather
- earthquakes

This also includes the proximity of the site to one or more of the following facilities:

- railroad tracks with the possibility of derailments and the release of hazardous material
- flight paths for major airports with the possibility of airplane crashes
- toxic chemical plants with the possibility of toxic releases
- industrial parks with a neighbouring propane storage facility, or refinery, with the possibility of industrial accidents
- military test ranges with the possibility of stray missiles

17.1.2 Criteria for Evaluating the Nuclear Safety Impact of the Nuclear Power Stations on the Surrounding Environment and Population

The criteria described here is related to the safety impact of the nuclear power station on the environment and the population under normal and accident conditions. The impact on the environment includes effects on the:

- water supply
- air quality
- wildlife
- lakes
- rivers

Such factors are assessed in the environmental impact study that is performed to satisfy relevant provincial and federal laws.

The safety impact on the population is related to the population dose from single and dual failure events. Given that the station will perform as designed under accident conditions, factors related to the population are important to consider if the radiation dose limits set by regulations are to be met. Such factors include the number, nature (subdivision, rural, industrial, school, hospital, etc.), and distribution of population around the facility.

17.2 THE IMPLEMENTING PROVISIONS FOR THE ABOVE-MENTIONED CRITERIA

The above-mentioned criteria are implemented through the siting regulatory requirements as summarized in Article 17.1. The following documents are produced:

- “Letter of Intent” and Site Evaluation Report

  The site-related demographics and ease of access, its susceptibility to flooding, earthquakes, etc. are addressed in the letter of intent and in the site evaluation report to be produced by the applicant.

- Environmental Assessment Report

  The impact of the nuclear power station on the environment is addressed in the environmental assessment report.

- Safety Report

  The calculated population doses and the verification of nuclear power station design to meet its safety targets are reported in the Safety Report.

The above reports are reviewed by AECB staff and/or the federal and provincial environmental agencies for compliance with relevant regulations. The public information meetings, and the discussions that follow, also assist in judging the acceptability of the site as related to the above criteria.

17.3 THE ACTIVITIES RELATED TO MAINTAINING THE CONTINUED SAFETY ACCEPTABILITY OF THE NUCLEAR POWER STATION, TAKING INTO ACCOUNT SITE-RELATED FACTORS

The continued acceptability of the criteria mentioned in Article 17.1 is periodically verified. Possible changes to the site demographics, or significant changes to the understanding of local environment, include:

- discovery of new fault lines affecting seismicity at the site
- changes to man made neighbouring facilities such as newly constructed:
  - oil refinery
  - rail corridor
  - airport flight path
  - chemical plant

The above changes must be examined. This is achieved through activities that include the annual reviews of Emergency Preparedness response measures and the Annual

Security response reviews. This is in addition to the regulatory requirement of updating the Safety Report at least once every three years. The Safety Report contains sections on site-related factors including:

- demographics
- weather experience
- seismicity
- neighbouring facilities
- air and rail transport corridor activity

For radiation safety, environmental radiological monitoring programs have been instituted by each licensee to make sure there is continued safety acceptability at the nuclear power stations. The four primary objectives of these programs are:

- to confirm that emissions of radioactive materials are within the derived emission limits for specific nuclides or nuclide groups;
- to verify that the assumptions made in deriving station emissions limits remain valid;
- to permit an independent estimate to be made of doses to critical members of the public resulting from emissions;
- to provide data to aid in the development and evaluation of models that adequately describe the movement of radionuclides through the environment.

The licence for the operations of each nuclear power station requires the submission of an annual report to the AECB detailing the results of environmental radiological monitoring programs, together with an interpretation of the results and estimates of radiation doses to the public resulting from the operation of the station. The results from these monitoring programs are used to make sure that the public legal limit in Canada for effective dose from the operation of nuclear power stations is not exceeded.

The first step in an effective environmental monitoring program is the determination of background levels (for external gamma, tritium in the atmosphere, and tritium and gross beta activity in waterways) away from the influence of stations emissions. Background levels are determined through the analysis of samples taken throughout Canada.

The next step in an effective environmental monitoring program is to quantify the effects of facility emissions. Samples are taken and analysed at and around each station. Using the data obtained from these samples, an assessment is made of the impact that station operations have on the general population and the critical group dose (that is, the maximum dose to an individual member of the public). The impact of station operations is calculated using metabolic and food consumption data and dose conversion factors from a number of scientific sources. These calculations are based on conservative values, which make sure that the estimated doses will likely be greater than the actual average dose received by the members of the most exposed group. For example, the models typically assume that a person would:

- live just outside the station boundary;
- be at their residence 24 hours a day;
- drink only local water and milk;
- eat only local fish and produce.

The 1996 assessments (most current data available) indicate that the critical group doses due to the operation of each nuclear power station is between approximately 4 and 11 µSv/year. This is a very small percentage of the annual legal limit of 5,000 µSv/year[4] and the annual average total background radiation level of 2,100 µSv/year.

17.4 INTERNATIONAL ARRANGEMENTS WITH NEIGHBOURING COUNTRIES THAT COULD BE AFFECTED BY NUCLEAR POWER PROGRAMS IN CANADA

The following is a description of the consultation with the United States during the siting of Canadian nuclear power station installations:

The Canadian legislation and process, and, in particular, the Canadian Environmental Assessment Act and Regulations (CEAAR) and the federal Environmental Assessment and Review Process, do not oblige proponents of domestic nuclear power station installations that could affect the United States (US) to consult with US jurisdictions or the US public regarding the proposed siting of these installations.

Canada and the US, however, are signatories to the international Convention on Environmental Impact Assessment in a Transboundary Context (Espoo, Finland on 25 February 1991). If Canada and the US ratify this Convention, they will be bound by the provisions of the Convention. Ratification would oblige both Parties, with the "Party of origin":

- to "take all appropriate and effective measures to prevent, reduce, and control significant adverse transboundary environmental impacts of proposed activities", including the siting, construction and operation of nuclear power station installations;
- to "ensure that affected Parties are notified" of the proposed installation;
- to "provide an opportunity to the public in the areas likely to be affected to participate in relevant environmental impact assessment procedures regarding proposed activities, and to ensure that the opportunity provided to the public of the affected Party is equivalent to that provided to the public of the Party of origin";

4. The public legal limit in Canada for effective dose from the operation of nuclear power stations in 5,000 µSv/year (see also the introduction section, Table 1.1). The Atomic Energy Control Board is in the process of lowering the legal limit to bring Canada in line with the international value of 1,000 µSv/year as recommended by the International Commission on Radiological Protection (ICRP Publication 60).

- to include in the notification "information on the proposed activity, including any available information on its possible transboundary impact".

In addition to any involvement in the above Convention, Canada, the US, and Mexico are currently involved in discussions regarding development and implementation of a similar North-American tripartite agreement on Transboundary impacts.

The Government of Canada and the Government of the United States of America, in cooperation with State and Provincial governments, are also obligated to have in place programs for the abatement, control, and prevention of pollution from industrial sources which include measures to control the discharges of radioactive materials into the Great Lakes System. This is by virtue of the Great Lakes Water Quality Agreement of 1978, as amended by Protocol signed November 18, 1987[5].

The AECB and the U.S. Nuclear Regulatory Commission, as the national regulatory authority of their respective countries, have a long practice of cooperation and consultation since the 1950s. On August 15, 1996, they entered into a bilateral administrative arrangement for "cooperation and the exchange of information on nuclear regulatory matters". This commitment includes, to the extent permitted under laws and policies, the exchange of certain technical information that "relates to the regulation of the health, safety, security, safeguards, waste management, and environmental protection aspects of the siting, construction, commissioning, operation, and decommissioning of any designated nuclear facility" in Canada and the US.

5. See the IJC (International Joint Commission) publication of Febuary 1994.

ARTICLE 18
DESIGN AND CONSTRUCTION

18.1 A DESCRIPTION OF THE LICENSING PROCESS, INCLUDING A SUMMARY OF NATIONAL LAWS, REGULATIONS, AND REQUIREMENTS RELATING TO THE DESIGN AND CONSTRUCTION OF NUCLEAR POWER STATIONS

The safety objective of the CANDU design is to protect the public and station workers from adverse health effects due to the release of radioactive materials during normal station operation and during accident conditions. This is achieved through accident prevention and mitigation of the consequence of the accident, if one were to occur.

The design philosophy of CANDU nuclear power stations emphasizes:

- defence-in-depth
- separation of special safety systems and process systems
- a “fail-safe” mode of operation should a component or a system failure occur

One of the ways the Atomic Energy Control Board (AECB) makes sure that nuclear power stations do not constitute an undue risk to the public is by establishing safety requirements for these nuclear power stations. Regulatory Documents R-7, R-8 and R-9 (Attachment 7.9) specify the safety design standards for the special safety systems that include:

- two shutdown systems
- containment
- emergency core cooling system (ECCS)

Consultative Document C-6 (Attachment 7.10) specifies:

- reference dose limits for a wide range of events
- standards for the station owner to perform safety analyses to demonstrate that the acceptable consequences are not exceeded
- standards for the station owner to review the design of the station in a systematic and auditable manner to identify any other events of potential concern

The licensing process, including a summary of national laws, regulations, and requirements related to the design and construction of nuclear power stations are described in detail in Articles 7.2 and 7.3.

18.2 THE IMPLEMENTATION OF THE DEFENCE-IN-DEPTH CONCEPT IN ACCORDANCE WITH THE PRINCIPLE OF MULTIPLE SAFETY LEVELS, INCLUDING INTEGRITY OF BARRIERS, TAKING INTO ACCOUNT INTERNAL AND EXTERNAL EVENTS

18.2.1 The Concept of Defence-in-depth

The application of the concept of defence-in-depth in the CANDU reactor safety philosophy was discussed in the Introduction, part 4. For design, the defence-in-depth approach to make sure that there is a low probability of failures or combinations of failures that result in significant radiological consequences includes:

- conservative design and high quality of construction to provide confidence that abnormal operation or failures will be minimized;
- provision of multiple physical barriers for the release of radioactive materials to the environment;
- provision of multiple means for each of the basic safety functions, for example:
  - reactivity control,
  - heat removal,
  - confinement of radioactivity;
- the use of reliable engineered protective devices in addition to the inherent safety features;
- supplementing the normal control of the station by automatic activation of safety systems or by operator actions;
- provision of equipment and procedures to back up accident prevention measures in order to control the course and limit the consequences of accidents.

18.2.2 Barriers to Radioactive Releases

A nuclear power station contains radioactive material that could be a potential threat to the public. Most of this material, however, resides in the fuel elements. There are several barriers between this material and the public as follows:

- Uranium oxide fuel: The radioactive material is produced and trapped in the solid fuel. More than 99% of it remains in the fuel and is never released under normal conditions. Only a fraction of 1% of this radioactive material, produced during fission, escapes the uranium oxide and is then contained within the fuel element by the action of the fuel sheath.
- Fuel sheath: It retains the small amount of volatile fission products which escape the fuel matrix.
- Heat transport system: The fuel is contained in the Heat Transport System (HTS). An intact HTS retains the fission products even if sheath failures occurred and the small amounts of fission products (usually known as free-gap inventory) that reside between fuel and the sheath are released.

- Containment system: The next barrier to the releases is the containment system that contains radioactivity if both the fuel sheath and the HTS have failed.
- Exclusion zone: It provides atmospheric dilution of any fission product releases from the containment if all of the above barriers are breached.

18.3 THE PREVENTION OF ACCIDENTS AND THEIR MITIGATION

18.3.1 Prevention of Accidents

The basic safety objective of the CANDU design is to make sure that the risk to public health is limited. Radioactive material trapped in the fuel can only be released to the public if the barriers, discussed above, are accidentally breached. The first line of defence is to prevent accidents. Accident prevention is incorporated into the CANDU design by providing:

COMMITMENT TO QUALITY

- high level of quality during all aspects of the project
- strict quality control during manufacturing and installation
- use of proven components
- use of well-trained staff
- periodic inspection and testing of components and systems
- safe and efficient operation within the operating envelope
- high level of automation to reduce the risk of operator errors

REDUNDANCY

Redundancy is the use of two or more components or systems when each are capable of performing the necessary functions. System redundancy is achieved by having independent systems (such as two shutdown systems) to perform equivalent functions, and by satisfying the “two group” design concept. Two groups of safety-related systems are provided in the station to maintain the station in a safe state even if a failure of one of the groups occurs. Group seperation provides inherent protection against common-cause failures from disturbances such as fires and third party acts that can influence a limited area of the station.

The two groups are:

- Group 1 systems

  Systems that provide a safety function to mitigate an event, and that also perform a safety function or power production function during normal station operation. Group 1 includes:
  - the power production systems
  - one group of special safety systems
  - a set of safety support systems

- Group 2 systems
  Systems that provide a safety function to mitigate an event, but perform no function during normal station operation, are allocated as Group 2 systems wherever possible. Group 2 includes:
  - the second group of special safety systems
  - a second set of safety-support systems

Component redundancy is built-in for the special safety systems (the two shutdown systems, emergency core cooling system, and the containment system) that makes sure that the single failure criterion is satisfied. special safety systems satisfy an unavailability target of 10^{-3}, which effectively requires redundancy of all critical components. The availability of these systems is verified during operating by regular safety system component tests.

Process systems also make extensive use of redundancy to improve the station availability in the production of electrical power. This redundancy minimizes the frequency with which serious process failures occur.

DIVERSITY
Diversity is the use of two physically or functionally different means of performing the same function. It provides protection against certain types of common-mode failures, such as those arising from design or maintenance errors.

Providing two shutdown systems for CANDU reactors is good example for diversity. The design concept of system diversity is also used in the design of independent emergency cooling water and power systems provided via the two-group approach which perform support services. In addition, CANDU nuclear power stations are required to design for dual failures that consist of a design-basis initiating event with an assumed coincident unavailability impairment of one safety system. This means that the station is designed, for example, to mitigate a Loss Of Coolant Accident (LOCA) combined with loss of ECCS injection and the moderator system is shown to be an adequate means of fuel cooling for this event.

SEPARATION
Separation refers to the use of barriers or distance to separate components or systems that perform similar safety functions. Therefore, if a failure or localized event occurs in or near one system or component, it is unlikely to affect the other. Separation provides protection against common-mode or cross-linked effects such as fires and missiles.

Physical and functional system separation is designed into CANDU nuclear power stations to satisfy the two-group concept. The components of special safety systems that perform similar functions are separated to the maximum practical extent. Redundant components within systems are physically separated according to their susceptibility and common hazards. Specific requirements are applied to the triplicated instrument cables and the duplicated power and control cables for safety-related systems. The odd and even concept of on-site power distribution is applied to equipment, the raceway system and junction boxes, to maintain physical separation between the odd and even systems. This results in maximum reliability under normal and abnormal conditions.

The separation of special safety systems from the systems used for power production (process systems) is one of the fundamental safety principles and a regulatory requirement in Canadian practice. Its objective is to make sure that events affecting a limited area of the station and functional interconnections between systems do not impair the capability to perform the required safety functions under accident conditions.

18.3.2 Mitigation of Accidents

Mitigation of the consequences of accidents is achieved by design provisions and operating procedures. These include:

- measures to prevent fuel failure following a serious process failure;
- provisions to contain radioactive materials in the event of fuel failure.

Accident mitigation is also achieved by incorporating reliable and effective special safety systems that are capable of:

- limiting the consequences of accidents;
- incorporating multiple barriers as described in Article 18.2.2;
- incorporating measures to protect these barriers from damage due to accidents.

Mitigation of accidents also includes building redundancy and diversity as described in Article 18.3.1 in order to continue to provide important safety functions, such as electric power and heat removal, even after some components have failed as a result of an accident.

18.4 MEASURES FOR MAKING SURE THERE IS APPLICATION OF TECHNOLOGIES PROVEN BY EXPERIENCE OR QUALIFIED BY TESTING OR ANALYSIS

The commitment to quality in the CANDU design is addressed in Article 18.3.1 as part of accident prevention. As discussed in the Introduction, part 4, the CANDU design criteria and requirements include design and construction of all components, systems, and structures to follow the best applicable code, standard, or practice; and be confirmed by a system of independent audit.

Measures for making sure the application of state-of-the-art proven technologies are embedded in the Canadian licensing process are described in Article 7.3 for the licensing of new nuclear power stations, and in Article 8.1.1 for operating licence renewal. In each phase of licensing, documents have to be submitted to describe the technology employed, and to verify and validate it. These include the Safety Report and the quality assurance program.

Tools and methodologies used in the Safety Report have to be proven according to national and international experiences, and validated against relevant test data and benchmark solutions. Site acceptance, the initial step in licensing a new station, will not be offered unless the preliminary Safety Report submitted satisfies such requirements. The Safety Report has to be completed and the methodology updated for both the construction licence and the operating licence.

Part of the Canadian licence requirement is to update the Safety Report at least once every three years for an operating nuclear power station. The following must be used or incorporated in the updated Safety Report:

- new methodologies
- computer codes
- experimental data
- research and development findings

As a result, many of the events in the Safety Report are often re-analysed in the updated version. The Consultative Document C-6 (Attachment 7.10) specifies the requirement for quality and validation for both analysis and computer codes to make sure there is adherence to current standards.

18.5 THE REQUIREMENTS FOR RELIABLE, STABLE, AND EASILY MANAGEABLE OPERATION WITH SPECIFIC CONSIDERATION OF HUMAN FACTORS AND MAN-MACHINE INTERFACE

Reliable, stable, and easily manageable operation of the CANDU nuclear power stations are facilitated by the design features of redundancy and diversity as explained in Article 18.3. The overlapping functions of the control and safety systems make it easier to operate the nuclear power station within its operating envelope.

