Government of Canada / Gouvernement du Canada

Canadian National Report for the Convention on Nuclear Safety

Second Review Meeting

Canada

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Catalogue number CC172-18/200E
ISBN 0-662-31059-4

Published by the Canadian Nuclear Safety Commission
CNSC Catalogue number INFO-0723

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October 2001

This report is produced by the Canadian Nuclear Safety Commission on behalf of Canada. Contributions to the report were made by Atomic Energy of Canada Limited, Ontario Power Generation, Bruce Power, New Brunswick Power, Hydro-Québec, the CANDU Owners Group, Health Canada, Natural Resources Canada, the Department of Foreign Affairs and International Trade, and the Emergency Response Organizations of the provinces of New Brunswick, Ontario and Quebec.

Canadian National Report
for
the Convention on
Nuclear Safety

The Canadian National Report is prepared in fulfilment of Canada's obligation pursuant to Article 5 of the Convention on Nuclear Safety coordinated by the International Atomic Energy Agency (IAEA). The report demonstrates how Canada has implemented its obligations under the Convention. This report has been prepared for the Second Review Meeting on the Convention on Nuclear Safety and is self-containing so that the reader does not have to refer to Canada's First Report to the Convention on Nuclear Safety.

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.

TABLE OF CONTENTS

INTRODUCTION

1. Canada's Nuclear Policy and the Related Government Structure 1

2. National Nuclear Programs Pertaining to Nuclear Power Stations 4
   2.1 AECL and its Role in National Nuclear Programs 4
   2.2 Research and Development 4

3. Basic Characteristics of the CANDU Reactor 5
   3.1 Brief Description of the CANDU Reactor 5
   3.2 CANDU Inherent Safety Features 6

4. A Survey of the Main Themes and Main Safety Issues Contained in the Report 7
   4.1 Summary of the Main Issues Discussed in Canada's Report to the First Convention on Nuclear Safety 7
   4.2 Main Themes in this Report 8
   4.3 The Main Safety Issues in this Report 8

5. Challenges to the Canadian Nuclear Program 9

ARTICLE 6 Existing Nuclear Power Stations

6.1 Canadian Philosophy and Approach to Safety of Nuclear Power Stations 11
6.2 List of Existing Nuclear Power Stations 16
6.3 Safety Assessments Performed and Major Results 17
6.4 Generic Safety Issues 17
6.5 OPG and NB Power Performance Improvement Programs 20
6.6 Corrective Actions and Monitoring Programs to Maintain and Improve Safety in Canadian Nuclear Power Plants 26
6.7 Canadian Position for Continued Operation of Nuclear Power Stations 26

ARTICLE 7 Legislative and Regulatory Framework

7.1 A Comprehensive Description of the Canadian Legislative and Regulatory Framework 27
7.2 A Summary of the Laws, Regulations and Requirements Governing the Safety of Nuclear Power Stations in Canada 30
   7.2.1 Regulations Issued under the Nuclear Safety and Control Act 30
   7.2.2 Nuclear Liability Act 31
   7.2.3 Canadian Environmental Assessment Act 32
   7.2.4 Nuclear Energy Act 32
   7.2.5 Nuclear Fuel Waste Act 32
   7.2.6 Regulatory Documents 33
7.3 A Description of the Licensing System for Nuclear Power Stations in Canada 35
   7.3.1 Operating Licence 37
7.4 A Description of the System of Regulatory Inspection and Assessment of Nuclear Power Stations to Ascertain Compliance with Applicable Regulations and Licences 39
   7.4.1 Licence Renewal 41
   7.4.2 Performance Indicators 42

7.5 A Description of the Process of Enforcement of Regulations and Conditions of Licences Used in Canada ... 43

ARTICLE 8 Regulatory Body
8.1 Position of the CNSC in the Government Structure ... 45

ARTICLE 9 Responsibility of the Licence Holder
9.1 Description of the Main Responsibilities and Activities of the Licence Holder Related to Safety Enhancement ... 47
9.1.1 Main Responsibilities of the Licence Holder Related to Safety Enhancement ... 47
9.1.2 Main Activities of the Licence Holder Related to Safety Enhancement ... 49
9.2. Description of the Mechanism by Which the CNSC Ensures that the Licence Holder Meets its Primary Responsibility for Safety ... 49
9.3 The CNSC Compliance Program ... 50

ARTICLE 10 Priority to Safety
10.1 Principles Emphasizing the Overriding Priority of Safety, and Their Implementation ... 53
10.1.1 Principles Directly Related to Safety ... 53
10.1.2 Design Safety Principles ... 53
10.1.3 Operational Safety Principles ... 53
10.1.4 Regulatory Control - Safety Principles ... 55
10.2 Safety Procedures ... 55
10.2.1 Safety Procedures at the Utilities ... 55
10.2.2 Safety Procedures at the Designer (AECL) ... 56

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 ... 57
11.2 The Financing of Safety Improvements Made to the Nuclear Power Station During its Operating Life ... 58
11.3 Provisions for Financial and Human Resources for Decommissioning the Nuclear Power Station and Radioactive Waste Management ... 58
11.4 Impact of Electricity Market Deregulation and Privatization in Canada ... 59
11.5 The Rules, Regulations and Resource Arrangements Concerning the Qualification, Training and Retraining of Personnel, Including Simulator Training for all Safety-related Activities for Each Nuclear Power Station ... 60
11.6 Capability Maintenance ... 62

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 ... 65
12.1.1 Operations Activities ... 65
12.1.2 Design Activities ... 66

12.2 Managerial and Organizational Issues 68
12.2.1 The Primary Responsibility for Human Performance of Each Individual 68
12.2.2 First Line Managers and Their Responsibilities in Human Performance Issues 68
12.2.3 Management's Roles and Responsibilities 69
12.2.4 Non-line Organizations Provide Independent Oversight of Human Performance 69
12.3 The Role of the Regulatory Body and the Operator 70
12.3.1 The Role of the Regulatory Body 70
12.3.2 Recent Human Factors Activities at CNSC 70

ARTICLE 13 Quality Assurance
13.1 Quality Assurance Policies 73
13.2 Life-Cycle Application of QA Programs 74
13.3 Methods Used for Implementation and Assessment of QA Programs 75

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 77
14.2 Licensee and Regulatory Control Activities Related to the Assessment and Verification of Safety 78
14.3 Safety Verification Programs 79
14.3.1 Maintenance Programs 79
14.3.2 Reliability Programs 80
14.3.3 Evaluation and Management of Ageing 82
14.3.4 Integrity of Pressure Retaining Components (Pressure Tubes, Feeder Pipes and Steam Generators) 83
14.3.5 Fire Protection 85
14.3.6 Environmental Qualification 86

14.4 Plant Return-to-Service and Plant Refurbishment Programs 87
14.4.1 Return to Service of the Pickering "A" Reactors: Licensee and Regulatory Activities 87
14.4.2 Refurbishment of the Point Lepreau and Gentilly-2 Reactors and Return to Service of the Bruce "A" Reactors 90

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 91
15.2 Dose Limits 91
15.3 Regulatory Control Activities and Radiation Protection 92
15.4 Environmental Radiological Surveillance 93

ARTICLE 16 Emergency Preparedness
16.1 A General Description of Laws, Regulations and Requirements for On-site and Off-site Emergency Preparedness 95