The AECB requires the licensee to identify the fundamental rules of their operations, including those related to management and reliability. This is demonstrated in the Operating Policies and Principles (OP&P) document prepared by the licensees. The OP&P is described in Article 10.1.2. It identifies:

- responsibilities;
- the operating envelope;
- the principles to be applied for safe, easy, and well-controlled operation.

The OP&P is reviewed and approved by the AECB before an operating licence is given. Any failure of the licensee staff to follow the requirements of OP&P represents a breach of the licence.

Consideration is given to human factors and man-machine interface throughout the entire life of the station to make sure that stations are tolerant of human error. Examples where the consideration of human factors and man-machine interface have been addressed are:

- Automatic actuation of controls or protection systems was developed in order to respond to equipment failure or human error, which could cause a station parameter to exceed normal operational limits or a safety system trip set-point. The overall station design and the specific design of protection systems make sure that operator intervention is only required in cases where there is sufficient time for the diagnosis of station conditions and the determination and implementation of operator actions.

- The design of the control room incorporated a strategic placement of the instrumentation and controls used in safety-related operations and in accident management. Specific attention is given to:
  - device grouping
  - layout
  - labelling
  - device selection

  Appropriate attention to human factors and man-machine interface concerns makes sure that the information available in the control room is sufficient for the diagnosis of anticipated events or transients, and for the assessment of the effects of any actions taken by the station operators.

- Reliable means of communication are provided between the control room and operating personnel at remote locations of the facility to facilitate the performance of manual actions. Effective use of communication protocols and operating personnel’s familiarization with the normal operation of systems, and the location of the system controls, minimizes the chances of human errors.

- Operations (both normal and abnormal) and maintenance procedures provide detailed instructions for the completion of assigned tasks. Procedural accuracy and compliance minimize the possibility for human error and assist in man-machine interface.

- Operations and maintenance training is provided to create and maintain job performance capability. This training normally includes:
  - classroom instruction
  - workshops
  - on-the-job instruction
  - supervisory coaching
  - informal briefings

  Training is designed to make sure that employees perform the tasks required for their positions competently, independently or in a team. Making sure employees are qualified and trained for their positions provides an additional barrier that minimizes the probability of human error.

- System alignment verifications and post-maintenance testing are routinely performed to detect and correct human errors that occur during system manipulation or maintenance.

More about human factors methods, managerial and organizational issues, and the role of the AECB is discussed in Article 12.

ARTICLE 19
OPERATION

19.1 A DESCRIPTION OF THE LICENSING PROCESS, INCLUDING A SUMMARY OF NATIONAL LAWS, REGULATIONS, AND REQUIREMENTS RELATED TO THE OPERATION OF NUCLEAR POWER STATIONS

The national laws and regulatory requirements are detailed in Article 7.2, and the licensing process is described in Article 7.3.

The prerequisites for a new power station operating licence are summarised in Article 7, Table 7.1; while the operating licence renewals, and the licensing requirements for continued operation, are described in Article 8.1.

19.2 A DESCRIPTION OF THE STEPS CANADA HAS TAKEN IN IMPLEMENTING THE FOLLOWING OBLIGATIONS UNDER ARTICLE 19 OF THE CONVENTION

19.2.1 Initial Authorization to Operate a Nuclear Power Station

The initial authorization to operate a nuclear power station is based upon an appropriate safety analysis and a commissioning program. These must demonstrate that the nuclear power station, as constructed, is consistent with design and safety requirements.

As stated under Article 7.3, before issuing an operating licence, the Atomic Energy Control Board (AECB) must be assured that:

- the construction of the station conforms to the design submitted and approved;
- the safety analysis is complete;
- the plans for operation are satisfactory.

Before a station is commissioned, at least one staff member of the AECB is located at the station to observe and report on the commissioning and start-up processes.

The AECB staff does not attempt to participate in all aspects of the licensee’s commissioning program. Reliance is placed on the licensee’s internal review process, which is mandated by the commissioning quality assurance plan. Typically, the licensee’s procedures require that the detailed commissioning specifications for a particular system or component be approved by the designers. The specifications define the acceptance criteria to be used in inspections and tests done as part of the commissioning program. The reason for insisting on approval by designers is to make sure that the commissioning program is consistent with the following design requirements:

- the inspection and test program is checking the right items;
- the acceptance criteria being used are appropriate to prove that the equipment can perform the safety functions intended in the design.

The quality assurance plan also requires that the process of approving the specifications and the results be documented. Any failure to meet the acceptance criteria is to be referred back to the design organization which decides what, if any, design changes are required. This allows the AECB staff to do audits, at any time, to confirm:

- that the procedural requirements are being complied with;
- that the decisions made are appropriate.

The AECB staff's direct involvement in commissioning concentrates on a few major tests, such as those that check the overall station response to specific events. One example is a test of the station response to a loss of the normal electrical power supplies. AECB staff witness major commissioning tests of special safety systems, such as the functional tests of the shutdown systems where the reactor is actually tripped and the rate of power reduction is measured. Measurements are compared to the rate assumed in safety analyses.

In other cases, a complete test is not practical, and partial tests are done instead. This is the case with commissioning tests of emergency core cooling systems. For example, while commissioning tests have been done involving injection of emergency coolant into the reactor core, tests have not been attempted in which cold water is injected into a hot core as they could lead to very high stresses in primary coolant system components. The components are designed to withstand these stresses in an emergency, with the assumption that the components will be subjected to these emergency stresses at some time during their design life but on a limited number of occasions. Exposing the components to such high stresses during a test is not justified.

The AECB staff's review of commissioning concentrates on these major tests because they are considered particularly important to safety. These tests check the overall performance of the station's safety features and can reveal problems that would not be detected by tests of individual components. The AECB staff's involvement with these commissioning tests includes reviews of the test proposals including the detailed commissioning specifications. These are examined to confirm that the acceptance criteria for the test are consistent with the system's safety design requirements as defined in the licence application. After the completion of the tests, AECB staff reviews the test results and the commissioning reports produced.

The AECB requires the licensee to submit commissioning completion assurances before starting up the reactor. Commissioning completion assurances are written certifications that indicate:

- commissioning has been completed according to the process described in the licence application;
- commissioning results were acceptable.

To prepare these assurances, the licensee needs to have a mechanism to confirm that the required commissioning has been done and that the results have been confirmed to be acceptable. The licensee's process for doing this is part of the commissioning quality assurance plan. Typically, the licensee holds a series of commissioning completion assurance meetings to review the work done on particular systems. The AECB site office staff attends some of these meetings. The AECB staff requires completion assurance statements before first loading of fuel, and of heavy water, and before first criticality of the reactor.

The completion assurance statements may contain lists of tasks not yet completed. These include tasks such as completion of commissioning reports that are not prerequisites to the approvals being sought. Nevertheless, these lists of incomplete items serve a useful purpose since they can be used later to verify that completion of these tasks is not overlooked.

19.2.2 Operational Limits and Conditions

Operational limits and conditions derived from the safety analysis, tests, and operational experience are defined and revised as necessary for identifying safe boundaries for operation.

It is a fundamental nuclear safety requirement that the stations be operated and maintained in accordance with the design intent and the licensing basis, that is, within the defined Safe Operating Envelope (SOE).

SOE is defined by a number of documented safe operation requirements. Among the main components of the SOE are:

- requirements on special safety systems, and safety-related standby equipment or functions, for example:
  - setpoint and other parameters limits;
  - availability requirements.

- requirements on process systems, for example:
  - parameter limits;
  - testing and surveillance principles and specifications;
  - performance requirements under abnormal conditions.

- prerequisites for removing special safety systems and other safety related, or process standby equipment, from service.

These requirements/prerequisites are derived from:

- higher level regulatory requirements
- design performance and reliability requirements
- design support analysis
- analysis of design basis accidents
- other safety analysis
- other types of analysis, for example, probabilistic risk assessments

The analysis, in principle, considers all allowable station states. However, it is not feasible to analyse in advance every potential station state that can occur throughout the life of the nuclear power station. Therefore, the analysis attempts to consider sufficient situations to allow an SOE to be defined that encompasses the expected variations in station conditions at a reasonable level of system/equipment performance details. Station states defined by the SOE are restricted to the analysed states, which have been shown to be safe and have been approved by the AECB. Analysis is continuously improved and updated to reflect technological advancements.

The analysed SOE should satisfy the regulatory requirements, standards, and guidelines, and the defence-in-depth principles in station design and operation. Other limitations considered are related to equipment and materials, production requirements, equipment ageing, etc. Failure modes and effects analysis can also contribute to the SOE.

The technical basis for SOE is found in the Safety Report. The Safety Report includes a description of the safety analysis which examines the station responses to:

- disturbances in process function
- system failures
- component failures
- human error

Safety analysis predicts the consequences of the design-basis accidents, and compares them to regulatory standards (see Consultative Document C-6, Attachment 7.10). SOE limits take into account instrumentation and analysis uncertainties.

The SOE is implemented in the following station documents:

- Operating Policies and Principles (OP&P) (see Article 10.1.2)
- Operating Manuals
- Abnormal Incident Manual (or Impairment Manual)

The latter document specifies actions to be taken when there are indications that operation is getting close or outside the limits of the SOE.

At Point Lepreau station, a major systematic review of each special safety system has been conducted to determine if the design, operation, and analysis of the station are consistent. The project is called Determination Of Allowable operating envelope

(DOA), and serves also to determine station parameters that define the allowable SOE. Detailed review of the following items was conducted with the review findings documented and periodically updated:

- system requirements
- original and current design
- operating history
- component ageing
- safety analysis assumptions and methodologies

19.2.3 Operation, Maintenance, Inspection, and Testing of Nuclear Power Stations

Operation, maintenance, inspection and testing of nuclear power stations are conducted in accordance with approved procedures.

The power reactor operating licences issued by the AECB contain a series of conditions that are designed to ensure that:

- a licensee conforms to AECB requirements;
- AECB staff is given the authority to obtain information and impose specific requirements where necessary.

The following sections examine some of the conditions that are included in nuclear power station operating licences. A typical licence (Pickering B) is included in Annex 7.1 as a sample case.

The licensee is required to operate in accordance with the OP&P document that sets out the fundamental rules that govern safe operation of the nuclear power station. This document is prepared by the licensee and must be approved by the AECB. It must contain:

- a definition of the authority and responsibilities of managers and operating staff;
- the principles to be applied for the safe operation of each of the station's systems;
- the specific numerical limits for operating parameters that must be maintained to make sure that the station always operates within its analysed safe operating envelope.

As with other documents specifically referenced in the operating licence, any failure of licensee staff to follow requirements contained in the OP&P would represent a breach of the licence.

The licence also specifies the minimum staff complement which must be present at the station at any one time. The AECB includes this requirement to make sure that there are always a sufficient number of appropriately qualified staff available to respond to an emergency.

Tests and inspections on systems, equipment, and components must be performed regularly by the licensee to confirm their availability, and maintenance must be carried out to a standard that is acceptable to the AECB. In particular, the licensee must make sure that the reliability and effectiveness of all equipment and systems continue to meet the standard claimed in the Safety Report and in the documents submitted as part of the application for the operating licence. (See also Articles 7.3.4, 8.1.1 and 14.1.)

Maintenance and testing for special safety systems is to be carried out according to special procedures that are set out in the OP&P document. These procedures are designed to make sure that no safety function is ever compromised by maintenance activities.

Safety system testing is required on a regular basis to demonstrate that each safety function is operating correctly. The AECB requirement is that each system must be capable of operating without fault for 99.9% of the time in accordance with the safety requirements. Each component of a special safety system is subject to a regular functional test. The frequency of each test is determined from reliability analysis by which the expected reliability of a system is determined from a knowledge of the reliability of its components.

In the operation of a facility, quality assurance is an essential aspect of good management that contributes to the achievement of quality and therefore to safety. Therefore, the AECB requires that the licensee prepare and maintain a quality assurance program incorporating a disciplined approach to all activities affecting safety in operation including, where appropriate:

- verification that tasks have been satisfactorily performed;
- documentary evidence to show that the required quality has been achieved.

The quality assurance program also contains procedures to make sure that any necessary remedial or corrective actions are implemented.

19.2.4 Establishing Response Procedures

Procedures are established for responding to anticipated operational occurrences and accidents.

The response to anticipated operational occurrences and accidents is controlled through a hierarchical system of station procedures. Although procedure variations exist between stations, the generic structure of this system is summarized as follows:

- Operating Manuals
- Alarm Manual
- Abnormal Incident Manual (or Impairments Manual)
- Radiation Protection Manual (or Radiation Protection Directives)

Procedures used by the station operations staff during routine operation of the nuclear power station and its auxiliaries are located in the Operating Manuals. There are typically two categories of procedures within the System Operating Manual:

- System-based procedures that control operation of station systems during:
  - normal operations
  - system start-up and shutdown
  - system failure
- Integrated procedures that coordinate major station evolutions such as station start-up and shutdown

Alarm Manual procedures provide the operations staff with information regarding alarm functions. Typical information provided within these procedures includes:

- set points
- probable cause of alarm
- pertinent information
- references
- operator response

Abnormal Incident Manual procedures provide information to the operations staff that may be helpful following safety system impairment, process system failure, or a common mode event. At Ontario Hydro Nuclear (OHN), there are three categories of procedures within the Abnormal Incident Manual:

- Abnormal State of Safety System procedures
- Emergency Operating Procedures
- Critical Safety Parameter Monitoring procedures

At other utilities, Abnormal Plant Operating Procedures and Emergency Operating Procedures are issued as separate manuals.

The Abnormal State of Safety System procedures direct compensatory actions to be taken when a safety system is impaired or unavailable. The Emergency Operating Procedures direct operator actions during accident conditions, and are designed to restore the station to a safe condition, and make sure that there is protection for the health and safety of station personnel and the general public. Critical Safety Parameter procedures provide augmented monitoring of critical station operating parameters during accident conditions, and provide a support feature to the Emergency Operating Procedures. See Article 16 for On-site Emergency Procedures (Contingency Plans).

Radiation Protection Manual procedures are provided to guarantee the safety of the operators and the general public in the event of a significant radiation incident. These procedures:

- direct event classification and categorization;
- provide provisions for off-site notification;
- direct protective actions and monitoring during accident conditions.

19.2.5 Necessary Engineering and Technical Support in all Safety-related Fields

Necessary engineering and technical support in all safety-related fields is available throughout the lifetime of a nuclear power station.

The financial and human resources of the licensees are addressed in Article 11. These resources are planned throughout the life of the station. They include improvements that have to be made as well as decommissioning. Budgets are also made available to hire external services and have service contracts in place to provide support in areas outside the technical or engineering expertise of the station full time staff. New Brunswick Power and Hydro-Québec have service contracts with Atomic Energy of Canada Limited (AECL), OHN, and many other Canadian companies to obtain services in many fields including:

- research
- engineering
- analysis
- assessment
- maintenance
- inspections
- design support

This is in view of the limited internal resources of these two utilities in various areas related to the support of their nuclear power stations.

OHN has undertaken initiatives to provide assurance that necessary engineering and technical support is available in all safety-related fields throughout the lifetime of each nuclear power station. A base work program model has been prepared by an external consultant to identify staffing needs for OHN organizations. This model considered industry data from more than fifty nuclear generating stations to determine target staffing levels at OHN. In addition to identifying station specific staffing targets, staffing targets have also been identified by job family, department, and work program. Following the Independent and Integrated Performance Assessment (see Article 6.2.3), an increase in staffing has been decided by management and included in the corporate business plan.

19.2.6 Reporting Incidents Significant to Safety

Incidents significant to safety are reported in a timely manner by the holder of the relevant licence to the regulatory body.

The reporting requirements in Canada are described in Regulatory Document R-99 (Attachment 7.9). This document contains the requirements for reports that operating nuclear power stations must make to the governing regulatory body. Each licensee understands the importance of keeping the AECB fully informed of issues that potentially impact the safety of its nuclear stations. Consequently, each licensee who operates a nuclear power station in Canada has implemented procedures and trained personnel to make sure that the following facility reports are submitted to the AECB in accordance with the requirements of R-99:

- Event Reports
- Quarterly Reports
- Safety Report Updates
- Radiological Environmental Monitoring Reports
- Research and Development Reports
- Periodic Inspection Program Reports
- Reliability Reports

The above reports are described in Article 9.2.

Incidents significant to safety are reported mainly in the Event Reports in addition to many of the other reports depending on the nature of the incident. The Event Reports provide description of the unplanned event and any violations to the licence conditions. They also provide information on the safety system response to the event, and on any radiation emissions and doses that might have occurred.

In each case, the licensee is required to make an oral event report to the AECB within one business day of the discovery of the reportable condition. In addition, a detailed written report must be subsequently submitted within the time-period specified by R-99.

In addition to the above, utilities voluntarily report any item that has potentially significant generic or safety implications (for example, a fuel fabrication defect) that is not explicitly required by R-99.

19.2.7 Programs to Collect and Analyse Information on Operating Experience

Programs to collect and analyse information on operating experience are established, the results obtained, and the conclusions drawn are acted upon. Existing mechanisms are used to share important experience with international bodies and with other operating organizations and regulatory bodies.