16.1.1 Overview of the Federal Nuclear Emergency Plan in Relation to Emergency Preparedness Measures ... 96
16.1.2 Dealing with Emergencies under the Federal Nuclear Emergency Plan and under Provincial Nuclear Emergency Plans ... 96
16.2 The Implementation of Emergency Preparedness Measures, Including the Role of the Regulatory Body and Other Entities ... 98
16.2.1 Measures for Informing the Public During a National Nuclear Emergency ... 98
16.2.2 Provincial Emergency Plans that Cover Nuclear Power Station Installations ... 99
16.2.3 Province of Ontario ... 99
16.2.4 Province of Quebec ... 100
16.2.5 Province of New Brunswick ... 101
16.2.6 Ontario Power Generation Nuclear Emergency Plan ... 102
16.2.7 Gentilly-2 NGS Nuclear Emergency Plan ... 103
16.2.8 Point Lepreau Nuclear Emergency Plan ... 104
16.2.9 Role of the Regulatory Body ... 105
16.3 International Arrangements, Including Those with Neighbouring Countries ... 107
16.4 Training and Exercises ... 107
16.4.1 Summary of the CANATEX 3/INEX 2 Exercise ... 107

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 ... 109

ARTICLE 18 Design and Construction

18.1 A Description of the Licensing Process, Including a Summary of the National Laws, Regulations and Requirements Relating to the Design and Construction of Nuclear Power Stations ... 111

ARTICLE 19 Operation

19.1 A Description of the Licensing Process, Including a Summary of the National Laws, Regulations and Requirements Related to the Operation of Nuclear Power Stations ... 113
19.2 A Description of the Steps Canada Has Taken in Implementing the Following Obligations Under Article 19 of the Convention ... 113
19.2.1 Initial Authorization to Operate a Nuclear Power Station ... 113
19.2.2 Operational Limits and Conditions ... 114
19.2.3 Operation, Maintenance, Inspection and Testing of Nuclear Power Stations ... 116
19.2.4 Establishing Response Procedures ... 117
19.2.5 Necessary Engineering and Technical Support in all Safety-related Fields ... 118
19.2.6 Reporting Incidents Significant to Safety ... 118
19.2.7 Programs to Collect and Analyse Information on Operating Experience ... 119
19.2.8 Radioactive Waste Management ... 121

ANNEXES

ANNEX 1.1	Research and Development Programs in Canada	125
ANNEX 1.2	A Brief Description of the CANDU Reactor	133
ANNEX 6.1	CNSC Generic Action Items	139
ANNEX 6.2	Systems Included in the Configuration Management Closure Project	159
ANNEX 6.3	Summary of Major Design and Operational Changes Resulting from Canadian Nuclear Safety Commission Actions	161
ANNEX 7.1	Description of the Nuclear Safety and Control Regulations	167
ANNEX 7.2	Descriptions of CNSC Regulatory Documents	173
ANNEX 7.3	Site Acceptance, Construction Approval and Commissioning of Power Reactors	177
ANNEX 7.4	List of Programs Required to Support a Nuclear Power Reactor Operating Licence Application	181
ANNEX 7.5	Sample Power Reactor Operating Licence	183
ANNEX 8.1	Canadian Nuclear Safety Commission (CNSC) Staff Oganization	205
ANNEX 9.1	Activities of the Licence Holder Related to Safety Enhancement	213
ANNEX 14.1	Key Issues for Return of the Pickering "A" Reactors to Service	217
ANNEX 14.2	Required Improvements and Modifications for Restarting the Pickering "A" Reactors	227
ANNEX 15.1	Dose to Personnel in Canadian Nuclear Powert Plants	231
ANNEX 15.2	Radiological Emmissions from Canadian Nuclear Power Plants	233
ANNEX 17.1	Siting	237
ANNEX 18.1	Design and Construction	241

LIST OF ATTACHMENTS

(provided under a separate cover)

1 Nuclear Safety and Control Act
2 General Nuclear Safety and Control Regulations
Radiation Protection Regulations
Class I Nuclear Facilities Regulations
Class II Nuclear Facilities and Prescribed Equipment Regulations
Uranium Mines and Mills Regulations
Nuclear Substances and Radiation Devices Regulations
Packaging and Transport of Nuclear Substances Regulations
Nuclear Security Regulations
Nuclear Non-proliferation Import and Export Control Regulations
Canadian Nuclear Safety Commission Rules of Procedure
Canadian Nuclear Safety Commission By-laws
3 Regulatory Policy P-119, Policy on Human Factors (2000)
4 Regulatory Policy P-211, Compliance (2001)
5 Regulatory Policy P-223, Protection of the Environment (2001)
6 Regulatory Policy P-242, Considering Cost-benefit Information (2000)
7 Regulatory Guide G-129, Guidelines on How to Meet the Requirement to Keep All Exposures As Low As Reasonably Achievable (1997)
8 Regulatory Guide G-149, Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors (2000)
9 Regulatory Guide G-206, Financial Guarantees for the Decommissioning of Licensed Activities (2000)
10 Regulatory Guide G-219, Decommissioning Planning for Licensed Activities (2000)
11 Regulatory Guide G-228, Developing and Using Action Levels (2001)

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
ALARA	As Low As Reasonably Achievable
CANATEX	Canadian National Exercises
CANDU	Canadian Deuterium Uranium
CEAA	Canadian Environmental Assessment Act
CMD	Commission Member Document
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
CSA	Canadian Standards Association
DRL	Derived Release Limits
ECCS	Emergency Core Cooling System
EMO	Emergency Measures Organization
EQ	Environmental Qualification
FNEP	Federal Nuclear Emergency Plan
GAI	Generic Action Item
HFEPP	Human Factors Engineering Program Plan
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
MCR	Main Control Room
MSV	millisieverts
NAOP	Nuclear Asset Optimization Plan
NB POWER	New Brunswick Power
NEA	Nuclear Energy Agency
NOC	Nuclear Oversight Committee
NPAG	Nuclear Performance Advisory Group
NRCAN	Natural Resources Canada
NSCA	Nuclear Safety and Control Act
OECD	Organization for Economic Cooperation and Development

OPG	Ontario Power Generation
OPEX	Operating Experience
OP&P	Operating Policies and Principles
PIP	Performance Improvement Program
PNEP	Provincial Nuclear Emergency Plan
PSA	Probabilistic Safety Analysis
PSAR	Preliminary Safety Analysis Report
QA	Quality Assurance
ROPT	Regional Overpower Protection Trip
SDS1	Shutdown System One
SDS2	Shutdown System Two
SDSE	Shutdown System Enhancement
SER	Significant Event Report
SOE	Safe Operating Envelope
SSFI	Safety System Functional Inspections
UEPS	Unusual Event Processing System
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 high priority to the safety and protection of persons 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 Nuclear Safety and Control (NSC) Act, which was promulgated in May 2000, and the Nuclear Liability Act, which was promulgated in October 1996, are the centrepieces of Canada's legislative and regulatory framework for the nuclear industry (refer to Article 7). Canada has updated its nuclear regulatory requirements to make sure that the most current standards and practices are in place.