OPERATING EXPERIENCE (OPEX) FEEDBACK SYSTEMS
The process of collecting, analysing, and disseminating lessons learned from information arising from the operating experience is known as a “feedback” process or system. Feedback systems established by the utilities in Canada are normally part of the utility’s or the station’s Quality Assurance system. In addition to the utilities, OPEX feedback systems also involve the AECB, the CANDU Owners Group (COG), and other organisations.

REQUIREMENTS AND OBLIGATIONS

Feedback systems in Canadian nuclear power stations are subject to quality assurance requirements. The Canadian Standard CAN/CSA-N286.5, Clause 3.9 (FEEDBACK) calls for measures to make sure that operations experience is documented, assessed, and incorporated into the operation of the station and/or its QA programs as appropriate. It also calls for making this information available to personnel in the other phases of the station's life cycle. Under this clause, the AECB has been conducting audits in nuclear power stations and utility corporate offices to make sure that the existing feedback systems achieve their objectives.

Because operating licenses stipulate requirements for reporting of certain occurrences to the AECB, and because the reporting of such occurrences is an essential element of the feedback process, the feedback system in the utility is partly subject to licence requirements. Licensees are required to report certain incidents or events to the AECB. The operating licence for a nuclear power station stipulates requirements for reporting these occurrences in accordance with the Regulatory Document R-99.

There are also international obligations that have to be met by the AECB. As a member in the International Atomic Energy Agency (IAEA) and the Organization for Economic Cooperation and Development (OECD), Canada is committed to report to both the Incident Reporting Systems (IRSs) of the IAEA and the Nuclear Energy Agency (NEA) of the OECD on significant events that occur in Canadian nuclear power stations. Canada, as a participant, appointed a member of the AECB staff as a national coordinator to collect and analyse information on events occurring in Canada, and to transmit them to the NEA and the IAEA.

Canada also participates in the International Nuclear Event Scale (INES) reporting system which is administered by the IAEA. The system uses a severity scale for use between countries to describe the safety significance of incidents and accidents. The purpose of the system is to give the media and the public a good perspective of the reported incidents and accidents. In the interest of expediting the transmission of information from Canada, the coordination, exchange of reports, and the classification of event severity are delegated to the CANDU Owners Group (COG) organization. For information on COG, see the Introduction, part 2.2, and Annex 1.1.

SOURCES OF INFORMATION

The primary source of information is the Significant Event Reports or Event Reports (SERs or ERs) which are written by the utilities. They provide information on undesirable events that are considered significant in the operation of nuclear generating stations and related facilities.

A number of criteria are defined for incidents requiring a report. These reports generally encompass all incidents that did, or could have had, an impact on station, personnel safety, the environment, or on unit production or other economic loss. They include, for example:

- failures of safety or process equipment
- releases of radioactivity
- releases of chemicals hazardous to the environment
- significant losses of heavy water
- unplanned shutdowns
- near-miss equipment faults or personal injuries
- human errors
- other events identified by station management

Other reports include the licensees' quarterly reports, in-service reports, and internal audit reports. On the regulatory side, the AECB issues Audit Reports on operations in nuclear power stations. These reports contain the AECB audit teams' findings and the deficiencies that the licensees are required to correct.

International sources include the Incident Reporting System (IRS) reports from the IAEA and OECD, and Information Notices and Bulletins of the U.S. Nuclear Regulatory Commission (USNRC). These reports are regularly distributed by the AECB to all licensees in Canada. When received by the licensees, these reports become part of their feedback systems, and should be reviewed for implications on their power stations.

CHANNELS OF FEEDBACK

There are a number of feedback channels/systems within the utility, among uitlities, and at the AECB.

- Within the utility

  Feedback systems within the utility are aimed at improving both the reliable production of electricity and nuclear power station safety. For utilities with single unit stations the feedback system involves the station only. If a corporate office exists, as in the case of OHN, several feedback systems exist; one in the corporate office and one in each station.

- Among utilities

  Another feedback loop exists between CANDU stations. A direct exchange of information on operating experience between CANDU stations is coordinated by the COG program. Reports are normally transmitted via the CANDU Network (CANNET) system. CANNET is a computerized electronic communications system designed for the exchange of information, amongst members of COG, pertinent to the following aspects of CANDU nuclear power stations:

  - design
  - construction
  - licensing
  - safety
  - operation
  - maintenance

Reports are posted on CANNET by member utilities to enable other stations to determine if they are susceptible to a similar occurrence.

In 1991, in a letter to all station managers, the AECB made it clear that it views the direct exchange between utilities of event reports that are judged to have safety implications as vital in preventing recurrence. The letter urged the licensees to make sure that such reports are transmitted to other utilities in a timely manner. In addition to urging the licensees to exchange information through COG, the AECB took another initiative. It requested and acquired access to the CANNET to observe the flow of safety information between utilities.

- At the AECB
The feedback system within AECB is concerned with the safety of the workers, the public, and the environment. The Unusual Event Processing System (UEPS) was set up by the AECB in 1983. It is a unified approach to the evaluation of events occurring at nuclear power stations. Staff members from a number of specialist groups participate in the related assessments.

  The objectives of the system are to conduct detailed review of operating experience to help identify safety concerns, to improve dissemination of such information, and to feed back the experience into licensing, regulations, and operations.

  To more effectively collect, screen, store, and retrieve operational data, the information is maintained and managed in a computerized database. There are currently over 12,000 records of events in the database that include both reportable and non-reportable occurrences. The use of a computerized database to process operational data does not arise only from the need to store large amounts of information. It also arises from the need to perform parametric trending of performance indicators and to produce periodic reviews. This is because some problems are not indicated by any one event in itself, but by a number of events over a period of time.

CONCLUSIONS AND ACTIONS ARISING FROM OPEX ANALYSIS
The results of the event analyses that are conducted by the AECB are disseminated to members of the AECB staff, Canadian utilities and some provincial authorities. Problems or issues that may be applicable to other stations are identified and brought to the attention of the AECB site project officers and different specialist groups in the AECB. They use this information to assess the licensee’s submissions.

The AECB site project officers incorporate the results of event analyses in their reviews and assessments of licensee’s corrective actions in response to a certain event. Where corrective actions undertaken by the licensee are considered inadequate, further actions are requested. In addition, the site project officers review the status of corrective actions to make sure that they are completed expeditiously.

AECB audit teams consult the operating experience in the AECB database in planning strategies for their audits, and in identifying problem areas in operation or maintenance such as:

- procedural noncompliance
- procedural deficiencies
- use of nonstandard components

Similarly, assessments conducted by AECB specialists often utilise the operating experience recorded in the AECB database.

The Reactor Projects Coordinating Group (RPCG) is an AECB team of senior project officers and senior managers. The team meets periodically to discuss important safety issues. At these meetings, OPEX lessons learned from one station are communicated to other stations. This practice allows the AECB to make sure that each utility benefits from the operating experience acquired in the other power stations.

MONITORING OF FEEDBACK
The AECB conducts audits at nuclear power stations. These audits are directed towards elements of the feedback process that include both the internally generated information such as event reports, and external information that are received from other nuclear generating stations in Canada and in foreign countries.

The AECB also monitors the feedback between the utilities through the CANNET network as described earlier.

19.2.8 Minimum Generation of Radioactive Waste

The generation of radioactive waste that results from the operation of a nuclear power station is kept to the minimum practicable for the process concerned, both in activity and in volume. Any necessary treatment and storage of spent fuel and waste directly related to the operation, and on the same site as that of the nuclear power station, takes into consideration conditioning and disposal.

The methods practised in Canada by nuclear power station operators for the management of low- and intermediate-level radioactive waste and irradiated nuclear fuel are similar to those practised in other countries. Primary emphasis is placed on minimization, volume-reduction, and long-term storage of the waste as disposal facilities are not yet available.

RESPONSIBILITY
The federal government has established a national comprehensive and integrated policy for the efficient and effective management and disposal of all radioactive wastes. However, primary responsibility for the management and long-term storage of radioactive waste and irradiated spent fuel rests with the operators.

OPERATIONS
All waste produced at nuclear power stations are segregated at their points of origin as either contaminated or non-contaminated waste. Low-level and intermediate-level contaminated wastes are further sorted into distinct categories such as waste that:

- can be incinerated;
- can be compacted;
- cannot be processed.

The further sorting of the waste helps to facilitate subsequent handling, processing, and storage. Because there are no fuel re-processing activities in Canada, spent nuclear fuel is automatically categorized as high-level waste and placed in long-term storage.

WASTE TREATMENT
Radioactive waste is usually volume-reduced by either compaction or incineration to minimize handling and storage space requirements. With the current techniques available in Canada, the radioactive waste can be reduced by about 60% of its volume prior to storage.

STORAGE
Storage is the main care-taking activity in Canada for radioactive waste produced at nuclear power stations. Low- and intermediate-level waste is stored either on-site or off-site in above or below-ground engineered structures. Some of the waste may be volume-reduced by compaction or incineration prior to storage as mentioned above. Spent nuclear fuel is placed in wet storage at the nuclear power station for a period of time after which it may be transferred to aboveground dry storage facilities for long-term storage. All radioactive waste is stored in such a way that it can be retrieved.

Operators have instituted methods to recover storage space by cascading the waste after sufficient radioactive decay, or reclaiming existing storage space through further compaction (super compaction) and/or segregation.

INITIATIVES IN DISPOSAL
Various initiatives are currently underway to develop and implement radioactive waste disposal systems in Canada for low- and intermediate-level radioactive waste and for irradiated spent fuel.

ANNEX 1.1
CANDU OWNERS GROUP
RESEARCH AND DEVELOPMENT PROGRAMS

In Canada, the Research and Development (R&D) in support of operating CANDU stations is administered and managed through an organization called the CANDU Owners Group. COG comprises the three Canadian nuclear utilities: Ontario Hydro Nuclear (OHN), New Brunswick (NB) Power, and Hydro-Québec; and Atomic Energy of Canada Limited (AECL). The jointly funded R&D projects are conducted in several technical areas, namely:

- Nuclear Safety and Licensing
- Radiological Health and Safety
- Fuel Channel Technology
- Chemical Engineering and Processes
- Process Systems and Equipment
- Instrumentation and Control
- Steam Generator Technology

The organization structure for COG is illustrated in Figure 1A.1. It shows the following:

- Directing Committee: That sets program objectives, assesses progress, and assigns budgets.
- COG Operations: That administers and monitors the technical program.
- Technical Committees: That manage the research work in the technical areas mentioned above.
- Working Parties: That direct research in specific disciplines for each technical area, and interact closely with the researchers.

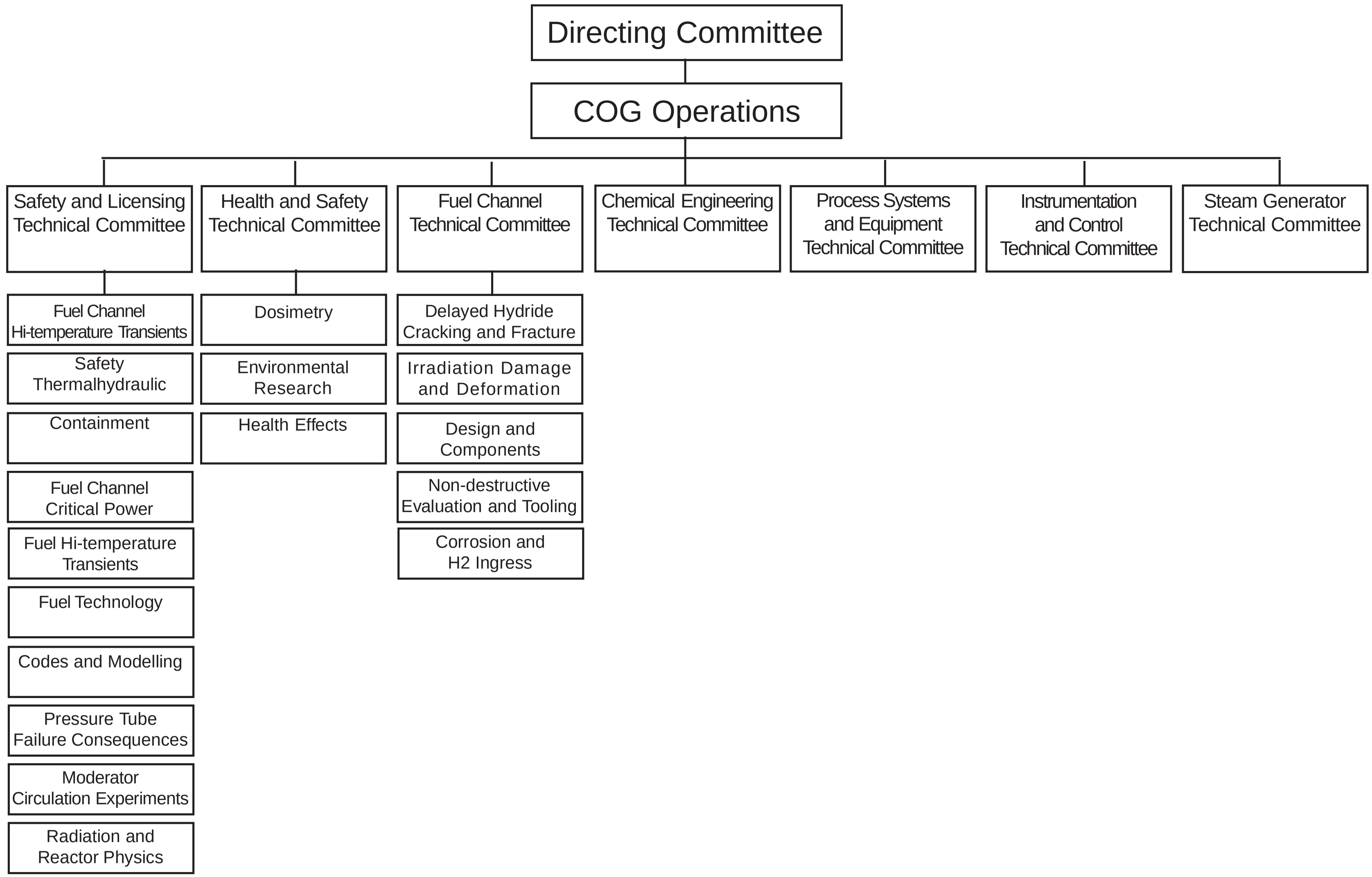

Figure 1A.1: CANDU Owners Group R&D Program Structure and Technical Areas

The COG R&D programs generate new information and data required for engineering and technical support to the CANDU nuclear power stations. The programs are described below.

1. NUCLEAR SAFETY & LICENSING PROGRAM

The Nuclear Safety & Licensing program advances the technologies that are basic to the resolution of nuclear safety issues. The experimental programs generate the data and the predictive tools required for quantifying and reducing the uncertainties in nuclear power station safety analysis. In addition, the program provides continued assurance of safety by developing the understanding of phenomena that influence safety. The goals of the program are to:

- support verification of the safety design basis and safe operating envelope for operating stations;
- contribute to resolving outstanding licensing and safety issues;
- maintain the core capabilities, the scientific expertise, and the infrastructure necessary to support a nuclear safety research program.

The sub-elements of the Nuclear Safety R&D Program are described below:

1.1 Fuel Channel High Temperature Transients

The Fuel Channel High Temperature Transients program addresses the physical phenomena related to the thermal, mechanical, and chemical behaviour of CANDU fuel channels under high temperature conditions that are typical of those encountered during accident conditions (for example, loss of coolant with loss of Emergency Core Cooling (LOCA/LOECC) type events). In this area of technology, experimental data and models are developed to:

- characterize pressure tube deformation behaviour at elevated temperature conditions that challenge fuel channel integrity;
- characterize fuel channel thermal-chemical behaviour at elevated temperatures under conditions typical of LOCA/LOECC events;
- quantify heat transfer from fuel channels to the moderator at elevated temperatures and confirm the effectiveness of the moderator as a heat sink;
- characterize the effects of aging on accident consequences.

1.2 Safety Thermalhydraulics

The Safety Thermalhydraulics programs provide the experimental data required for analysis of the thermalhydraulic aspects of CANDU system behaviour under postulated upset and accident conditions. Model development and validation are provided through:

- improved understanding of the behaviour of CANDU system components through experiments and data analysis;

- improved understanding of the thermalhydraulic behaviour of an integral figure-of-eight loop possessing many of the physical and geometric characteristics of a CANDU reactor heat transport system;
- development of improved instrumentation to aid in the qualification and quantification of thermalhydraulic test facilities;
- development of software tools to aid in the documentation and analysis.

1.3 Containment

The Containment program focuses on four areas covering aspects of hydrogen behaviour in containment, fission product behaviour, aerosols behaviour, and equipment performance.

Specifically, technologies are developed to:

- support the resolution of safety concerns related to post-accident hydrogen behaviour;
- obtain the information and tools needed to assess and minimize the release of fission products aerosols from containment following a reactor accident;
- obtain the information and tools required to assess and minimize the release of radioiodine and other gaseous fission products from containment following a reactor accident;
- obtain the necessary understanding and demonstration of the performance of specific equipment used in containment under normal and accident conditions.

1.4 Fuel Channel Critical Power Ratio

The Fuel Channel Critical Power Ratio (CPR) program provides reliable thermalhydraulic prediction methods for Loss Of Regulation (LOR) accident and LOCA conditions, with emphasis on the prediction of Critical Channel Power (CCP) for both normal and abnormal conditions. It also emphasizes the prediction of the consequences of exceeding the Critical Heat Flux (CHF).

Specific objectives of this technology area are to:

- develop techniques for increasing CCP and lowering post-dryout temperatures;
- improve understanding of the mechanisms controlling CHF, CHF modelling, onset of dry sheath quenching, film boiling, void migration, and two-phase pressure drop;
- develop, validate, and support the ASSERT subchannel code for predicting local thermalhydraulic conditions in fuel channels;
- assemble and update a package of thermalhydraulics prediction methods for use in CANDU fuel channels and other reactor components;
- generate Critical Heat Flux (CHF) and pressure drop data and prediction methods for CANDU fuel bundle design over a wide range of bundle operating conditions and coolant flow conditions;
- perform CHF separate effects tests and CHF modelling experiments.

1.5 Fuel High Temperature Transients

The Fuel High Temperature Transients program provides experimental data required to quantify the physical and chemical behaviour of CANDU fuel at high temperatures for conditions typical of accident scenarios leading to LOCA/LOECC.

Specific objectives of this technology area are to:

- perform separate effects studies of fission product release, transport, and deposition in the Primary Heat Transport System;
- perform in-reactor, all effects experiments to determine fuel and fission product behaviour under accident conditions;
- perform out-reactor studies of physical and chemical behaviour of CANDU fuel at high temperature;
- integrate international safety research into the understanding of severe accident phenomena in CANDU reactors;
- develop a suite of verified, portable, quality-assured mechanistic computer codes to simulate the behaviour under conditions ranging from off-normal to postulated high temperature accident conditions.

1.6 Fuel Technology

The Fuel Technology program provides experimental data to define fuel operating limits, post-defect release behaviour, and interpretation of the modelling of fuel interactions with fuel channels under normal operating conditions.

The specific objectives of this technology area are to:

- provide an understanding of CANDU fuel through assessments of the limits of fuel performance, and develop materials data for the use and validation of models;
- provide data from examinations of power reactors fuel for improved fuel designs;
- characterize fuel behaviour during operational reactor power transients.

1.7 Pressure Tube Failure Consequences

The Pressure Tube Failure Consequences program addresses the potential safety issues that are related to core disassembly and the loss of power control for the design basis accident. In this program, mechanisms are examined that can lead to potential loss of moderator which is the ultimate heat sink under severe accident conditions. The results from the program provide the support required to confirm the basis of safety analysis, and to provide the data for validation of the codes used in the safety assessments.

The specific objectives of this technology area are to:

- establish the information for determining the potential for, and extent of, in-core structural damage following fuel channel failure;

- establish the range of expected in-reactor pressure tube failure geometries and quantify the existing margin against failure of the calandria tube over the full range of potential failure modes for pressure tubes;
- provide information on the important variables for use in safety analysis codes.

1.8 Moderator Circulation

In the Moderator Circulation program, an objective is to demonstrate the availability of adequate subcooling in licensing analyses by providing detailed predictions of moderator temperature distribution in the calandria prior to and during an accident. These predictions are made using state-of-the-art Computational Fluid Dynamics (CFD) codes. Knowledge of factors affecting moderator flow patterns and temperature distribution in the calandria, derived from experiments, as well as experimental database for CFD code validation are essential for improving the numerical modelling and for gaining confidence in the predictions. The subject is generic, affecting all CANDU stations.

The specific objectives of this technology area are to:

- create an experimental data base helpful in modelling CFD codes used for this purpose;
- support the verification of the safety design basis and safe operating envelope for operating stations.

1.9 Reactor and Radiation Physics

The Reactor and Radiation Physics program provides measurements for validation of coolant void reactivity calculations used in LOCA analyses; maintenance and development of computer codes; and establishing data libraries for reactor physics, radiation physics, and shielding analyses of CANDU reactors.

The specific objectives of this technology area are to:

- provide measurements of data for validation of coolant void reactivity calculations used in LOCA analysis;
- maintain and develop the computer codes and data libraries for reactor physics, radiation physics and shielding analyses of CANDU reactors.

1.10 Codes and Modelling

The Codes and Modelling program documents the knowledge of physical phenomena and incorporates the data into tools that can be used to design reactors and perform safety analysis. The mathematical models of the various phenomena are incorporated into computer codes which are then validated against experiments to make sure accuracy requirements are met for reactor analysis.

The specific objectives of this technology area are to:

- ensure the effectiveness of Emergency Core Cooling Systems (ECCS);
- provide core cooling information in the absence of forced flow;
- provide analyses of the behaviour of fuel and fuel channels at high temperatures;
- provide data on trip effectiveness criteria and post-dryout behaviour;
- provide an understanding of molten fuel-moderator interactions.

2. RADIOLOGICAL HEALTH & SAFETY PROGRAM

The Radiological Health & Safety program is directed at improving techniques necessary to assess radiation exposures of occupational workers, the public, and the environment. This program also advances the understanding required to predict the effects of radiation on human health and the environment, and provides the tools to improve radiation protection in CANDU operating stations.

2.1 Dosimetry and Dose Control

The Dosimetry and Dose Control program provides improved external dosimetry systems in CANDU stations, validated models for the internal dosimetry of tritium and carbon-14, developed methods for analysing of Pu-239 levels in urine samples of CANDU workers and radiochemical detection of organically bound H-3 and C-14. Canadian University programs in Radiation Physics and Dosimetry are also supported.

The specific objectives of this program are to:

- make available methods and systems for cost-effective dosimetry and dose control, with emphasis on tritium and carbon-14;
- identify or develop appropriate instruments and techniques to characterize and monitor radiation fields and radioactive contamination;
- develop and transfer the tools and techniques needed to improve the efficiency and effectiveness of operational health physics.

2.2 Environmental Emissions and Dose Assessment

In the Environmental Emissions and Dose Assessment program, techniques are developed for measuring tritium concentrations in water and plant materials, and comparative studies are conducted for tritium dose estimates. In addition, the development of analytical techniques for C-14 and atmospheric dispersion models for dose assessment are provided. Radiological environmental programs in Canadian Universities are also supported.

The specific objectives of this program are to:

- provide reliable and cost-effective techniques for the assessment of environmental radiological impacts arising from releases of carbon-14 and tritium;

- validate the models consistent with quality assurance standards to be used for Derived Emission Limites (DEL) calculations, accident/consequences analyses, and emergency response dose assessments;
- provide reliable cost-effective techniques for monitoring radionuclide releases and radionuclide concentrations that demonstrate that CANDU facilities comply with the appropriate licensing requirements and operating constraints.

2.3 Health Effects of Ionizing Radiation

The Health Effects of Ionizing Radiation program provides:

- data to identify the effects of radiation exposures for occupational and accident situations;
- validation of data in the National Dose Registry;
- information on the socioeconomic status of nuclear industry workers in Canada;
- identification of low dose effects on carcinogenesis;
- qualification of the damage to DNA by tritium beta particles.

The specific objectives of this program are to:

- identify and quantify biological indicators of dose and dose quality (tritium, X-rays) to predict future health effects for individual risk;
- establish the effects and risks of low dose and low dose rates by understanding the mechanisms of carcinogenesis, thereby providing a basis for more cost effective radiation protection practices.

3. FUEL CHANNEL TECHNOLOGY PROGRAM

The Fuel Channel Technology program provides the predictive capability to avoid any unstable pressure tube ruptures. The R&D program addresses corrosion and hydrogen ingress, fracture and delayed hydride cracking, and irradiation damage and deformation.

The specific objectives of this technology area, with respect to safety, are to:

- validate fuel channel component design specifications to avoid pressure tube rupture;
- develop predictive capability for pressure tube blister growth, crack initiation, crack growth, leak rates from growing cracks, and the ability to assess critical crack sizes to end of life;
- provide a predictive deformation model for the assessment of stresses at pressure tube flaws;
- improve defect detection and characterization;
- develop a predictive capability for deuterium ingress.

The Fitness-For-Service Guidelines for CANDU pressure tubes incorporates the acceptance criteria, procedures, material property data, and derived values used in the evaluations and assessments to confirm the acceptability of in-reactor pressure tubes for continued service. These guidelines standardize the methods and the material property values used in the evaluations and assessments. Research and development work provides the databases used to establish the material property values and to gain a better understanding of the degradation mechanisms of pressure tubes.

The Guidelines comprise the following:

Section I	Acceptance Criteria and Evaluation Procedures
Section II	Material Properties and Derived Values

Currently, the flaw evaluation methodology and acceptance criteria specified in Section XI of the ASME Boiler and Pressure Vessel Code are not directly applicable to zirconium alloy pressure tubes used in CANDU reactors. A Technical Task Team was formed by COG to develop Fitness-For-Service Guidelines that would provide flaw evaluation procedures and acceptance criteria for pressure tubes in CANDU reactors. In developing these Guidelines, the applicable rules of Section XI of the ASME Code were utilized, in conjunction with rules developed specifically to address delayed hydride cracking in CANDU pressure tubes.

Clause 12 of CAN/CSA-N285.4-94 specifies the requirements for periodic inspection and surveillance of pressure tubes in operating CANDU nuclear power reactors. The following requirements related to detected flaws are to be followed:

- If the inspection results reveal a flaw exceeding the acceptance criteria of the Standard, the flaw must be evaluated to determine if the pressure tube is acceptable for continued service.
- If the inspection results reveal that pressure tube-to-calandria tube contact has occurred, an assessment must be completed to determine if pressure tube integrity will be affected.
- Fracture toughness of surveillance tubes must be assessed to demonstrate their acceptability for continued service.

Section I provides acceptance criteria and evaluation procedures to perform the following:

- evaluate if pressure tubes containing detected flaws are acceptable for continued service;
- evaluate if pressure tubes in contact with their calandria tubes are acceptable for continued service;
- evaluate generic changes to fracture properties of pressure tubes in service to determine if the tube properties are acceptable for continued service.

Appendices A to E of Section I provide procedures to be used to perform the following evaluations:

- sharp crack-like flaws;
- blunt notch-type flaws;
- pressure tube to calandria tube contact;
- leak before break and flaw stability analysis.

Section II provides the material property data and derived values required to perform the assessments of Section I.

4. CHEMICAL ENGINEERING & PROCESSES PROGRAM

The Chemical Engineering & Processes R&D program supports safe operation through proper control of chemistry and material selection.

The specific objectives of this technology are to:

- minimize radiation fields and occupational dose in CANDU reactors by developing chemistry specifications and procedures to minimize activity transport and to manage tritium. This is in addition to improving and developing decontamination solutions to use in reactor systems and subsystems;
- minimize corrosion failures and degradation (for example, pressure tubes, heat exchangers, and moderator systems) in reactor components and systems by understanding corrosion mechanisms and by developing specifications, procedures, and monitoring techniques;
- minimize chemical and radiochemical emissions and minimize radioactive waste production by modifying procedures and substituting advanced technologies for existing technology.

5. PROCESS SYSTEMS & EQUIPMENT PROGRAM

The Process Systems & Equipment R&D program provides solutions to generic operational problems and improves safety and reliability of systems and equipment in CANDU operating plants. This is achieved through optimized operating conditions, mitigating ageing degradation, and extending service life of key CANDU components.

The specific objectives of this technology area are to:

- develop new materials and guidelines to support regulatory, operational, and design safety issues. Specific components targeted are: nuclear check valves, concrete containment structures, fuelling machines, control cables, and primary heat transport pumps;
- develop design and analysis computer software to improve operating performance and safety margins, particularly for fuel vibration, fretting, and water hammer;
- develop the capability to predict ageing degradation of cables, containment structures, elastomers and seals;

- develop environmental qualification requirements to mitigate CANDU station ageing;
- develop rapid and reliable inspection and repair tooling to minimize reactor outage time and lower radiation exposure.

6. INSTRUMENTATION AND CONTROL PROGRAM

The Instrumentation and Control R&D program provides enhanced safety of CANDU operating stations through the application of improved instrumentation and control technology, information technology, and human factors engineering.

The specific objectives of this technology area are to:

- prevent spurious reactor trips and safety system impairments through improved maintenance, surveillance, and diagnostics;
- develop improved instrumentation and control systems for CANDU plant normal and abnormal operations. This is achieved through human factors engineering in software change processes, and the development of electromagnetic interference immunity for CANDU safety control systems.

7. STEAM GENERATOR TECHNOLOGY

The Steam Generator R&D program provides the technology to ensure the safe and reliable operation of steam generators and associated systems over the operating life of the station.

The specific objectives of this technology area are to:

- reduce steam generator tube degradation due to corrosion, cracking, and pitting by improved understanding of corrosion chemistry and the application of high temperature electrochemical monitoring.
- reduce steam generator tube degradation due to mechanical damage by conducting fatigue, vibration, and fretting wear analysis;
- develop fitness-for-service guidelines for CANDU steam generators;
- develop rapid and reliable inspection and maintenance techniques.

ANNEX 6.1
ATOMIC ENERGY CONTROL BOARD
GENERIC ACTION ITEMS

The Atomic Energy Control Board (AECB) uses Generic Action Items (GAI) as one method of pursuing concerns that may affect more than one station. At the end of 1997, 14 GAI remained open. The following is a brief description of each item.

1. HYDROGEN BEHAVIOUR IN THE CONTAINMENT

Following a Large Loss Of Coolant Accident (LLOCA), there may be a risk of hydrogen combustion in the containment. AECB requested that the licensees demonstrate that hydrogen combustion, if it occurs, will not damage containment or other important safety-related systems.

Based on the licensees current safety analyses, Hydro-Québec and New Brunswick (NB) Power predict that hydrogen combustion will not occur because the models calculate that the average hydrogen concentrations will not reach the flammability limit. The two utilities plan to support the existing one-dimensional analyses with more detailed three-dimensional calculations. They also note that radiolysis could produce more hydrogen in the long term following the Loss Of Coolant Accident (LOCA). To maintain low concentration in the containment, both Hydro-Québec and NB Power are considering installing passive catalytic recombiners.

Ontario Hydro Nuclear (OHN) stations have hydrogen igniters to burn hydrogen before concentrations reach dangerous levels. They believe that the resulting low energy combustion would cause little, if any, damage. The effectiveness of these igniters, however, depends on good mixing. The OHN plans include completing more detailed calculations to better support their position. OHN indicates that it does not need to install catalytic converters because of the venting characteristics of their containment systems.

2. CORE COOLING IN THE ABSENCE OF FORCED FLOW

Failure of the primary heat transport pumps to provide heat transport water for fuel cooling is a possibility in some accident sequences. Licensees then rely on natural circulation of coolant for removing residual heat from the fuel. AECB staff is satisfied that natural circulation with the Primary Heat Transport System (PHTS) full will be effective. Some partial inventory natural circulation experiments done at Whiteshell Laboratories, however, have shown degraded cooling in some channels. These results seemed to contradict the safety analyses assumptions regarding the effectiveness of natural circulation in partial inventory conditions. AECB requested that licensees identify the causes for those degraded cooling conditions and that, if needed, they review their safety analysis or implement adequate design changes.

In December 1994, Hydro-Québec proposed a design fix that would allow the heat transport feed system to provide water to the PHTS after an accident involving loss of forced circulation. This should reduce the void, and therefore optimize the effectiveness of natural circulation. AECB review concluded that the modification would resolve the problem. This action item remains open for the other licensees.

3. ASSURANCE OF CONTINUED NUCLEAR POWER STATION SAFETY

Safety-related functions in nuclear power stations must remain effective throughout the life of the station. AECB expects the licensees to have a program in place to prevent, detect, and correct any significant degradation in the effectiveness of important safety-related functions. AECB requires all licensees to demonstrate that they could give assurance of continuing nuclear power station safety. AECB review of the responses of the licensees determined that the licensees did not have a systematic and integrated approach. Without such an approach, AECB could not conclude that their activities are sufficient. Licensees continue to work to address AECB concerns.

4. POST-ACCIDENT FILTER EFFECTIVENESS

Venting of the containment following an accident may be needed to reduce the risk of an uncontrolled release of radioactive material. All operating nuclear power stations in Canada have filters in their venting system to limit the release of radioactive material when the stations vent the containment atmosphere. AECB has requested all licensees to show that the filters are properly qualified for their intended function, and to show that adequate surveillance and maintenance activities are in place for these systems.

5. REACTOR OPERATION WITH A FLUX TILT

AECB staff identified that the effectiveness of the regional overpower protection system was not analysed over the whole range of permissible operating conditions. The assessment of the Regional Overpower Protection Trip (ROPT) setpoint assumes an initial nominal flux distribution. Licensees’ Operating Policies and Principles (OP&P), however, allow reactors to be operated with a flux tilt for an extended period of time. AECB requested licensees to demonstrate the effectiveness of the regional overpower protection system over the full range of permissible operational flux tilts.

All licensees have submitted analyses and instituted operational constraints to address the issue of operation with a flux tilt. Hydro-Québec and NB Power reviewed existing analyses which included highly perturbed flux distributions. They determined that they could maintain confidence in the ROPT coverage using a trip setpoint selection criterion dependent on the measurable deviation in zonal fluxes. Both utilities have incorporated this into their procedures. OHN performed additional analyses using a different methodology, and identified that their operational procedures are sufficient to make sure there is trip coverage in the presence of flux tilts. Such procedures limit regional overpower protection system detector calibration.

6. BEST EFFORT ANALYSIS OF EMERGENCY CORE COOLING SYSTEM (ECCS) EFFECTIVENESS

The effectiveness of the ECCS has not been fully demonstrated for any CANDU nuclear power station to the satisfaction of AECB. Computer codes may not be sophisticated enough to predict all relevant phenomena occurring during LOCA. Furthermore, the available experimental database was not adequate to demonstrate that the system would be fully effective. In the early 1980s, AECB asked the licensees to develop improved computer codes to predict, with sufficient confidence, the effectiveness of the ECCS during a LLOCA. In 1993, the licensees submitted their Best Effort Analysis which relied on improved computer codes. This GAI requested a demonstration of adequate validation of these methods and codes.