The major federal government organizations involved in the Canadian nuclear program are:

- The Canadian Nuclear Safety Commission (CNSC): created through the NSC Act, responsible for the legislative and regulatory requirements for the safety of the nuclear industry. Through the CNSC, the federal government regulates the development, production and use of nuclear energy in Canada and ensures that Canada's international commitments on the peaceful use of nuclear energy are respected. The term "Commission" is used in this document to refer to the seven Commission members appointed by the Governor-in-Council. The initials "CNSC" are used to refer to the commission and its staff;

- Atomic Energy of Canada Limited (AECL): responsible for the design, marketing, construction and servicing of CANDU power reactors;

- Natural Resources Canada (NRCan): the Nuclear Energy Division and the Uranium and Radioactive Waste Division provide leadership in the development and implementation of Canadian government policy on uranium, nuclear energy and radioactive waste management;

- Department of Foreign Affairs and International Trade (DFAIT): this Agency is charged with promoting, bilaterally and multilaterally, nuclear cooperation and safety, and the implementation in Canada and abroad of key non-proliferation and disarmament agreements. Implementation of these Agreements requires, among other things, that Canadian domestic law be consistent with Canada's responsibilities under the Agreements. It also requires the capacity to ensure effective monitoring to verify that treaty obligations and commitments are being honoured. The Agency is also responsible for the implementation of *The Chemical Weapons Convention (CWC) and The Comprehensive Nuclear-Test-Ban Treaty (CTBT)*. Canada signed the *CWC* and the *CTBT* respectively in 1993 and 1996 and ratified them respectively in 1995 and 1998; and

- Health Canada: the Radiation Protection Bureau (RPB) of Health Canada investigates and manages the risks from natural and artificial sources of radiation. This is accomplished through the Federal Nuclear Emergency Plan (FNEP), the National Radioactivity Monitoring Network, by developing guidelines for exposure to radioactivity in water, food and air following a nuclear emergency, and by providing a full range of dosimetry services to workers through the National Dosimetry Services, the national Dose Registry, the National Calibration Reference Centre and biological dosimetry services.

The federal government has funded nuclear research and supported the development and the use of nuclear energy and related applications for several decades. Federal government funds for research are approximately $100 million annually for research and development activities related to CANDU technology. Four utilities (Ontario Power Generation (OPG), Bruce Power, Hydro-Québec and New Brunswick Power (NB Power)) also contribute funding for the program under auspices of the CANDU Owners Group (COG).

Canadians have benefited in many ways from this investment:

- Nuclear energy supplies on average about 15% of Canada's electricity without creating emissions that contribute to air pollution, acid rain and global warming.

- 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 30,000 highly skilled jobs. These jobs are concentrated in the uranium industry, three provincial nuclear utilities, one privately-owned nuclear utility, 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.

NUCLEAR POWER STATIONS IN CANADA

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

- Pickering "A" units 1 through 4: A business decision was made by the licensee in 1997 to shutdown the units, at least temporarily, to allow redeployment of resources to work on performance improvements at the remaining units of Pickering "B", Bruce "B" and Darlington. They were placed in the guaranteed shutdown state at the end of 1997 because they did not meet the regulatory requirement for an enhanced shutdown system. OPG then made the decision to place these units in a lay-up state. OPG has undertaken an extensive program to refurbish the reactors and return them to service between 2002 and 2004. The return of the Pickering "A" units to service is discussed further in Article 14, and Annexes 14.1 and 14.2.

- Bruce "A" units 1, 3 and 4 have been defuelled and placed in a laid up state during 1998 as a result of a business decision taken by the licensee in 1997. These reactors, along with the Bruce "B" units have been leased to Bruce Power by OPG. Bruce Power is a company owned by British Energy, with minority interest held by Cameco, the Ontario Power Workers Union and the Society of Energy Professionals. Bruce Power has initiated programs to return the Bruce "A" units 3 and 4 to service. Bruce Unit 2 will remain shut-down as the steam generators were damaged by lead contamination.

  The lease of the Bruce "A" and "B" units to Bruce Power came into effect on May 12, 2001. The status of programs and discussion of operations at the Bruce "A" and "B" units, as presented in this report, pertains to operation of these units by OPG. Bruce Power will be referred to as a licensee throughout this report; however, at the time of writing of this report, Bruce Power had just assumed responsibility for operation of the Bruce "A" and "B" units. It is important to note that OPG is the owner of these units, but Bruce Power is the licensee.

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 1.1.

TABLE 1.1 List and Status of Nuclear Power Stations in Canada

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

NOTES:

1. Pressurized Heavy Water Reactor
2. 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. Work is ongoing by OPG to refurbish these stations to meet current regulatory requirements and to re-start these reactors over a period from 2002 to 2004.
3. Placed in a lay up state as a result of a business decision taken by the licensee in 1997. Bruce Power has applied to the CNSC to return Bruce Units 3 and 4 to service.
4. Heavy Water/Light Water Reactor

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 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, AECL:

- 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; and
- offers radioactive waste management products and services.

AECL also plays a leadership role in the 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.

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 four Canadian nuclear utilities and Atomic Energy of Canada Limited (AECL). AECL performs in-house research and development that is used in support of product development for advances in reactor design, and in support of products and services for CANDU and light water reactors. An overview description of the Research and Development programs underway in Canada is provided in Annex 1.1.

The Research and Development programs funded by COG are Safety and Licensing, Fuel and Fuel Channels, Chemistry, Materials and Components and Health, Safety and the Environment.
COG Operations is a separate entity that administers and monitors the technical program. Technical Committees manage the research work in different technical areas. COG plays a key role in co-ordinating the exchange of operational experience information between CANDU utilities (see Article 19 for more details). The research work is mainly done at the laboratories of:

- AECL in Chalk River and Mississauga, Ontario; and Pinawa, Manitoba;
- Kinetrics in Toronto, Ontario;
- Stern Laboratories in Hamilton, Ontario; and
- subcontractors, including some Canadian universities.

The COG program has decreased from $92,700,000 in 1997/98 to approximately $30,000,000 in 2000/2001. As will be discussed later in the Introduction and in Article 11, overall reductions in funding have resulted from the rapidly changing environment and have presented challenges to the Canadian nuclear industry.