The licensees have initially concentrated their validation efforts on the first few seconds of the LLOCA. They have submitted a number of validation reports for phenomena occurring in the first few seconds. The future plans aim at validating codes for the remainder of the event.

7. IMPACT OF FUEL BUNDLE CONDITION ON REACTOR SAFETY

Fuel and pressure tube inspections done during the late 1980s and early 1990s at OHN nuclear power stations seemed to suggest that fuel bundle degradation was beyond design assumptions. The observed degradations were related to such aspects as bundle end-plate cracking, spacer pad wear, positive sheath strain, etc. AECB asked all licensees to review the situation and show that they have a proper management process for making sure that the fuel components (for example, bundle, pencils, spacer pads, etc.) are meeting design and safety performance requirements.

8. MOLTEN FUEL-MODERATOR INTERACTION

Reduced flow to a fuel channel during full power operation is a design basis event. If this happens, it will lead to degraded cooling conditions in the channel. Fuel temperature will increase rapidly, possibly causing fuel to melt. Eventually, the channel will fail and the very hot material will be ejected within the calandria, potentially causing damage to in-core components and to the calandria. Safety analyses should demonstrate that such a design basis event does not impair the shutdown systems' capability to shut down the reactor, and does not threaten the integrity of other channels and of the calandria. Current safety analyses make the conservative assumption that all the fuel in the channel would melt and interact with the moderator after channel failure. The analytical models, however, lacked adequate validation. Therefore, the AECB has requested validation of the codes and methods used.

In 1995, licensees decided to replace this conservative assumption with a more realistic approach because of the difficulty of supporting predictions for such extreme conditions. In 1996, licensees submitted a report that described their new approach and presented their findings. The report concluded that only a small amount (15 kg) of molten material will exist at the time of fuel channel failure. It also concluded that the interaction of this material with the moderator would not result in significant damage to the reactor systems structures or components beyond that already recognized in the Safety Reports.

9. PRESSURE TUBE FAILURE WITH CONSEQUENTIAL LOSS OF MODERATOR INVENTORY

If a spontaneous pressure tube rupture occurs, it can develop into a guillotine rupture. In such a case, the end-fitting could be ejected or displaced. If the ECCS was unavailable, the moderator would be needed as a heat sink for maintaining the integrity of the reactor core. However, the moderator liquid would partially drain through the ejected end-fitting, which would leave some fuel channels uncooled. Consequences could be as severe as widespread core damage. This event was not included in the safety report as licensees considered it to be very unlikely.

Licensees' experiments on fuel channel rupture, and review of the results by an expert consultant on behalf of the AECB, as well as actual station events, have indicated that pressure tube guillotine rupture may be more likely than was at first believed. AECB requested the licensees to properly analyse this dual failure event.

To support their statement that end-fitting ejection followed by loss of moderator is a low probability event, the licensees are currently performing experiments on the forces required to displace an end-fitting. They believe that the feeder loads, bearing friction and the channel bellows would restrict motion of the end-fitting sufficiently to prevent its ejection. The industry is also continuing to study fuel channel rupture behaviour, especially the conditions that could lead to calandria tube failure and pressure tube guillotine rupture.

10. COMPLIANCE WITH BUNDLE AND CHANNEL POWER LIMITS

The operating licence for each nuclear power station specifies the maximum channel and bundle power limits based upon a defined operating envelope. AECB staff has reviewed the processes that licensees use to demonstrate compliance with the channel and bundle power limits. The AECB review has identified various concerns about the consistency and accuracy of compliance analyses and procedures. AECB has asked all licensees to address its concerns. All licensees plan to reassess the uncertainties in the channel and bundle power calculations and modify the compliance procedures.

11. VOID REACTIVITY UNCERTAINTY ALLOWANCE IN LOCA ANALYSES

Early in a large loss of coolant event, the reactor power will increase due to coolant voiding. This increase in power must be stopped and kept to a minimum to limit the consequences of the event. Safety analyses use a calculated value for the amount of void generated in the accident. An AECB staff review identified a number of concerns regarding the uncertainties associated with void predictions. AECB asked licensees to address these concerns and, in the interim, to increase the safety margins in the void reactivity uncertainty allowance. A CANDU industry Research and Development (R&D) program has been undertaken to validate void reactivity calculations.

12. MODERATOR TEMPERATURE PREDICTIONS

During certain LOCA events, the integrity of fuel channels depends on the capability of the moderator to be the ultimate heat sink. A channel will likely fail if dry out on the calandria tube surface occurs. Calculations done to show that pressure tube integrity will be maintained depend on a number of computer codes. The AECB considers that moderator temperature predictions have not been adequately validated, given the tight safety margins that currently exist. The AECB believes that three-dimensional moderator circulation experiments are required to validate current analytical tools. The AECB has directed the licensees to perform these three-dimensional experiments.

13. FIRE PROTECTION FOR CANDU NUCLEAR POWER STATIONS

The AECB expects CANDU stations to be operated with minimal risk from fire, as fire can be a major risk contributor to the overall station safety. In 1996, the Canadian Standards Association (CSA) issued CSA standard N293-95; Fire Protection for CANDU Nuclear Power Plants. AECB staff considers that the nuclear industry should meet relevant sections of this standard, and has requested the licensees to assess the adequacy of their fire protection program against them.

14. FEEDER PIPE FITNESS-FOR-SERVICE

Inspections in several CANDU reactors have revealed unexpected reduction in the wall thickness of some outlet feeders. The rate of this degradation represents a departure from the original design predictions. When these findings were observed, feeder wall thickness was still adequate and were predicted to remain adequate for several years. The expected lifetime of some of the feeders, however, could be limited by the current rate of degradation. AECB asked the licensees to show that feeders are fit for service, and that they have sufficient understanding of the thinning phenomenon to prevent it from threatening the integrity of the feeders. Licensees have responded to the findings by forming an inter-utility project team.

ANNEX 7.1
SAMPLE POWER REACTOR OPERATING LICENCE

26-1-8-1-0
26-1-8-1-6

Ontario Hydro
700 University Avenue
Toronto, Ontario
M5G 1X6

AMENDMENT No. 1
to
POWER REACTOR OPERATING LICENCE 8/98
PICKERING NUCLEAR GENERATING STATION 'B'

Pursuant to subsections 27(1) and (2) of the Atomic Energy Control Regulations, Power Reactor Operating Licence 8/98, Pickering Nuclear Generating Station 'B', is hereby amended, in accordance with the request attached as Schedule A, as follows:

1. Condition A.A.4 is deleted and replaced by the following:

"A.A.4 The nuclear facility radiation emergency procedures shall be governed by and be in accordance with the document entitled "Consolidated OHN Emergency Plan", dated April 15, 1998, and prepared by Ontario Hydro. Amendments to this document shall require prior written approval of the Board."

The foregoing amendment is consolidated in the revised licence, Power Reactor Operating Licence 8.1/98 attached hereto as Schedule B, which is substituted in form for Power Reactor Operating Licence 8/98.

DATED at OTTAWA this _____ day of April, 1998.

ATOMIC ENERGY CONTROL BOARD

By: ____________________
J.D. Harvie
Director General
Directorate of Reactor Regulation

Amendment No. 1
Schedule B

26-1-8-1-0
26-1-8-1-6

Ontario Hydro
700 University Avenue
Toronto, Ontario
M5G 1X6

POWER REACTOR OPERATING LICENCE 8.1/98
PICKERING NUCLEAR GENERATING STATION 'B'

The Atomic Energy Control Board hereby authorizes Ontario Hydro to operate Pickering Nuclear Generating Station 'B' (hereinafter called "the nuclear facility"), located in the Town of Pickering in the regional municipality of Durham, Province of Ontario and described in "Pickering N.G.S. Site Plan" drawing number NK30-DOA-10200-0001, Rev. 7, dated September 1992 and "Pickering GS Site Security Taut-Wire Fence Layout and Survey" drawing number 8690-DOH-14100-1003, Rev. 5, dated September 1992, submitted under cover of a letter from Mr. R.O. Schuelke to Mr. G.R. Schwarz, dated September 30, 1992, and further described in:

(a) the report prepared by Ontario Hydro entitled "Pickering Generating Station 'B' Safety Report" dated February 1974, as revised in October 1997 (Vol. 1) and June 1997 (Vol. 2) (hereinafter referred to as the "Safety Report"), and

(b) the documents referred to in Ontario Hydro's application for this licence in the letter from G.C. Andognini to J.D. Harvie entitled "Renewal of Pickering NGS A & B Power Reactor Operating licences", dated 21 November, 1997.

The operation of the nuclear facility is subject to the conditions specified in Attachments "A.A", "A.B" and "A.C" to this licence, with which Ontario Hydro shall comply.

The Board further authorizes Ontario Hydro to possess, within the nuclear facility, the fissionable substances described in Schedule 1 of Attachment "B" to this licence, and prescribed substances other than fissionable substances insofar as necessary for, and in accordance with the operation authorized by this licence. Possession of the fissionable substances and the prescribed substances is subject to the conditions specified in Attachments "A.A", "A.B" and "B" to this licence, with which Ontario Hydro shall comply.

../2

-2- PROL 8.1/98

This licence comes into effect on the 1st day of April, 1998 and expires on the 31st day of March, 1999 unless sooner suspended or revoked.

DATED at OTTAWA, this _______ day of April, 1998.

ATOMIC ENERGY CONTROL BOARD

By: ________________________

J.D. Harvie
Director General
Reactor Regulation

ATTACHMENT "A.A" TO PROL 8.1/98

PICKERING NUCLEAR GENERATING STATION 'B'

CONDITIONS GOVERNING OPERATION OF THE NUCLEAR FACILITY

GENERAL REQUIREMENTS

A.A.1 Operation of the nuclear facility shall be governed by and be in accordance with the document entitled "Pickering Nuclear Generating Station 'B', Operating Policies and Principles" dated February 1982, as revised (R10) in December 1997, and prepared by Ontario Hydro and the documents entitled "Radiation Protection Policies and Principles", dated July 1993, and prepared by Ontario Hydro, and "Radiation Protection Regulations Part 1, Nuclear Facilities", Rev. 2, issued April 1995, and prepared by Ontario Hydro; these documents shall not be amended except on the written instruction of or with the prior written approval of the Atomic Energy Control Board, hereinafter referred to as "the Board".

A.A.2 Measures to ensure the physical security of fissionable substances and other prescribed substances and of the nuclear facility shall be maintained to the satisfaction of the Board.

A.A.3 i) The appointment of any person by the licensee to the position of Operations Manager at the nuclear facility is subject to the prior written approval of the Board.

Any person appointed to this position shall only delegate the authority or responsibilities of the position, as set out in the Operating Policies and Principles referred to in condition A.A.1, to another individual who has received the prior written approval of the Board to be delegated the authority of the position.

The appointment of any person by the licensee to any of the following operating positions at the nuclear facility is subject to the prior written authorization by the Board:

a) Shift Superintendent
b) Shift Supervisor
c) Shift Operating Supervisor
d) Authorized Nuclear Operator.

When requesting authorization by the Board to appoint a person to an operating position, the licensee shall submit evidence that the person has successfully completed the training programs, the examinations set by the Board and the co-piloting periods relevant to the position, and shall also submit a written recommendation from the Operations Manager.

ii) As set out in section 2.2.2 of the “Radiation Protection Regulations” referred to in condition A.A.1, no person shall be appointed as the responsible health physicist at the nuclear facility, or shall be permitted to exercise the authority of the position, without the prior written approval of the Board, except in cases where the “Radiation Protection Regulations” permit that authority to be exercised by a delegate.

iii) There shall be in attendance at all times sufficient qualified personnel to ensure the safe operation of the nuclear facility. The minimum station complement shall be as specified in Appendix 1 and Appendix 2 of the Pickering Operations Department Procedure P-ODP-2-1, entitled “Station Shift Complement”, dated 18 April, 1994, and prepared by Ontario Hydro. Except as otherwise approved in writing by the Board, there shall be in the nuclear facility at all times at least four Authorized Nuclear Operators, one Shift Operating Supervisor and one Shift Superintendent (or Shift Supervisor). At all times there shall be in the station main control room a minimum of two persons who have been authorized in writing by the Board and who are qualified to operate the controls unless, in the judgement of the senior operations person at the nuclear facility, the hazard to personnel would be unwarranted, in which case the reactors shall be placed in an assured shutdown and safe condition.

iv) Subject to subconditions A.A.3 i), ii) and iii), the staffing and organization of the nuclear facility shall be in accordance with the Pickering Station Instruction entitled “Station Organization and Responsibilities”, P-SI 0.6-8 dated May 1997, and prepared by Ontario Hydro. Significant changes in staffing and organization shall be reported to the Board at least 30 working days prior to being implemented and shall require prior written approval of the Board.

A.A.4 The nuclear facility radiation emergency procedures shall be governed by and be in accordance with the document entitled “Consolidated OHN Emergency Plan” dated April 15, 1998, and prepared by Ontario Hydro. Amendments to this document shall require prior written approval of the Board.

A.A.5 i) The total power generated in any one fuel bundle shall not exceed 750 kilowatts.

ii) The total power generated in any fuel channel shall not exceed 6100 kilowatts under normal steady-state operating conditions.

iii) The total thermal power from the reactor fuel shall not exceed 1744 megawatts under normal steady-state operating conditions.

A.A.6 The trip setpoints in shutdown system no. 1 and shutdown system no. 2 shall be maintained at values approved in writing by the Board.

A.A.7 The rate of release of prescribed substances from the nuclear facility shall be monitored and controlled and such releases shall not exceed the limits identified in the document entitled “The Derived Emission Limits for Radionuclides in the Airborne and Liquid Effluents from Pickering Nuclear Generating Station ‘B’”, HSD-HP-91-07, dated May 1992, and prepared by Ontario Hydro. Changes to this document shall require prior written approval of the Board.

A.A.8 Ownership and control shall be maintained by Ontario Hydro and no use shall be made, without prior written approval of the Board, of any land within the Exclusion Zone referred to in the Safety Report that is owned by Ontario Hydro as of the date of this licence, and within 914 metres of any reactor building.

A.A.9 Persons appointed under Section 12 of the Atomic Energy Control Regulations shall at all reasonable times be provided access to the nuclear facility and to all plans, drawings, documents and records pertaining to the design, construction, testing and operation of the nuclear facility.

A.A.10 i) Maintenance at the nuclear facility shall be of such a standard that, in the opinion of the Board, the reliability and effectiveness of all equipment and systems as claimed in the Safety Report and the documents listed in the application are assured.

ii) Notwithstanding i) above, maintenance shall be in accordance with the document entitled “Pickering Nuclear Division Maintenance Program Execution Plan for 1997 and 1998”, dated September 1996, and prepared by Ontario Hydro. The licensee may make revisions to this document during the term of this licence. Prior to implementation of any such revision, the licensee shall provide written notification and justification for the revision to the Board.

A.A.11 Operations, reports, tests, inspections, analyses, modifications, or procedural changes requested by the Board are to be completed expeditiously.

A.A.12 Except as otherwise directed in writing by the Board, all systems shall be tested at a frequency sufficient in the opinion of the Board to substantiate the reliability claimed or implied in the Safety Report or in the documents listed in the application.

REQUIREMENTS FOR PRIOR APPROVAL OF DESIGN MODIFICATIONS AND OPERATIONAL CHANGES

A.A.13 Except with the prior written approval of the Board, no change which would render inaccurate the descriptions and analyses in the Safety Report or in the documents listed in the application shall be made to the reactor shutdown system no. 1, shutdown system no. 2, the containment system, the emergency core cooling system or associated systems necessary for the proper operation of these systems.

A.A.14 Except with the prior written approval of the Board, no change shall be made in any equipment or procedure that could result in possible hazards different in nature or greater in probability or magnitude than those stated or implied in the Safety Report and in documents listed in the application.

A.A.15 i) Measures for the application of safeguards at the nuclear facility in accordance with the Treaty on the Non-Proliferation of Nuclear Weapons shall be established and maintained to the satisfaction of the Board.

ii) Except with the prior written approval of the Board, no action shall be taken which will interfere with the operation of equipment installed for or on behalf of the International Atomic Energy Agency in respect of the safeguards referred to in condition A.A.15 i).

iii) Except with the prior written approval of the Board, no change may be made to any aspect of the storage and handling of fuel or any equipment or procedures with respect thereto which could affect the safeguards referred to in condition A.A.15 i).

A.A.16 No fuel shall be loaded into the reactor unless the fuel design has been approved by the Board.

REPORTING REQUIREMENTS

A.A.17 Reporting shall be in accordance with AECB Regulatory Document R-99, entitled “Reporting Requirements for Operating Nuclear Power Facilities”, effective January 1, 1995, as amended from time to time.

REQUIREMENTS FOR RECORD KEEPING

A.A.18 Adequate records shall be kept of operation, maintenance, test results, periodic inspections, any occurrence which increased or could have increased the risk to persons, personnel radiation exposures, prescribed substances released from the nuclear facility, and disposition of prescribed substances to demonstrate compliance with the Atomic Energy Control Regulations and this licence.

A.A.19 A register of all current documentation relevant to the licensing of the nuclear facility shall be established and maintained by Ontario Hydro to the satisfaction of the Board.