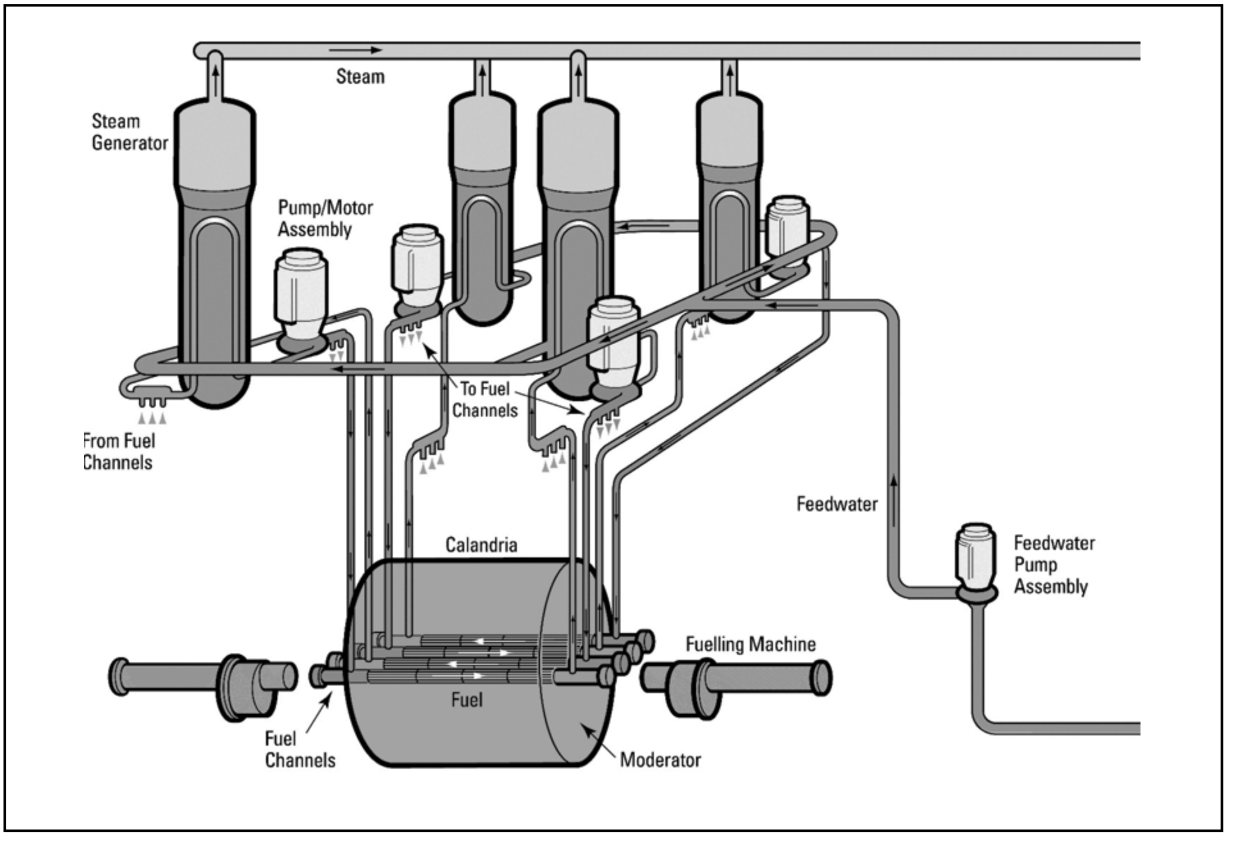
3. Basic Characteristics of the CANDU Reactor

3.1 A Brief Description of the CANDU Reactor

A CANDU reactor utilizes controlled fission in the reactor core as a heat source to supply steam and generate electrical power. A CANDU reactor 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 a heavy water (D_2O) moderator at low pressure and low temperature. 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 (HTS) is shown schematically in Figure 1.1. Pressurized heavy water (D_2O) coolant is circulated through the fuel channels and steam generators in a closed circuit using heat transport system pumps. The fission heat produced in the fuel is transferred to heavy water coolant flowing through the fuel channels. The coolant carries the heat to steam generators, where the heat is transferred to the light water side of the steam generator to produce steam. The steam is used to drive the turbine generator to produce electricity.

FIGURE 1.1: CANDU Reactor Heat Transport System



The reactor, fuel and fuel changing systems are described in more detail in Annex 1.2.

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 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 ensure shutdown.

- Due to 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 operating life of the reactor.

- The pressure tube concept aids 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 practical. This helps to keep the heat transport system essentially free from fission products. A clean heat transport system allows maintenance work to be done with minimal 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 relatively 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 milli-k (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 would be 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.

- The 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 Emergency Core Cooling System (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, modern CANDU includes engineered systems to enhance reactor safety. They include the following systems:

- Two redundant, independent, diverse, separated shutdown systems, with the capability for testing during operation to demonstrate an unavailability of less than 10^{-3}. They share no devices with the control system nor with each other. This double 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.

- Safety systems are placed in two groups, Group 1 and 2, to ensure separation of safety systems, to provide two independent means of achieving the same safety function, and provides defence against common cause events. In addition to the standby generators in the Group 1 safety systems, there are also independent, separately-located emergency generators in Group 2 so that a loss of all AC power occurs at a very 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.

System performance is evaluated in Safety Report and/or Probabilistic Safety Assessment documents.

4. A Survey of the Main Themes and Main Safety Issues Contained in the Report

4.1 Summary of the Main Issues Discussed in Canada’s Report to the First Convention on Nuclear Safety

The main issues discussed in the first report to the Convention on Nuclear Safety 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.

- Canadian reactor safety philosophy and reactor safety requirements, applied through the regulatory process, ensuring that the risk to the workers, the public and the environment associated with nuclear power station operation is acceptably low.

- 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.

- Responsibility for safety rests with the licensees. Nuclear power stations are licensed to provincial crown corporations which report to a government body (as discussed, as of May 2001, the Bruce “A” and “B” stations are operated by privately-owned Bruce Power.) All licensees 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 and international organizations (see Article 16).

The main issues raised in the review of Canada's report to the First Review Meeting of the Convention on Nuclear Safety were:

- Enforcement powers of the Canadian Nuclear Safety Commission (CNSC), the regulatory framework in Canada, and the role and decision making power of the Commission, President and staff of the CNSC.
- Enforcement strategy to deal with licensees with deteriorating performance record which has not yet reached an unacceptable safety level.
- Impact of competition/privatization on nuclear safety and the nuclear option in Ontario.
- Role of the CNSC in Ontario Power Generation (OPG) plant lay-ups and current role of the CNSC in the asset recovery program.
- Advantages and disadvantages from a safety perspective of a two-year licence period.
- Uncertainty of void reactivity calculations, severe accidents, pressure tube life, pressure tube replacement, and feeder thinning.
- Use of performance indicators by operating organizations and regulators.

4.2 Main Themes in this Report

Particular attention has been given to the following subjects in this report:

- the Nuclear Safety Control Act and accompanying Regulations;
- progress made by the utilities in their performance improvement programs and the operating status of its stations;
- return to service of the Pickering "A" reactors;
- performance indicators developed by the CNSC;
- capability maintenance (supporting research and development, and human resources);
- impact of privatization;
- the improved licensing framework; and
- updates on reactor safety issues.

4.3 The Main Safety Issues in this Report

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

- The licensees of Canadian nuclear power plants have put performance improvements plans in-place over the last few years in response to declining standards of operation and maintenance that were evident in the mid-1990's. Licensee performance has improved over the last few years; however, the rate of improvement has been slower than anticipated. This is more a reflection of the nature of nuclear power technology, rather than failing to move forward. Significant progress is still required in the areas of quality assurance and configuration management. However, industry strengths remain in the areas of overall station operation, technical surveillance, outage management, non-radiological health and safety, environmental protection and radiation protection.

- A limited number of outstanding safety issues remain to be resolved (see Articles 6.3 and 6.4, and Annexes 6.1 and 6.3). In some cases, these issues limit the power output of the nuclear power stations.

5. Challenges to the Canadian Nuclear Program

There have been significant changes and challenges in the Canadian nuclear program since 1997. These include:

- introduction of new legislative requirements (the Nuclear Safety and Control Act and accompanying regulations);
- break-up of the largest Canadian utility in a move to privatization of the electricity market by the Province of Ontario;
- issues regarding staffing, knowledge retention and the supporting research and development infrastructure; and
- ageing plants and utilities moving towards life extension.