REQUIREMENTS FOR BOARD STAFF

A.A.20 The licensee shall provide, at the facility in respect of which this licence is issued and at no expense to the Board, office space for employees of the Board who customarily carry out their functions on the premises of that facility (on-site Board staff). The licensee shall keep the office space of on-site Board staff separate from the remainder of the building in which it is located by walls, partitions or other suitable structures.

ATTACHMENT "A.B" TO PROL 8.1/98

PICKERING NUCLEAR GENERATING STATION 'B'

REQUIREMENTS FOR IN-SERVICE TESTING

A.B.1 In-service tests to measure the rate of leakage from the reactor buildings when subjected to full design pressure shall be carried out:

before 31 December 2000 in the case of Unit 7;
before 30 June 1999 in the case of Unit 6;
before 31 December 1999 in the case of Unit 8; and
before 30 June 2000 in the case of Unit 5;

except as otherwise approved in writing by the Board.

A.B.2 The internal structures of and components within the vacuum building shall be inspected at least once during any period of 10 years. The next inspection shall be carried out before 31 December 2000, except as otherwise approved in writing by the Board.

REQUIREMENT FOR PROGRESS REPORTS

A.B.3 Unless otherwise approved in writing by the Board, the licensee shall make a presentation to the Board, on or before 1 October, 1998, demonstrating that it is continuing to operate the nuclear facility safely, and that it is making progress in the work improvement programs.

REQUIREMENTS FOR RELIABILITY OF THE SITE ELECTRICAL SYSTEM

A.B.4 Unless otherwise approved in writing by the Board, the licensee shall submit on or before April 30, 1998, information to demonstrate that, with the shut down of the Pickering A reactors, the site electrical system is capable of meeting the reliability requirements for power supply to the high pressure injection pumps of the emergency core cooling system at Pickering B.

A.B.5 If the required reliability referred to in condition A.B.4 cannot be assured with less than three units operating, the licensee shall place the operating reactor(s) in a shutdown state approved in writing by the Board.

REQUIREMENTS FOR INFORMATION ON ENVIRONMENTAL PROTECTION POLICIES AND PROGRAMS

A.B.6 Unless otherwise approved in writing by the Board, in the interests of health and safety, the licensee shall submit on or before June 30, 1998, a summary of existing environmental protection policies and programs, including the results of these programs, pertaining to the operation of the nuclear facility. The report shall address both radiological and non-radiological aspects of environmental protection and shall include but not be limited to the following information:

(i) a summary of federal, provincial and municipal legal and regulatory requirements applicable to the nuclear facility;

(ii) an identification and summary of existing policies and programs relating to environmental protection and management;

(iii) an identification and summary description of public information and consultation programs, including any recommendations for changes to these programs, and related activities involving the use of independent expertise;

(iv) a summary review of the past and current environmental performance of the facility, including but not limited to:

- emissions monitoring program results
- environmental monitoring program results
- estimation of public doses
- identification of environmental stressors included in the review
- performance criteria and valued ecosystem components used in the analysis and interpretation of effects on the biophysical environment
- nature and significance of identified effects on the biophysical environment;

(v) a summary review of specific environmental issues arising from past operations at the nuclear facility, including those involving special assessment and potential remediation effort, and an identification of the potential effects on the biophysical environment; and

(vi) recommendations for additional assessment studies and for improvements to existing environmental programs to address the environmental performance and remediation issues identified in the review.

ATTACHMENT "A.C" TO PROL 8.1/98

PICKERING NUCLEAR GENERATING STATION 'B'

REQUIREMENTS FOR PRESSURE RETAINING COMPONENTS

For the purpose of this Attachment, "registered", "accepted" and "approval" means either by the Board or by an authority identified by the Board for that purpose.

A.C.1 The licensee shall design, manufacture, fabricate, procure, install, modify, repair, test, examine, inspect, or otherwise perform work related to vessels, boilers, systems, piping, fittings, parts, components and supports according to the technical requirements in CSA standards N285.0-95 and B51-95. Where indicated by these standards, the licensee shall obtain the following regulatory approvals for this work:

(a) registered designs;
(b) accepted overpressure protection reports;
(c) approval of applicable standards and code classification;
(d) registered welding and brazing procedures;
(e) qualified welders, welding operators, brazers, and examination personnel;
(f) accepted quality assurance programs; and
(g) accepted plans and procedures.

A.C.2 The licensee shall operate vessels, boilers, systems, piping, fittings, parts, components, and supports safely and keep them in a safe condition. The licensee shall:

(a) follow accepted work plans and procedures to test, maintain, or alter overpressure protection devices;
(b) comply with operating limits specified in certificates, orders, designs, overpressure protection reports, and applicable codes and standards;
(c) inspect and perform material surveillance according to the technical requirements in CSA standards N285.4-94 and N285.5-M90 and accepted schedules, plans and procedures;
(d) have any certified boiler or vessel that is in operation or use inspected and certified by an authorized inspector according to an accepted schedule; and
(e) ensure that vessels, boilers, systems, piping, fittings, parts, components and supports have markings as specified in the applicable standards.

A.C.3 The licensee shall keep records of regulatory approvals and other documents required under A.C.1, A.C.2 and the standards applicable to the work or equipment.

A.C.4 In addition to the reporting requirements under A.A.17, the licensee shall report promptly to the Board when it learns of any failure of a pressure boundary that has caused injury, death, or property damage.

ATTACHMENT "B" TO PROL 8.1/98

PICKERING NUCLEAR GENERATING STATION 'B'

CONDITIONS GOVERNING THE ACQUISITION, POSSESSION AND USE OF

PRESCRIBED SUBSTANCES

GENERAL REQUIREMENTS

B.1 Possession of fissionable substances is restricted to those listed in the attached Schedule 1.

B.2 Personnel supervising operations involving fissionable substances shall have been trained in the fundamentals of radiation protection and criticality control.

PHYSICAL SECURITY REQUIREMENTS

B.3 Access to fissionable substances shall be controlled and restricted to those persons authorized by Ontario Hydro.

REQUIREMENTS FOR TRANSFER

B.4 Transfers of prescribed substances in Canada shall be made only to persons who have previously agreed to the transfer and who are authorized by virtue of the Atomic Energy Control Regulations to possess such substances.

SCHEDULE 1 OF ATTACHMENT "B" TO PROL 8.1/98

PICKERING NUCLEAR GENERATING STATION 'B'

POSSESSION OF FISSIONABLE SUBSTANCES

ITEM NO.	FISSIONABLE SUBSTANCE	QUANTITY POSSESSION
1	natural and depleted fuel bundles	as required for operation
2	depleted uranium	as required for shielding
3	enriched uranium	as required for possession and use of fission chambers for the Pickering NGS-A SDS-E systems, (including spares)

ANNEX 7.2
SUMMARY OF MAJOR DESIGN AND OPERATIONAL CHANGES RESULTING FROM ATOMIC ENERGY CONTROL BOARD ACTIONS

This Annex provides examples of significant AECB regulatory actions. This includes instructions to the licensees by the AECB, as well as major undertakings or initiatives by licensees in response to AECB insistence and inquiry. The list is intended to illustrate the type of actions taken by the AECB that have resulted in either the enhancement of safety or in the avoidance of unsafe conditions. In most cases, these actions have either required extensive work by the licensees or resulted in significant operational penalties.

This list represents only a small portion of the total number of actions assigned to the licensees by the AECB. Consideration was given only to those regulatory actions related to nuclear power stations in operation at the time the action was taken. The large number of regulatory actions taken by the AECB about nuclear power stations in their design or construction phases, including stations intended for offshore use, are not included.

1. THE YEAR 1997

- Bruce B Power Restriction
  Bruce B reactors were derated from 94% to 90% full power operation after the AECB questions revealed an error in sumulating one of the empirical correlations in one of the safety analysis computer codes.

- Bruce A Fuel Reordering
  AECB inspections determined that physics procedures were not being followed and a suspension of fuel reordering (required to restore Bruce A to high power operation) was ordered until corrective action had been taken.

- Darlington Guaranteed Shutdown State
  AECB review determined that the concentration of gadolinium in the moderator system, which maintains the reactor in the guaranteed state, was non-conservative and Ontario Hydro Nuclear (OHN) was instructed to increase the amount substantially.

- Pickering Emergency Core Cooling System (ECCS)
  The AECB has required an evaluation of both equipment changes and human factors improvements to increase the reliability of the system, following a review of the Pickering Probabilistic Risk Analysis. OHN's position was that the risk was acceptable and no changes were required.

- Gentilly 2 Pressure Tube Inspection
  The AECB denied a request to continue operation in the light of new analysis which indicated that the pressure tubes might no longer meet the fitness-for-service guidelines.

2. THE PERIOD 1992-1996

- Bruce A and B Power Restrictions
  AECB questions in 1993 about the reactivity effects for the movement of fuel bundles during a postulated Loss Of Coolant Accident (LOCA) resulted in the Bruce reactors being derated to 60% full power operation. Since that time, OHN has had to implement a number of equipment and operational changes, and provide better technical support and analysis prior to raising to full power operation. These reactors are still derated to 90% full power operation (although Bruce was briefly allowed to operate at 94% full power).

- Bruce A Unit 2 Lay Up
  The AECB imposed criteria to make sure of the safety for continued operation of Unit 2 following the discovery of degraded steam generator tubes. OHN's decision to shut down Unit 2 was made following the questions raised by the AECB.

- Bruce A Unit 4 Preheaters
  Following the discovery of debris in the preheaters, the AECB denied OHN's request to return the unit to service until substantially more inspection and analysis work was completed. Operational restraints on fuelling were also imposed by the AECB as a result of the incident.

- Bruce B Steam Generators
  Following a review of the information relating to fretting degradation of the steam generator tubes, the AECB required a significant expansion of OHN's inspection and technical support activities.

- Bruce A Loss of Regulation Incidents
  The AECB set up an investigation following two successive failures of the reactor control system at Bruce A. This investigation led to significant operational and procedural changes to the reactor regulating systems at all Bruce units.

- Gentilly 2 Spent Fuel Transfer
  The AECB ordered that transfers of fuel from the spent fuel bay to the dry storage area be stopped until corrective measures were implemented in the management of the transfer activities.

- Pickering A Unit 2 LOCA
  The AECB requested that all units remain shut down until extensive corrective actions were completed. These included analysis and testing to confirm the relief integrity of the valves and associated piping of the primary heat transport system. Subsequently, design changes and improved maintenance were required on all CANDU reactors.

- Pickering A Reactor Building Leakage
  In 1992, the AECB required repairs to the dome of the Unit 1 reactor building to make sure safety margins were maintained.

- Pickering A Seismic Margins
  In 1993, the AECB required that OHN initiate a seismic margin assessment program and to implement any necessary station changes.

- Point Lepreau Steam Generators
  The AECB denied a request to increase power following an unexpected shutdown system trip. Subsequent inspection revealed major damage to the steam generator internals had occurred. Remedial action took several months to complete.

- Point Lepreau Containment Testing
  The AECB denied a request to extend the test interval for the Point Lepreau containment from three to five years.

- Safety Analysis Standards
  AECB reviews confirmed that both safety analysis standards and code validation were unsatisfactory (contradicting an internal review carried out by OHN at AECB’s request), indicating a lack of senior management control. This resulted in the introduction of improved quality assurance by OHN and an extensive program of code validation work by the nuclear power industry.

- Human Factors
  The AECB has required many actions to improve the human factors aspects of design and operation. Examples of individual issues addressed are:
  - root cause analysis to be carried out following events;
  - consideration of human factors in design (Bruce rehabilitation project; Pickering A Shutdown System Enhancement (SDSE), Pickering B digital trip meters, and Darlington shutdown system computer software redesign);
  - fuel handling changes at Bruce A and Bruce B;
  - the introduction of 12-hour shift working (see below).

- Fire Protection
  Following a detailed review of several CANDU nuclear power stations which revealed deficiencies related to fire protection, detection, and

mitigation, the AECB (in 1996) instructed all licensees to assess and upgrade the fire protection programs in their stations.

3. THE PERIOD 1987-1991

- Bruce A Station Ordered Shut Down
  In 1988, the AECB ordered the Bruce A station to be shut down because OHN failed to comply with a licence condition regarding the installation of a primary coolant system pump trip. The reactors were started only after commitments were made by OHN to complete the necessary modifications by a designated date.

- Bruce A Operating Practices Assessment
  An AECB review led to the introduction of routine station inspections (“rounds”) by operating staff at both Bruce A and B. Changes to the responsibilities of certain operators were made to improve field supervision.

- Bruce A Operational Complexity
  The AECB required that steps be taken to reduce the complexity of reactor operation. A major study by an independent consultant to OHN resulted in many design and procedural changes.

- Bruce A Shut Down
  OHN shut down Unit 1 when the AECB indicated that operating with a steam leak from the de-aerator was unacceptable. A cracked weld was subsequently discovered and repaired before the unit was placed back in service.

- Darlington Shutdown Systems Software
  Extensive verification was required by the AECB, resulting in a delay in the start-up date of the first unit by five months. OHN was subsequently requested to rewrite the software.

- Pickering A Shutdown System Enhancement (SDSE)
  The AECB’s rejection of analysis carried out by OHN for Pickering A in 1990 led to an order in 1993 to make major improvements to the reactor shutdown systems. These changes were to be completed no later than December 31, 1997, or the reactors were to be shut down.

- Pickering Relief Duct Seals
  Testing of elastomeric seals, as requested by the AECB, led to a change to an improved material. The AECB also requested the installation of a backup seal to be installed.

- Pickering Over-exposure Incident in 1990
  The AECB successfully prosecuted OHN on the basis of an investigation carried out following a radiation over-exposure during removal of cobalt adjuster rods.

- Pressure Tubes
  The AECB instructed licensees to avoid operation of reactors under conditions where the pressure tubes could form hydride blisters. This led to a large engineering program to inspect pressure tubes and make sure that spacers between the pressure tubes and calandria tubes were positioned properly.

- 12-Hour Shift Working at OHN Power Stations
  The introduction of 12-hour shift working was significantly delayed by AECB requirements to conduct extensive trials. As a result of these trials, the shift schedules were modified.

4. BEFORE 1987

- Shutdown Systems
  The AECB instructed that Bruce A and all subsequent reactors be fitted with two fully effective shutdown systems. Subsequently, improvements to the Pickering A shutdown systems were also requested (see above).

- Emergency Core Cooling Systems (ECCS)
  Bruce A reactors were back fitted with a high pressure emergency cooling system and heat exchangers at the request of the AECB. High pressure emergency cooling systems and heat exchangers were subsequently incorporated into all subsequent reactors.

  The AECB also requested that major shielding be added to the Bruce A ECCS. This request has also led to a redesign of the Bruce B ECCS, then under construction.

- Containment Systems
  Tests ordered by the AECB at Bruce A revealed that designs of dousing system headers were inadequate, and would require a major redesign. These tests also indicated that changes were necessary at Pickering, and these were back fitted.

  The AECB required major improvements to the emergency filtered air discharge systems at both Pickering and Bruce.

  Improvements to the devices isolating the Pickering A units from the main containment duct were required. These changes were required to meet a Board request to increase the time at which venting the Pickering containment becomes necessary following an accident.

- Pressure Tubes
  The AECB rejected OHN's case for restarting Pickering Units 1 and 2, following the 1983 pressure tube failure. The reactors consequently remained shut down until their pressure tubes had been replaced several years later.

  The AECB imposed requirements for monitoring and inspecting ageing pressure tubes, which went far beyond the industry's own proposals. Experience has demonstrated that these requirements have placed a high work load on the licensees, but that they were fully justified.

- Powerhouse Design and Environmental Qualification
  AECB questions on the effects of high pressure piping failures in the powerhouse at Bruce A and B resulted in:
  - a major program of work on all stations, including the installation of large pressure relieving devices in Bruce, Pickering, and Point Lepreau;
  - the back-fitting of a qualified electrical power supply at Bruce A;
  - a major environmental qualification program on all OHN nuclear power stations.

- Gentilly-2 Reactor Start-Up
  The AECB delayed the start-up of the Gentilly-2 reactor for several months in 1983 because of a shortage of qualified operators.

ANNEX 8.1
ATOMIC ENERGY CONTROL BOARD
STAFF ORGANIZATION

The major divisions of the AECB are:

- the President's Office
- the Secretariat
- the Directorate of Corporate Services
- the Directorate of Fuel Cycle and Materials Regulation
- the Directorate of Reactor Regulation
- the Directorate of Environmental and Human Performance Assessment

The AECB staff organization chart is shown in Figure 8A.1.

1. THE PRESIDENT'S OFFICE

The President's Office provides administrative support services directly to President.

2. THE SECRETARIAT

The Secretariat consists of the New Act Implementation Group, the Board Services Group, the External Relations and Documents Division, the Non-Proliferation, Safeguards and Security Division, and the Communications Division.

- The New Act Implementation Group is responsible for implementation of the Nuclear Safety and Control (NSC) Act and new regulations under the Act.

- The Board Services Group makes sure that the five-member Board has the administrative and technical support it needs to function efficiently and effectively. It also provides the following functions:
  - manages the Board meeting process;
  - provides recording services at Board meetings;
  - drafts minutes of Board meetings;
  - drafts and coordinates responses to submissions to the Board;
  - prepares announcements of Board deliberations and decisions;
  - provides scientific and administrative support to the Advisory Committees and the Group of Medical Advisers (GMA);
  - drafts policies, procedures and rules on Board operations.

- The External Relations and Corporate Documents Division performs the following functions:
  - manages AECB interactions with the Minister's Office;
  - manages AECB relations, agreements, cooperation, and involvement with various international organizations, agencies, and governments;

- administers the AECB nuclear emergency preparedness plan in cooperation with federal, provincial and municipal agencies;
- administers AECB compliance with the federal Access to Information and Privacy Acts.

In addition to designing and implementing frameworks and work processes to produce and manage corporate regulatory documents, the division coordinates the preparation and management of AECB corporate documents.

- The Non-Proliferation, Safeguards, and Security Division which:
  - makes sure that the AECB meets its domestic and international obligations about nuclear non-proliferation, safeguards and the physical security of nuclear facilities' materials, and technology;
  - advises the federal Department of Foreign Affairs and International Trade on matters relating to the development and implementation of Canada's nuclear non-proliferation and export control policies;
  - administers Canada's bilateral nuclear cooperation agreements;
  - implements the agreement between Canada and the International Atomic Energy Agency (IAEA) on the application of safeguards in Canada;
  - manages the Canadian Safeguards Support Program;
  - issues licences for the export and import of nuclear items;
  - makes sure there is compliance with the Physical Security Regulations.

- The Communications Division which:
  - provides information and publishing services on behalf of the AECB;
  - responds on an as-required basis to oral and written enquiries, and requests from AECB staff, the public, and the news media;
  - issues news releases, notices, and information bulletins concerning regulatory developments, nuclear safety, and Board decisions;
  - generates, publishes, stocks, and distributes documents that describe nuclear technology, the nature and effects of radiation, the organization, mandate and activities of the AECB, Board policies and decisions, the results of AECB-funded research, the conclusions and recommendations of AECB Advisory Committees, nuclear legislation, and AECB regulatory requirements and expectations.

3. THE DIRECTORATE OF CORPORATE SERVICES

The Directorate of Corporate Services manages the AECB's human, financial, physical, and information resources. It makes sure that the AECB complies with the Official Languages Act, the Employment Equity Act, the Financial Administration Act, the Government Employees Compensation Act, the Canadian Human Rights Act, and the Public Service Staff Relations Act. The Directorate administers the AECB's security program and Conflict of Interest and Post-Employment Code.

The Directorate of Corporate Services consists of the Human Resources Division, the Finance Division, and the Information Management Division.

- The Human Resources Division provides specialist support to the AECB in all areas of the human resources field, including:
  - planning
  - policy development
  - staffing
  - compensation
  - staff relations

  The Human Resources Division also represents the AECB in interactions with federal bodies such as the Treasury Board, the Human Rights Commission, and the Public Service Staff Relations Board in matters that relate to human resources.

  The Non-Technical Training Unit of the Human Resources Division provides various services related to the non-technical training needs of AECB staff, as follows:
  - assesses staff training needs, and recommends programs to meet these needs;
  - provides related advice and information to AECB staff;
  - administers the corporate budget for non-technical training;
  - coordinates the delivery of non-technical training programs;
  - maintains lists of trainers and records of employees' training.

- Within the AECB Directorate of Corporate Services, the Finance Division provides financial services on behalf of all employees and units in such areas as:
  - planning
  - budgeting
  - purchasing
  - expenditure tracking and control
  - accounts receivable
  - accounts payable
  - contract administration
  - travel planning and accounting
  - office accommodations
  - supplies and services
  - cost recovery

  The Finance Division also participates in the development or revision of financial policies, produces reports for AECB staff and management, and interfaces with other central agencies, such as Treasury Board, on financial matters.

The Finance Division also:
- administers the AECB's corporate security program on behalf of the President;
- obtains security clearances for AECB staff in accordance with federal government policies;
- makes sure that physical security systems to protect AECB property and information systems are developed and maintained;
- investigates potential or actual breaches of security within the AECB.

- The Information Management Division performs the following functions:
  - operates the AECB Records Office and the AECB Library;
  - administers and maintains the AECB's electronic systems for information management and exchange;
  - provides technical support services on related equipment and systems to AECB staff and work units.

4. THE DIRECTORATE OF FUEL CYCLE AND MATERIALS REGULATION

The Directorate of Fuel Cycle and Materials Regulation regulates the construction, operation, and decommissioning of uranium mining facilities: mines, mills, refineries, and conversion plants, in addition to radioactive waste management facilities. The Directorate provides laboratory and compliance services for various AECB activities, and regulates accelerators, radioisotope production and use, research and test facilities. In addition, it regulates the transport packaging of radioactive materials, and regulates the decommissioning of all nuclear facilities, including nuclear reactors, as defined in the AEC Regulations.

The Directorate of Fuel Cycle and Materials Regulation consists of the Uranium Facilities Division, the Wastes and Decommissioning Division, the Materials Regulation Division, and the Research and Production Facilities Division.

- The Uranium Facilities Division makes sure that uranium mines and uranium processing facilities are safely:
  - designed
  - developed
  - constructed
  - operated
  - maintained
  - decommissioned

  To promote compliance with applicable legislation, licences, and standards, staff:
  - assesses applications, reports, and submissions;
  - conducts surveys, inspections, audits, reviews, and investigations;
  - recommends appropriate licensing and enforcement actions.

  The Director and Uranium Mines Section of the Uranium Facilities Division operate from an AECB regional office in Saskatoon, Saskatchewan.

- The Wastes and Decommissioning Division makes sure that radioactive waste facilities are designed, developed, constructed, operated, and maintained safely, and all nuclear facilities and activities are decommissioned safely. To promote or verify compliance with applicable legislation, licences, and standards, staff:
    - assesses applications, reports, and submissions;
    - conducts or participates in surveys, inspections, audits, reviews, and investigations;
    - recommends appropriate licensing and enforcement actions.

- The Materials Regulation Division (MRD) regulates the possession and use of radioisotopes in education, medicine, research, and industry to ensure the health and safety of workers and the public, and the protection of the environment. MRD staff:
    - assesses licence applications;
    - prepares and issue licences;
    - inspects uses of radioactive materials;
    - drafts regulatory standards.

    Inspection and compliance services are provided by staff located in:
    - Ottawa, Ontario
    - Laval, Quebec
    - Mississauga, Ontario
    - Calgary, Alberta

- The Research and Production Facilities Division supplies laboratory services for all AECB activities. It also makes sure that non-medical accelerators, research reactors, nuclear research and test establishments, and radioisotope production facilities are designed, developed, constructed, operated, maintained, and decommissioned without undue risk to health, safety, and the environment. Staff also:
    - assesses submissions;
    - conducts or participate in audits, inspections and investigations;
    - makes recommendations regarding performance, licensing, or enforcement;
    - drafts regulatory standards.

5. THE DIRECTORATE OF REACTOR REGULATION

The Directorate of Reactor Regulation evaluates and regulates the safety of nuclear power reactors. The Directorate consists of the Power Reactor Operations Division, the Power Reactor Evaluation Division, the Safety Evaluation Division “Analysis”, and the Safety Evaluation Division “Engineering”.

- The Power Reactor Operations Division regulates the construction, commissioning, and operation of nuclear power station installations on a day-to-basis. To promote or verify compliance with applicable legislation, licences, and standards, staff:

- assesses applications, reports, and submissions;
- conducts or participates in surveys, inspections, audits, reviews, and investigations;
- recommends appropriate licensing and enforcement actions.

• The Power Reactor Evaluation Division evaluates the performance of nuclear power reactor installations in Canada. Staff:
  - manages technical reviews of issues that the AECB considers pertinent to the safe design, construction, commissioning, operation, and maintenance of Canadian power reactors;
  - coordinates review activities, reports findings, and recommends follow up.

• Safety Evaluation Division "A" participates in safety evaluations of nuclear power reactors and other facilities regulated by the AECB. The division provides technical expertise in the areas of:
  - thermal hydraulics
  - reactor physics
  - reactor protection
  - plant behaviour

  Staff conducts or participates in assessments, reviews, inspections, and audits involving their areas of expertise.

• Safety Evaluation Division "E" also participates in safety evaluations of nuclear power reactors and other facilities regulated by the AECB. The division provides technical expertise in the areas of:
  - reliability and risk assessment;
  - control and instrumentation;
  - electrical systems;
  - civil engineering;
  - pressure-retaining components and systems.

  Staff conducts or participates in assessments, reviews, inspections, and audits involving their areas of expertise.

6. DIRECTORATE OF ENVIRONMENTAL AND HUMAN PERFORMANCE ASSESSMENT

The Directorate of Environmental and Human Performances Assessment assesses the adequacy of the radiation safety and environmental protection measures proposed by AECB licensees and by applicants for AECB licences. The Directorate also conducts audits of the radiation training and safety programs that are in place at AECB-licensed facilities, certifies key operating personnel at nuclear power reactor installations, administers the AECB's mission-oriented research and support program, and delivers the AECB technical training program.

The Directorate of Environmental and Human Performances Assessment consists of the Radiation and Environmental Protection Division, the Personnel Qualification Assessment Division, the Performance Evaluation Division, the Research and Support Group, and the Technical Training Group.

- As requested by the Board or AECB licensing divisions, the Radiation and Environmental Protection Division conducts specialist assessments of the adequacy of radiation safety and environmental protection proposals and programs. The division:
  - initiates, organizes, and manages audits of existing radiation safety and environmental protection measures;
  - manages the licensing and inspection of companies that provide radiation dosimetry services to AECB licensees;
  - assesses the emergency preparedness plans of AECB licensees;
  - participates in external federal and provincial environmental assessment projects;
  - drafts policies, procedures, and criteria to assure and assess the adequacy of radiation safety and environmental protection programs.

- The Personnel Qualification Assessment Division is responsible for making sure that those persons required by nuclear legislation to be qualified are competent, and maintain their competence through continuing training. The division assesses the competence of key personnel retained by AECB licensees, and the adequacy of the corresponding employment standards and employee training programs. In cooperation with other AECB units, the Personnel Qualification Assessment Division develops standards and criteria to define the necessary qualifications and training of licensees' staff.

- The Performance Evaluation Division assesses the performance of AECB-licensed operations and activities, with respect to health, safety, and protection of the environment. The division identifies the human factors, quality assurance, and other issues that could affect safety; evaluates the implications of these factors and issues, and recommends appropriate follow up actions. In cooperation with other AECB units, the Performance Evaluation Division develops regulatory guidance on:
  - human factors policies and programs;
  - operational processes and methods;
  - personnel systems.

- The Research and Support Group administers the AECB mission-oriented research and support program. Staff:
  - plans and develops this program on the basis of AECB needs;
  - manages its budget;
  - monitors and directs the progress of individual research projects;
  - coordinates input from AECB sponsors and clients.

- The Technical Training Group develops, delivers, and reports upon training programs to meet the technical needs of AECB staff and the needs of other nuclear regulatory agencies.

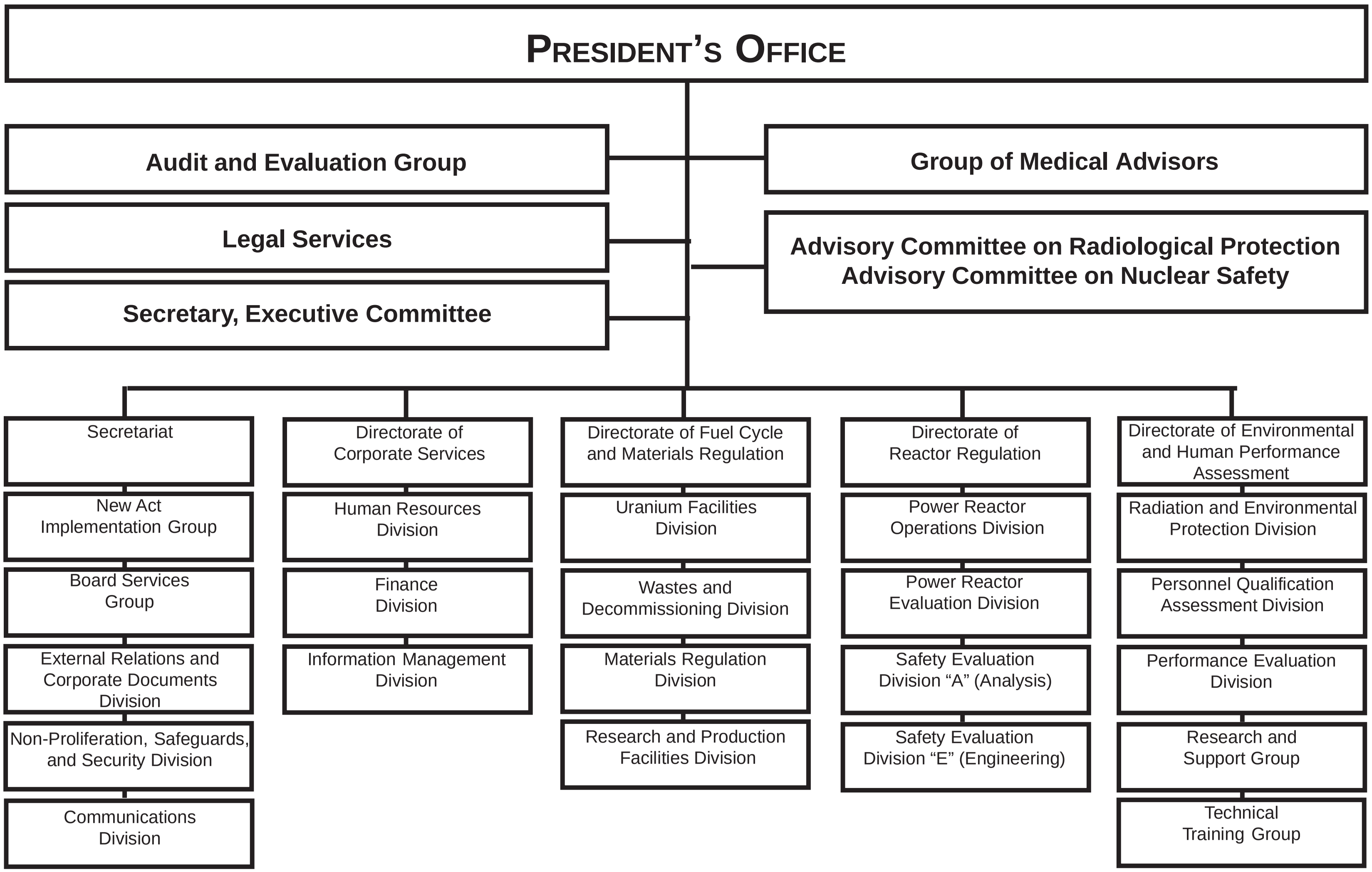

Figure 8A.1: AECB Staff Organizational Chart

ANNEX 16.1
NUCLEAR EMERGENCY PLANS IN CANADA

The following are brief summaries of the on-site emergency plans at the nuclear power stations, the off-site emergency plans at the provincial level, and the Atomic Energy Control Board (AECB) emergency preparedness and response plans.

1. ON-SITE EMERGENCY PREPAREDNESS AND RESPONSE PLANS AT NUCLEAR POWER STATIONS

1.1 Ontario Hydro Stations' Nuclear Emergency Plan

The Ontario Hydro Nuclear (OHN) Emergency Plan is a corporate-level plan, which serves as the common basis of site-specific nuclear emergency preparedness and response arrangements at OHN's Bruce, Darlington and Pickering stations. It describes concepts, structures, roles, and processes to implement and maintain an effective OHN response to radiological emergencies that could endanger on-site staff, the public, or the environment. It is designed to be compatible with the Province of Ontario Nuclear Emergency Plan.

The OHN Emergency Plan defines a nuclear power plant emergency to be a sudden unexpected occurrence of unusual radiological conditions that have the potential to expose staff or public to radiation in excess of regulatory limits.

The OHN Plan focuses on the release of radioactive materials from fixed facilities, and on OHN interfaces with the Province of Ontario Nuclear Emergency Plan. The formal scope of the Plan excludes hostile (security) action incidents at OHN nuclear plants, as these incidents are dealt with in detail in other OHN documents. However, the Plan's provisions regarding potential releases of radioactive materials also apply to security incidents. These include the requirements for off-site notifications, situation updates, and confirmations of any radioactive releases.

To augment its corporate Emergency plan, OHN has developed site-specific nuclear emergency preparedness and response arrangements for its Bruce, Darlington, and Pickering stations. These emergency plans are consistent with the corresponding OHN nuclear safety analyses and reports that were provided to the AECB in support of construction and operating licences applications.

In the event of an on-site nuclear emergency at an OHN power plant, OHN staff would immediately classify the nuclear emergency in accordance with criteria specified in the station emergency plan. Should this emergency have off-site implications, OHN staff further categorizes it according to criteria contained in the Province of Ontario Nuclear Emergency Plan. To simplify this step, the events identified in the OHN nuclear power plant's Abnormal Incidents Manual have been categorized according to the Province of Ontario notification designations. The result of this categorization exercise is Appendix D, "Notification Criteria Matrix", of OHN's corporate Emergency Plan. The site-specific emergency response plans for Bruce, Pickering, and Darlington include derivatives of Appendix D of the OHN Emergency Plan.

Emergency drills and exercises are part of OHN's overall process of program assessment, as discussed in sections 7.0 and 9.0 of the corporate Emergency Plan. These exercises are conducted periodically at all OHN power installations, in cooperation with other organizations and jurisdictions that have an interest in nuclear emergency preparedness and response.