A significant factor in Canada is the coming into force of the Nuclear Safety and Control Act in May 2000. This modern legislation replaces a 1946 Atomic Energy Control Act. The Nuclear Safety and Control Act, and its associated regulations, are much more specific about the powers and duties of the regulator than the previous law. Among other provisions, the new Act gives the Commission much clearer responsibilities in the area of the environment and gives powers to require financial assurances. Clause 24(4) of the Nuclear Safety and Control Act forbids the Commission from granting a licence unless it is satisfied that the applicant is competent and will make adequate provision for health and safety of persons, the environment, national security and Canada’s international obligations. In response to the requirements of the Nuclear Safety and Control Act, a new, integrated approach to reactor licensing has been developed.

The electricity industry in the Province of Ontario is moving towards privatization. A major step involves the lease of the Bruce “A” and “B” reactors to Bruce Power, a consortium with investment from British Energy, Cameco, the Society of Energy Professionals and the Power Workers Union. The changing environment is challenging resource management throughout the industry, from the human resource, research and development support, and infrastructure perspectives.

In parallel with these developments, the nuclear power industry in Canada, and the Canadian Nuclear Safety Commission, are faced with similar challenges to those of many other nations with respect to “succession planning”. This refers to the challenges of replacing ageing equipment, people, knowledge, and supporting infrastructure, of maintaining a capability to conduct research, and to respond promptly to unexpected problems.

The CNSC has requested the industry to evaluate the status of research and development in Canada, where there had been a decline in funding over the last 5 years. The CANDU Owners Group (a company that directs and manages cost-shared research and is funded by AECL, OPG, NB Power, Hydro-Quebec and Bruce Power) produced a report that evaluated the issue and identified a number of actions to implement before the support capability reaches a critical level. In concert with the assessment of the research support capability, the CNSC has required that the nuclear utilities develop an agreed definition of the term ‘design authority’ for the knowledgable entity that holds the design basis information, the design status and is able to competently approve design changes for each reactor. The utilities have been requested to propose a definition of the minimum competence that the design authority shall have.

As is the case in many countries, the oldest power reactors in Canada are approaching the end of their original design life. Changing circumstances are causing licensees to seriously consider life extension for these plants. Ontario Power Generation and New Brunswick Power have undertaken major performance improvement programs. In addition, Ontario Power Generation is moving towards returning the Pickering “A” units to service over the time frame from 2002 to 2004. New Brunswick Power and Hydro-Quebec are currently evaluating refurbishment of the Point Lepreau and Gentilly-2 stations, while Bruce Power intends to return Bruce “A” Units 3 and 4 to service.

The applicability of new standards to older plants, and the requirements for environmental assessments associated with refurbishment and return-to-service of existing facilities are significant challenges for the industry. The industry is following the concept of Periodic Safety Review and licensees are being requested to essentially follow the IAEA standard in seeking to justify further operation. Further discussion of the Pickering “A” return-to-service project and the possible refurbishment of Point Lepreau is provided in Article 14.

In addition to the above, the current, and rapidly changing environment, has posed specific challenges to the utilities and the regulator. Areas of concern are:

- utilization of staff (e.g, skills, knowledge, experience, stress, fatigue);
- reduced safety margins (e.g, ageing equipment, power upgrades, increased fuel burn-up);
- impact on plant aging and equipment reliability (e.g., reduced funds for capital investment, reduced scope in maintenance programs); and
- changing requirements for the regulator (e.g., maintaining technical competence and identifying/ developing new competencies required, self-assessment of regulatory effectiveness, pressure to reduce the regulatory burden, adequacy of legislation, more prescriptive regulation, positions regarding closure of outstanding safety issues, the possibility of diminished dissemination of proprietary and/or commercially sensitive information from licensee to the regulator).

ARTICLE 6
Existing Nuclear Power Stations

6.1 Canadian Philosophy and Approach to Safety of Nuclear Power Stations

In Canada, the primary responsibility for safety rests with the licence holder of the nuclear power station. It is the licensee's responsibility to operate the station safely, and to demonstrate to the satisfaction of the regulator that the nuclear power station will continue to 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 CNSC as the regulator has produced only 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 CNSC. When accepted by the CNSC, 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). The intention of this regulatory approach 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.

To supplement the general requirements, the CNSC has issued regulatory policy statements on the requirements for the special safety systems. The regulatory laws and requirements governing the safety of nuclear power stations are described in detail in Article 7.2. Other regulatory documents (guides, policies, standards) are also described in Article 7.2.

The essential principles underlying the Canadian philosophy and approach to nuclear power station safety evolved from the recognition that even well-designed and well-built systems may 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 CNSC 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 their safety function:

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

Subsequently, the protective shutdown systems (SDS1 and SDS2), the Emergency Core Cooling System (ECCS) and containment system were combined into a single category of "special safety systems".

The criteria in the Siting Guide 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). Table 6.1 lists the Power Reactor Safety Criteria and Principles.

TABLE 6.1 Power Reactor Safety Criteriaand Principles[1]

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 releases 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.

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 (a dual failure), the predicted dose to any individual shall not exceed: 250 mSv whole body, 2.5 Sv thyroid.

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[2].
(ii) conversion factors as given by Beattie[3].

NOTES:
1. from D. G. Hurst, F. C. Boyd, "Reactor Licensing and Safety Requirements", AECB-1059, 1972.
2. Bryant, P.M., UKAEA report AHSB(RP)R42, 1964.
3. Beattie, J.R., UKAEA report AHSB(S)R64, 1963.

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. The population reference dose limit for the dual failure scenario was established on the basis that it would have a relatively small effect. On the basis of data available at the time, it was estimated that exposure to the dose incurred during a dual failure might lead to about a 0.1% increase in the lifetime incidence of cancer.

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 systems 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 had to 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 defined 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. Containment must be designed to limit public dose and release of radionuclides to the environment. Similarly, a LOCA with impaired containment sets the effectiveness required of the ECCS.

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 five elements to the defence-in-depth approach:

- prevention of abnormal operation and failures;
- control of abnormal operation and detection of failures;
- control of accidents within the design basis;
- control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents; and
- mitigation of radiological consequences of significant releases of radioactive materials.

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); and
- 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 ensuring 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;
- ensuring plant staff are appropriately trained;
- employing appropriate quality control and quality assurance methods in all phases of design, construction, and operation; and
- monitoring events in other similar facilities to anticipate problems before they occur.

The defence-in-depth approach also requires that provisions and procedures are in place to mitigate the consequences of accidents. These include measures to prevent fuel failure following a serious process failure as well as provisions to contain radioactive materials, in the event that fuel failures occur. 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; and
  - 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; and
  - 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.

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.

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, and 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, or evacuation.

Although the single/dual failure approach adequately defined the required effectiveness of the special safety systems, some problems in coverage for accident scenarios 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; and
- 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 CNSC and the utilities, but also by independent advisory groups set up by the CNSC. The concerns were addressed as follows:

- a more probabilistic approach, the “safety design matrix”, was introduced to assess support system and human failures;
- the safety design matrix approach was also used to assess post-accident reliability of systems; and
- 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 1975, AECL proposed an analytical technique called a safety design matrix to deal with matters of interdependency, post-accident operation and 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 inter-dependencies between systems and 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 (e. G. CANDU 9), a preliminary PSA has been used to define system reliability requirements and subsequently a detailed PSA has been used to confirm the adequacy of the design.

In June 1980, the Consultative Document "Requirements for the Safety Analysis of CANDU Nuclear Power Plants" (C-6), providing the requirements for safety analysis of CANDU reactors, was issued for comment. This document addressed deficiencies in the basic single/dual failure safety analysis requirements, and reflected Canadian experience in applying the single/dual failure analysis approach. In 1983, agreement was reached to apply this document, 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; and
- 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 CNSC 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. Consultative Document "Safety Analysis of CANDU Nuclear Power Plants" (C-006, Rev. 1, September 1999) supercedes consultative document C-6 and consolidates lessons learned from the trial application and comments received since then.

The requirements contained in "Safety Analysis of CANDU Nuclear Power Plants", (C-006, Rev. 1), represent an increase in both the scope and the rigour of design basis accidents, which operating power plants licensed after to the Darlington station would have to consider. For example, C-006, Rev. 1 will:

- identify approximately 200 potential initiating events considered to be pertinent to the safety of CANDU nuclear power station; and
- 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.

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.

To summarize, the Canadian approach ensures 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 CNSC, although the latter has, in turn, influenced the design.

6.2 List of Existing Nuclear Power Stations

A listing of the nuclear power stations and their operational status is provided in the Introduction (Table 1.1). 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, were explained in the Introduction and in Annex 1.2.

6.3 Safety Assessments Performed and Major Results

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.

A summary of these safety assessments performed following major national and international incidents and in response to operating experience are listed in Table 6.2.

6.4 Generic Safety Issues

Generic safety issues are issues that may affect more than one station. Several generic safety issues have been grouped by CNSC staff under the label Generic Action Items (GAIs). GAIs are used as a regulatory tool to define the scope of key safety issues, to identify outstanding technical issues and to specify requirements for resolution of the safety issue. The GAIs are also used to monitor the progress of licensees with regards to safety issues and to provide a basis for communication of licensee progress.

Many such GAIs have been raised over the years; some of them have been closed and others are still open. The following is a list of GAIs which are currently "open" for at least some of the Canadian stations:

GAI 88G02	"Hydrogen Behaviour in CANDU Nuclear Generating Stations"
GAI 90G02	"Core Cooling in the Absence of Forced Flow (CCAFF)"
GAI 91G01	"Post Accident Filter Effectiveness"
GAI 91G02	"Operation with a Flux Tilt"
GAI 94G02	"Impact of Fuel Bundle Condition on Reactor Safety"
GAI 95G01	"Molten Fuel / Moderator Interaction"
GAI 95G02	"Pressure Tube Failure with Consequential Loss of Moderator"
GAI 95G03	"Compliance with Bundle and Channel Power Limits"
GAI 95G04	"Positive Void Reactivity - Treatment in Large LOCA Analysis"
GAI 95G05	"Moderator Temperature Predictions"
GAI 98G01	"PHT Pump Operation Under Two-Phase Flow Conditions"
GAI 98G02	"Validation of Computer Programs Used in Safety Analysis of Power Reactors"
GAI 99G01	"Quality Assurance of Safety Analysis"
GAI 99G02	"Replacement of Reactor Physics Computer Codes Used in Safety Analysis of CANDU Reactors"
GAI 00G01	"Channel Voiding During a Large LOCA"

The following are examples of GAIs which have been satisfactorily dispositioned and hence "closed":

GAI 89G03	"Ontario Hydro's Pressure Tube Inspection Program"
GAI 89G05	"Use of Mercury-Wetted Relays in Safety Related Systems"
GAI 92G01	"Treatment of Human Factors in Ontario Hydro Reliability Analyses"
GAI 96G02	"Feeder Pipe Fitness for Service"

Descriptions of both open and closed GAI are given in Annex 6.1. GAIs have been used more as a regulatory tool to enforce safety enhancements in addition to highlighting generic safety concerns.

TABLE 6.2 Lessons Learned and Corrective Actions Resulting from Significant Incidents and Operating Experience: The Canadian Perspective

National and International Incidents	**Lessons Learned/ Corrective Actions**
National Research Experimental (NRX) loss-of-regulation and failure to shutdown accident (1952).	• a reactor should always have a fast shutdown capacity available and that this capacity should be independent of any control system; • the 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
Three Mile Island - small break loss of coolant accident (1979)	• 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
Chernobyl (1986)	• the importance of flux tilt as an initial operating state. The CNSC 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 • OPG was requested to re-examine 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 fast-acting shutdown system. They were licensed before the requirement for two independent shutdown systems) • the CNSC mandated the installation of a shutdown system enhancement in each of the Pickering “A” stations. This is a major part of the Pickering “A” return to service project (refer to Article 14.4 for more details)

TABLE 6.2 Lessons Learned - Continued

Operating Experience	Lessons Learned/ Corrective Actions
Pickering "A" loss of regulation (1971 to 1975)	• improvements in design, equipment and operational changes undertaken to improve the reliability of the reactor regulating system
Pickering "A" pressure tube rupture (1983)	• recognition of the importance of deuterium ingress into pressure tube material (hydriding) and of avoiding contact between calandria and pressure tubes • replacement of all pressure tubes at Pickering "A" station. A Zirconium alloy containing 2.5% Niobium is now used as pressure tube material in all CANDU reactors following this event • garter spring (spacers between the pressure and calandria tube) relocation programs implemented for all reactors (except Darlington) • a new garter spring design for Darlington • development of fitness-for-service requirements preclude operation of reactors under conditions where pressure tubes could form blisters • engineering programs at all stations to inspect pressure tubes and make sure that garter springs are properly positioned
Loss of regulation from low power at Bruce Unit 2 (1992)	• significant operational and procedural changes to the reactor regulating systems at all Bruce units
Darlington Shutdown System Software Re-design (1990 to 1998)	• a program of software redesign to improve documentation, modularity and software maintenance was introduced by OPG as the software was difficult to change and it was difficult to validate the software after changes
Pickering Unit 2 small loss of coolant accident (1994)	• design changes (improved bleed condenser overpressure relief system) and improved maintenance (replacement of existing primary heat transport liquid relief valve actuator diaphragms) • similar deficiencies in design and procedural changes to the primary heat transport overpressure relief systems at all CANDU reactors (except Pickering "B")
Gentilly-2 and Point Lepreau HTS flow reduction and its impact on regional overpower protection trip (ROPT) setpoints (1997 to 2001)	• Gentilly-2 and Point Lepreau self-imposed a reduction in their ROPT detector trip setpoints, which consequently led to a reduced reactor power • reduction in secondary side pressure and primary side steam generator cleaning implemented to maintain appropriate safety margins

6.5 OPG and NB Power Performance Improvement Programs

OPG INDEPENDENT INTEGRATED PERFORMANCE ASSESSMENT PROGRAM (IIPA):

In January 1997, OPG formed the Nuclear Performance Advisory Group (NPAG) to perform an Independent, Integrated Performance Assessment (IIPA) within OPG. The IIPA was conducted over a three month period, and it consisted of a series of detailed reviews and reports of OPG's operations at its Pickering, Bruce and Darlington nuclear generating stations as well as its head office groups. In addition, there were special reviews and reports prepared on selected safety systems which were called Safety System Functional Inspections.