OHN maintains emergency public response capabilities within its Nuclear Public Affairs Department and the Ontario Hydro Corporate Affairs Department. The primary targets of OHN's nuclear emergency public information program are the public who live or work near OHN nuclear power plants, and select OHN employees and contacts who need to know. In the event of a nuclear emergency involving an OHN facility, OHN emergency response procedures and agreements require the corporation to coordinate its public information efforts and activities with those of other participating jurisdictions or organizations, such as provincial agencies operating within the framework of the Ontario Provincial Nuclear Emergency Plan (PNEP).

The OHN public affairs response in a given emergency will depend upon the related circumstances.

For events that are not severe enough to warrant activation of the PNEP, but may interest neighbours and other stakeholders, OHN issues news releases or verbal briefings to the local media, with copies to provincial and municipal officials. If the situation warrants, OHN may activate its on-site or near-site Hydro Media Centre (HMC) for briefing or interview purposes.

More severe events may require the activation of the PNEP and its Joint Information Centre (JIC). Pending activation and operation of the JIC, OHN's emergency response organization will, on an interim basis, communicate relevant information to the public and the media. With the JIC in operation, the provincial government assumes control of information services regarding the incident.

OHN provides training, financial, and personnel assistance to the JIC.

1.2 Gentilly-2 NGS Nuclear Emergency Plan

The Hydro-Québec publication, "Plan des mesures d'urgence", describes the utility's arrangements to cope with actual or potential nuclear emergencies at its Gentilly-2 nuclear power plant. This publication and various supporting documents define the Gentilly nuclear emergency preparedness and response plan in detail, including application criteria, roles and responsibilities, requirements for coordination, classification of emergency alerts, communications with the media and the public, emergency procedures, response logistics, technical and equipment support, and emergency training and drills.

The above plan stipulates that abnormal on-site events that increase the radiological risk to employees, the public, or the environment shall be announced by the declaration of an appropriate level of radiation alert, indicating the severity or potential severity of the incident.

An area alert is to be declared when the radiation field or concentration of airborne contamination over a localized on-site area increases to 2 to 10 times normal levels, or when these risks are increasing unusually rapidly.

A site alert is to be declared when radiological conditions pose a general, significant risk to Gentilly site personnel.

A general alert is to be declared following radiological releases in excess of regulatory limits, or after releases that could result in radiation exposures in excess of dose limits.

Should abnormal events or conditions at Gentilly-2 lead to a potential, or an actual, off-site nuclear emergency, the “directeur du Comité de gestion du centre d’urgence d’Hydro-Québec[1]" is responsible for notifying the “Organisation de la Sécurité Civile du Québec (OSCQ)” of the threat or emergency. The “OSCQ” would lead any necessary off-site nuclear response, as discussed above.

As a follow-up to the radiation alerts described above, Hydro-Québec management, the “Groupe Communications et relations avec le milieu (GCRM)” in nearby Trois-Rivières, and communications staff at the Gentilly-2 emergency centre cooperates to provide information to site personnel, the public, and the media. In the case of a General Alert, the “GCRM” move to the “Centre de coordination des communications de l’OSCQ” where “Communication-Québec” coordinate all public relations for the “Gouvernement du Québec”.

The Gentilly-2 plant conducts radiation emergency drills at least once per year, as required by its “Directives de Santé et Normes de Radioprotection”. It also participates in externally organized drills, in cooperation with international, national, and provincial agencies and organizations.

Gentilly-2 managers, staff and workers receive both basic and specialized instruction in nuclear emergency preparedness and response, on an as-required basis.

1.3 Point Lepreau NGS Nuclear Emergency Plan

New Brunswick (NB) Power includes its response arrangements for on-site nuclear emergencies at its Point Lepreau Generating Station (PLGS) in the group of documents it terms the PLGS “On-site Contingency Plan” (OCP). These documents describe NB Power’s planned responses to specific types of emergencies at the Point Lepreau nuclear power plant and associated facilities, responsibilities, agreements, training, and procedures.

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1. Référence: Centrale nucléaire Gentilly 2 - Document de référence - DR-32/Rev. 4 / Plan des mesures d’urgence (Plan de base); Octobre 1997

The overall objective of the “On-site Contingency Plan” is “to control and ameliorate” the consequences of abnormal events at the Point Lepreau site in order to protect workers, the public, and site property. The Plan is made up of a “General Plan” and “Specific Contingency Plans”. These plans interface with each other, and with off-site provincial contingency plans.

The “General Plan” is the overview and coordinating document for four types of emergencies: radiation, fire, medical, and chemical. It describes the general organization, responsibilities, and preparatory measures that apply to all situations.

The “Specific Contingency Plans” consist of four separate plans that describe the detailed arrangements and specific actions and arrangements required, in addition to those of the “General Plan”, to deal with the threat or occurrence at Point Lepreau of each of the four types of potential emergencies. The detailed procedures for implementation of the four plans are consolidated in PLGS Operating Manual OM-78600.

The PLGS “Radiation Contingency Plan” (RCP) deals with the potential or actual radiation consequences of an abnormal event at the plant. It is designed to interface with the provincial off-site nuclear emergency plan, the “Point Lepreau Nuclear Generating Station Off-site Emergency Plan”. The off-site plan deals only with actions outside the PLGS boundary. The PLGS on-site plan deals with measures to address on-site radiation incidents and to minimize off-site impacts. On-site and provincial plans, therefore, are designed to operate independently but in harmony, with a high degree of liaison and cooperation. For example, Point Lepreau personnel are responsible for providing the information that could lead to initiation of the corresponding provincial plan.

The Point Lepreau RCP defines two levels of response: Alert and Emergency. Minor contingencies that are not significant enough to warrant either an Alert or Emergency response are dealt with using normal plant operating procedures.

The Alert response serves events that pose no immediate general threat either on or off the site but that warrant prompt action using on-site, and possibly outside, resources. Examples include minor spills or releases of radioactive materials.

2. PROVINCIAL OFF-SITE EMERGENCY PREPAREDNESS PLANS

2.1 Province of Ontario

The Province of Ontario Nuclear Emergency Plan extends to all situations where there is an actual or potential hazard to public health, property, or the environment from ionizing radiation or from a nuclear facility. By definition, these situations include:

- accidents at nuclear installations[2];
- accidents at nuclear establishments[3];
- accidents during the transport of radioactive material;
- the loss of control over radioactive material;
- hostile action directed at a nuclear facility[4], or involving radioactive material.

Typically, the Province of Ontario Nuclear Emergency Plan does not deal with the following:

- an accident in which the effects, both actual and potential, are expected to be confined within the boundaries of the nuclear facility;

- an accident with effects so localized that their impact can be satisfactorily dealt with by local emergency response personnel (police, fire), with or without outside technical assistance.

The containments for the multi-unit nuclear power stations in Canada are connected to a common vacuum building. This design is unique to the province of Ontario, and has fundamental bearing on Ontario's approach to emergency preparedness and response arrangements.

2.2 Province of Quebec

Within the province of Quebec, the "Organisation de la Sécurité Civile du Québec (OSCQ)" has lead responsibility for emergency planning and response to nuclear emergencies regarding their impact outside the bounds of the Gentilly site. The plan of OSCQ to cope with such emergencies is described in the document "Plan des mesures d'urgence nucléaire externe à la centrale nucléaire Gentilly 2 (PMUNE-G2)". This Plan conforms with Quebec legislation, "la Loi sur la protection des personnes et des biens en cas de sinistre".

The "PMUNE-G2" is intended to prepare government agencies to react to an emergency situation at the Gentilly site, for purposes of minimizing the effects of accidents, protecting and assisting the public, and providing assistance to municipalities.

In the event of an accident at the Gentilly-2 nuclear power plant with significant off-site implications, Hydro-Québec and the "OSCQ" have separate but complementary responsibilities for emergency planning and response. For example, the Gentilly-2 Shift Supervisor is responsible for recognizing and declaring the appropriate level of radiation alert. In the case of a site or a general alert (see section 4 under "Gentilly-2 Nuclear Power Plant), the Shift Supervisor informs the "Direction de la Sécurité Civile du Québec du Ministère de la Sécurité publique". Depending on the urgency of the

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2. A facility or vehicle containing a nuclear fission or fusion reactor.
3. A facility that uses, produces, processes, or disposes of a nuclear substance, or a vehicle that carries such a substance, but does not include a nuclear installation.
4. A nuclear installation or an establishment.

emergency, the “OSCQ” will either assume a standby, or initiate an off-site emergency response in accordance with the “PMUNE-G2”.

As part of its off-site emergency response, the “OSCQ” would establish a centre to coordinate various elements of the response, including communications and public relations activities. Typically, this centre would issue any necessary safety advisories to the public (such as those concerning the need for confinement or evacuation), respond to media enquiries, and coordinate the administration of precautionary measures.

2.3 Province of New Brunswick

Under the Emergency Measures Act, the New Brunswick Emergency Measures Organization (NB EMO) of the provincial Department of Municipal Affairs has the lead responsibility to develop provincial emergency action plans, and to direct, control, and coordinate emergency responses.

The New Brunswick Emergency Plan, prepared by NB EMO defines an emergency to be any abnormal situation requiring prompt action beyond normal procedures to limit damage to persons, properly, or the environment. The stated aim of the Plan is to designate responsibility for actions to mitigate the effects of any emergency, other than war, in the Province of New Brunswick.

The Plan defines the lead responsibilities of the Department of Municipal Affairs and the supporting roles of some twenty-three departments, agencies, or organizations. Representatives of these players comprise the Provincial Emergency Action Committee (PEAC). The PEAC directs, controls, and coordinates provincial emergency operations, and assists and supports municipalities as required.

The PEAC maintains two states of readiness, Standby and Emergency as follows:

- The Standby State is a state of readiness that requires representatives of departments to be available on call.

- An Emergency State is a state where action by EMO and/or other departments is required. During an Emergency State, departmental representatives are called to headquarters and briefed on the corresponding emergency.

The province of New Brunswick is divided into eleven EMO Districts, coinciding with the eleven Municipal Services Regions. The regional municipal services representatives of the Department of Municipal Affairs are designated EMO District Coordinators. EMO District Coordinators stimulate the development and refinement of emergency planning by municipalities, and provide advice and assistance on the development of emergency plans. They coordinate the use of provincial resources to deal with emergency situations in rural areas and urban municipalities. To accomplish this, District Emergency Committees are formed along lines similar to Provincial Action Committees. Their main objective is to provide assistance to municipalities and the populace of unincorporated areas. They consist of representatives from the departments of:

- Municipal Affairs
- Environment
- Health
- Justice (RCMP)
- Natural Resources
- Social Services
- Transportation

Local authorities are responsible for emergency planning and response within their physical boundaries, and in some cases for certain areas outside their boundaries. Communities may assist each other in accordance with mutual aid agreements. When, however, an emergency arises in which the resources of a community, or group of communities are insufficient, the Province will provide assistance through the district Emergency Committee.

Where required and possible, District Emergency Operations Centres are established in the Regional Department of Transportation Office. Also, NB Power, owner and operator of the Point Lepreau nuclear power plant, maintains an Off-site Emergency Centre (OEC). This facility will be occupied by company staff following an incident involving abnormal radiation releases to the off-site environment. In such incidents, the OEC will be used as a communications centre, and to direct radiation monitoring programs.

The PLGS Off-site Emergency Plan was developed by the NB OME, in accordance with the framework described above. It delineates the roles and responsibilities of, and the immediate actions to be taken by, those involved if an incident at the Point Lepreau nuclear creates an off-site emergency.

Section A of the plan lists the names and responsibilities of the provincial and federal agencies that may have a role to play. Representatives of these potential players comprise the Control Group. Under the direction of the NB EM, the Control Group coordinates implementation of the plan.

The PLGS Off-site Emergency Plan classifies potential off-site emergencies as Type A or Type B as follows:

- Type A incident is defined to be an emergency that can be handled by on-site resources, and does not present a danger to the general public.
- Type B incident is one that is dangerous to the general public.

If it is necessary to alert the public to the occurrence of an off-site emergency, strategically placed sirens will sound to warn residents. To augment the fixed sirens, police may use their car-sirens and public address systems to alert the public. Wardens will oversee designated areas to ensure residents are appropriately informed of any actions required of them. Radio, TV, and wardens will advise the public of the need for an protective actions. Arrangements are in place to help individuals who may require physical assistance should evacuation prove necessary. Full details are provided in Volume 2 of the New Brunswick plan.

3. THE AECB EMERGENCY PREPAREDNESS AND RESPONSE PLANNING

3.1 Description of the AECB Emergency Response Plan

The AECB Emergency Response Plan is the document that describes the strategies and guidelines that the AECB will follow to cope with a nuclear emergency. It describes:

- emergency situations that could require AECB involvement
- the role of the AECB in nuclear emergencies
- the role of interfacing parties
- the AECB emergency preparedness organization
- the concept of operations
- the AECB equipment infrastructure
- preparedness and training requirements and exercises

The plan is issued under the authority of the President of the AECB, in accordance with the objectives of the AEC Act and regulations and the federal Emergency Preparedness Act. It is designed to provide a compatible interface with the emergency plans and procedures of AECB licensees, provincial governments, the federal government, and international organizations. The plan draws upon provisions of the Transport Packaging of Radioactive Materials Regulations and the Transportation of Dangerous Goods Act and regulations, and includes formal agreements with various organizations and jurisdictions.

Ultimately, implementation of the AECB Emergency Response Plan in the event of a declared emergency could involve:

- the AECB emergency organization shown in Table 16A.1;
- AECB employees;
- AECB licensees;
- transporters, shippers and others involved in, or affected by, the transport of radioactive materials;
- departments and agencies of the national government;
- departments and agencies of the provincial government;
- news media organizations;
- the United States Nuclear Regulatory Commission;
- the International Atomic Energy Agency.

The AECB plan applies to all nuclear emergencies or potential nuclear emergencies that could require the regulatory or technical involvement of the AECB. It is in effect at all times, in one of four operating modes: normal, standby, activated, or recovery.

- In the normal mode, the AECB plans, trains, and exercises to maintain its emergency preparedness. In this mode, the AECB also responds to events which do not warrant activation of the emergency organization.

TABLE 16A.1

AECB EMERGENCY ORGANIZATION

Normal Operations Role	Emergency Operations Role
• AECB President • AECB Directors-General • selected AECB administrative staff	Executive Team
• AECB Director-General of Reactor Regulation	Emergency Director
• AECB Emergency Plan Coordinator • selected AECB administrative staff	Emergency Operations Centre (EOC) Support Team
• AECB Power Reactor Evaluation Division/ Power Reactor Operations Division, • AECB Materials Regulation Division, and • AECB Non-Proliferation, Safeguards and Security Division	On-site Team that monitors on-site events
• AECB Radiation and Environmental Protection Division, and • AECB site officers	Off-site Team that monitors off-site events
• AECB Communications Division	Public Information Team
• AECB Directorate of Corporate Services	Headquarters Logistics Team

- In standby mode, the AECB alerts responders and monitors the status of events which may require an emergency response at some stage.
- The AECB Emergency Response Plan enters the activated mode of operations when the AECB decides that an emergency response is necessary, and activates preparations for such a response.
- The recovery mode follows the activated mode, and consists of activities to restore a non emergency state, such as the standby or normal modes.

Within the context of the AECB emergency response plan, a nuclear emergency is any abnormal situation associated with a radiological activity, or an AECB-licensed activity or facility, that could require prompt action beyond normal procedures in order to limit damage to persons, property, or the environment. These nuclear emergencies could be off-site or on-site emergencies.

For example, a nuclear emergency could be created by events related to:

- the release, or potential release, of radioactive contaminants from a Canadian or foreign nuclear power plant, or any other AECB-licensed facility or activity;
- the release, or potential release, of any substance prescribed in the Atomic Energy Control Regulations;
- the loss, theft, discovery, or transport of radioactive material within or outside of Canada.

The nature of the above involvement could range from exchanging ideas and information to coordinating plans, attending training programs, participating in exercises, and responding to actual emergencies. The AECB Policy Statement on Emergency Response provides corporate guidelines for employee involvement. Emergency procedures set out the roles and responsibilities of various participants.

AECB membership in the AECB emergency organization is defined in the Plan, and depends upon the nature of the emergency. AECB staff responsibilities in the event of a nuclear emergency parallel their responsibilities during routine AECB operations. During a nuclear emergency, any member of the emergency organization, or the AECB, could be called upon to perform special tasks. AECB employees who are members of the AECB emergency organization learn and contribute through in-house planning and training sessions, and emergency exercises and drills. The AECB Emergency Response Plan is available at all times to all employees.

In the event of an emergency at a nuclear power plant, the AECB emergency organization would be formed from members of its normal organization as shown in Table 16A.1.

3.2 AECB Participation in Tests and Drills

In its normal mode of operation the AECB conducts routine tests of its emergency response arrangements and capabilities. These tests may be conducted independently or in parallel with other agencies. For example, the AECB participates in CANATEX and INEX exercises, United States Nuclear Regulatory Commission exercises, and the drills of Canadian nuclear power plants.

During the 1996-97 operating year, the AECB Duty Officer received calls concerning 165 separate occurrences:

- 53 for actual or potential incidents
- 23 for simulated incidents (drills and exercises)
- 25 for administrative assistance
- 64 for non-emergency items

During this period, AECB staff participated in one AECB exclusive exercise, one international exercise sponsored by the NEA, 23 checks of the AECB Duty Officer’s communications system, and several emergency drills at each of the seven nuclear power plant installations in Canada.