The overall conclusion of these assessments and inspections was that all OPG's nuclear stations were being operated in a manner that met the defined regulations and acceptance standards for nuclear safety. However, the reports were also critical and identified a large number of shortcomings in its operation and maintenance programs against standard industry practice. The NPAG team ranked all of the OPG's operating stations as only "minimally acceptable" and stated that immediate attention was required in order to improve or even just to maintain their current performance.

The problems identified by the NPAG included poor managerial practices, corporate cultural problems, poor and outdated procedures, loss of configuration control and more. These problems were found to be "wide ranging and deep".

In recommending a remedy to the findings, the NPAG developed the Nuclear Asset Optimization Plan (NAOP) to recover the performance of OPG's nuclear program. In the opinion of the NPAG, OPG did not have the resources to both operate and improve the performance levels of OPG's 19 operating units. The NAOP plan called for the lay-up of the Pickering "A" and Bruce "A" nuclear generating stations. This lay-up would allow a redeployment of OPG's resources to work on performance improvements at the remaining twelve stations - Pickering "B", Bruce "B" and Darlington.

The specific performance improvement program is called the Integrated Improvement Program (IIP). Originally, the IIP consisted of 66 projects in 15 issue areas, and represented a total effort requiring almost $1.7 billion dollars and approximately 2,100 OPG staff over a five year period. OPG's commitment to complete the IIP was part of its basis for its continued operation. The completion date of the IIP is currently 2004.

The issue areas in the scope of the IIP are:Requisite Management, Operations, Maintenance, Material Condition/Capital Upgrades, Training, Engineering, Radiation Protection, Regulatory Affairs, Performance, Quality, Emergency Preparedness, Security, Organizational Effectiveness, Chemistry and Environmental.

From the original 66 IIP projects, the CNSC selected a subset of 44 IIP projects having regulatory interest and pertaining to reactors, workers, and public safety. Of the initial 66 projects, 37 remained open as of January 2001. Seven of the CNSC-monitored IIP projects were completed by OPG. Progress in these 37 projects are monitored by the CNSC (the IIP projects are listed in Table 6.3).

One of the first initiatives was to review the projects to determine if it would be appropriate to create new licence conditions which would require their completion. Subsequent review of fire protection, quality assurance and environmental qualification at all Canadian nuclear power plants led to the introduction of licence conditions for these three areas, in all power reactor licences.

During its initial start-up phases of the IIP in 1998, OPG encountered significant delays mainly due to lack of manpower. About a year and half into the improvement program, OPG undertook a full review of the IIP and adjusted it ("rebaselined") by merging related projects. In early 2000, OPG delivered a set of revised project execution plans and the CNSC staff reviewed them to ensure that there were no scope changes from the original IIP. OPG also identified a set of "core" projects which would receive increased attention as being key projects which required integration with each other and with station operations. The core projects were:

- Conduct of Operations/Plant Status Control;
- Conduct of Maintenance;
- Conduct of Engineering/Engineering Programs;
- Configuration Management Restoration;
- Preventative Maintenance Optimization;
- Safe Operating Envelope; and
- Environmental Qualification.

During the spring of 2000, OPG started the Bruce "B" Emergency Coolant Injection (ECI) Pilot Project. This was a project to develop a system based, integrated approach to the implementation of the core projects. The plan was to complete all the tasks from core projects on one system, with the result being a fully configured system. Efficiencies would be achieved by identifying where the same or similar tasks were required for the different deliverables and only completing that task once. Once the process was developed, the intention was that it would be applied to other systems at Bruce "B" and at the other sites.

TABLE 6.3 Ongoing IIP Projects

• Management of Hazardous Materials • Engineering Governing Documents • Systems Engineering and Engineering Programs • Probabilistic Risk Assessment • Analysis Code Verification and Validation • Safe Operating Envelope • Chlorine Reduction / Zebra Mussel Migration • Emergency Preparedness Program • Environmental Management System • Radioactive Emissions Reduction • Upgrade Environment Impact Monitoring • Conventional Fuel Oil Storage Tank • Contaminated Land Maintenance • Maintenance Management • Conduct of Maintenance • Backlog Reduction • Preventive Maintenance Optimization	• Fire Protection Upgrade Program • Environmental Qualification • Industrial Health and Safety Standards • Configuration Management Restoration • Operations Structures and Staffing • Conduct of Operations • Plant Status Control • Performance Assurance • Assessment and Audit • Corrective Action / Trending Project • Pressure Boundary Programs • Licensing Basis • Regulatory Commitment Reconciliation • Improve Contamination • Security Program • Access Control • Authorization Training • Non-Licensed Training Recovery Project • Building Managerial Leadership in OPG • Year 2000 Project

The ECI Pilot was an intense effort for OPG and, aside from the process developed, many deliverables were completed. The Bruce B ECI pilot project is 70 to 100% complete in most areas. The Bruce B ECI Pilot Project approach was a breakthrough for the IIP program. It clearly demonstrated what was required to complete the deliverables for a core set of projects on one system, and is regarded as the approach to integrate the key projects and bring them under the larger, full scope Configuration Management umbrella.

OPG is currently applying the system approach from the Bruce pilot project to other “critical/safety systems” following a framework plan entitled: the Configuration Management Closure Project (CMCP). The goal of the CMCP was to establish the baseline for configuration management on selected systems, on a system by system basis. The CMCP is implemented on the Safe Operating Envelope systems, with the exception of Fuel and Reactor Physics (Annex 6.2) lists the systems covered by the Configuration Management Closure Project).

OPG’s IIP has resulted in improvements in station performance. OPG publishes a number of performance indicators in monthly and annual report cards. Performance indicators such as the number of reportable events, the industrial safety accident rate, the capacity factor, the nuclear performance index and the radiation dose to the public provide a measure of overall station performance. The performance indicators for 1998, 1999 and 2000 are listed in Tables 6.4 through 6.8.Data in Tables 6.4 through 6.8 show, in general, reduction in dose to the public, a decrease in the industrial safety accident rate, and improvements in the capability factors and nuclear performance index. The decrease in reportable events is noted at Pickering and Bruce, while there was an increase at Darlington. The increase in reportable events can be linked to increased awareness of the requirements for reportable events by station staff, but does not imply that staff at other stations do not have a high level of awareness of reporting requirements.

Following the requirements of CNSC Regulatory Document, “Reporting Requirements for Operating Nuclear Power Facilities”, R-99, (1995), the licensee must report on reportable events. The number of reportable events at the Bruce, Darlington and Pickering sites from 1998 to 2000 are listed in Table 6.4. Requirements of R-99 are discussed further in Article 9.2.

TABLE 6.4 Reportable Events

	1998	1999	2000
Bruce	19	5	3
Darlington	28	15	35
Pickering	12	2	4
All OPG Sites	59	22	42

The industrial safety accident rate (Table 6.5) is the number of employee accidents per 200,000 hours worked that result in lost time, injuries that restrict work, or fatalities.

TABLE 6.5 Industrial Safety Accident Rate (#/2000,000 Work Hours)

	1998	1999	2000
Bruce	0.61	0.37	0.34
Darlington	0.5	0.5	0.21
Pickering	0.43	0.36	0.00
All OPG Sites	0.43	0.44	0.26

The capacity factor (Table 6.6) is the amount of electricity the stations are actually capable of producing per month as a percentage of their potential capacity - in other words, their capacity if all reactors and related systems were operating with no down-time.

TABLE 6.6 Capacity Factors (%)

	1998	1999	2000
Bruce	70.1	81.7	84.7
Darlington	78.3	83.5	87.0
Pickering	73.7	77.1	57.0
All OPG Sites	76.5	81.3	79.1

The nuclear performance index (Table 6.7), reported on a quarterly basis, is measured out of 100 and provides an overview of performance based on 11 key statistics that cover a number of areas, including safety and productivity. The index is used by the World Association of Nuclear Operators (WANO) to measure performance of nuclear plants worldwide.

TABLE 6.7 Nuclear Performance Index

	1998	1999	2000
Bruce	70.4	82.8	90.8
Darlington	61.6	79.9	87.2
Pickering	71.5	79.9	72.0
All OPG Sites	67.8	80.9	83.3

Radiation Exposure to the Public

The radiation exposure to the public (Table 6.8) is an estimate of the radiation dose people would receive if they lived just outside the station boundary, at their residences, 24 hours a day, drank local water and milk, and ate local fish and produce. The dose is measured in microsieverts (µSv).

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TABLE 6.8 Radiation Exposure to the Public (µSv)

	1998		1999		2000	
	Year End Actual	Year End Target	Year End Actual	Year End Target	Year End Actual	Year End Target
Bruce	2.2	10	2.39	10	3.3	8.0
Darlington	6.0	10	2.99	10	2.0	7.5
Pickering	7.0	10	12.6	20	5.1	16.0

As of January 2001, OPG transferred responsibility for completion of the IIP and CMCP projects to each of the four divisions (i. E., Nuclear Operation Station Support and the Pickering, Bruce and Darlington sites). Individual division-specific project expansion plans would be developed at each division and the divisions would have the responsibility to execute them following station-specific priorities. OPG has committed to the scope, schedule, funding and continued reporting to the CNSC. The transfer of the IIP programs to the divisions was made with care, with each project execution plan being signed off by the Chief Nuclear Officer. Any changes to the PEP also requires approval by the Chief Nuclear Officer.

Bruce Power intends to continue with the principles of the IIP program put in place by OPG for the Bruce “A” and Bruce “B” reactors.

POINT LEPREAU PERFORMANCE IMPROVEMENT PROGRAM (PIP):

In 1996, as a result of an observed decrease in safety performance, the CNSC added a licence condition for NB Power to report to the Board every six months on actions being taken for improvement (this condition is no longer included in the Point Lepreau operating licence). NB Power initiated a number of independent assessments to clarify specific areas where improvement was necessary. By April 1997, NB Power had initiated the Performance Improvement Program (PIP), comprising 47 separate projects. It had also determined that the primary root causes for the decreased performance were: corporate failure to understand lifetime management of the station, and failure to develop strategic planning and allocate appropriate resources.

In August 1998, the PIP program was expanded to 52 projects. The key focus areas were: Supervisory Effectiveness, Work Processes, Plant Aging and Sustaining Performance. These projects were prioritized and placed in three groups (see definitions of the groups below). The Performance Improvement Projects are listed in Table 6.9:

Group A: containing milestones and targets to address the four main focus areas

Group B: projects scheduled consistent with the ability to acquire qualified resources

Group C: projects scheduled as resources become available within the organization

As of April 2000, the Work Control Process (A2), Maintenance Backlog (A3), Strategic Plan Development (A9) and Operating Policies & Principles Compliance Improvement (B1) were complete. A total of 41 of 57 Group “A” targets have been met and 42 of 53 Group “B” targets have been met as of April 2000.

The success of the PIP was measured using station-specific performance indicators. Indicators used to measure performance improvement were:

- R-99 reportable events;
- INES level events;
- accident severity rate;
- number of outstanding safety work orders;
- combined maintenance backlog;
- preventative maintenance schedule;
- operational documentation deficiencies;
- station staffing levels;
- chemistry procedures;
- drill performance completion ratio; and
- outstanding corrective actions.

The performance measures, although not as strong in some cases, show that the PIP has had a positive impact at Point Lepreau.Reportable events have increased; however, this is attributed to:

- an increased recognition by staff on the importance of promptly reporting incidents outside the norm, and a better understanding of what needs to be reported;
- changes in the interpretation of reporting requirements; and
- stronger front-end process to screen potential reportable events.

The accident frequency rate has reduced slightly from 1996 to 2000, while the accident severityrate has dropped significantly over this time. The number of outstanding safety work orders, preventative maintenance work, operational documentation deficiencies have steadily decreased from 80 in June 1998 to less than 20 in March 2000. The combined maintenance backlog has decreased by approximately 50% over the time the indicator has been tracked. In addition, INES level events have decreased significantly. NB Power has also maintained its commitment to reversing their historical resource deficit.

Overall, the PIP has been partially successful in its goal to realize improved safety performance. Good progress has been made on the majority of the PIP projects that have been undertaken. To effect a higher level of improvement, success is necessary in improving organizational effectiveness and quality assurance. Establishment and implementation of a suitable quality assurance program is essential to success of the Business Planning process.

In May 2000, NB Power incorporated the Performance Improvement Projects in the Business Planning Process. NB Power intends to use the Business Planning Process as an effective method of initiating and tracking improvement projects in an integrated manner. NB Power intends to phase out the Performance Improvement Projects as a specific initiative, and use the Business Plan as an integrated method of controlling all future project work.

TABLE 6.9 Point Lepreau Performance Improvement Program Projects

Group “A”	Group “B”	Group “C”
A1- Supervisory Effectiveness	B1- OP&P Compliance Improvement	C- Define Communications Protocol
A2- Work Control Process	B2- Safety Culture Training and Development	C- Emergency Preparedness
A3- Maintenance Backlog	B3- Configuration Assurance Program	C- Chemistry Program
A4- Corrective Action Follow-up	B4- Procedural Improvement	C- Airlock Seal Replacement Program
A5- CNSC Action Item Progress	B5- Design Change Process	C- Active Waste Management Strategy
A6- Problem Identification and Corrective Action (PICA)	B6- Staffing and Succession Planning	C- Valve Program-AOVs, MOVs, Check Valves
A7- Work Management for Technical Work	B7- Comprehensive Training Program for Work Groups	C- Computerized Work Management System
A8- Plant Aging Program		C- Health Physics
A9- Develop a Strategic Plan		C- Seismic Qualification Program
A10- Self-Assessment Program		C- Communication Plan
		C- Revise PMS Program
		C- Improve OPEX
		C- WANO Audit

6.6 Corrective Actions and Monitoring Programs to Maintain and Improve Safety in Canadian Nuclear Power Plants

Corrective actions/programs for safety upgrading applied to specific power stations as results of safety assessments are listed in Annex 6.3. Such actions/programs have either been completed or are in the process of being completed.

6.7 Canadian Position for Continued Operation of Nuclear Power Stations

Although there has been some apparent reduction in safety margins 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. It is generally recognized that continuous improvement programs have been effective in improving nuclear power station design, operation and maintenance in Canadian nuclear power plants.

The industry has therefore committed to plans and programs to improve the standards of nuclear power station operations in a timely manner. The CNSC monitors the execution of these plans to verify that:

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

ARTICLE 7
Legislative and Regulatory Framework

7.1 A Comprehensive Description of the Canadian Legislative and Regulatory Framework

The *Constitution Acts* of 1867 and 1982 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; and
- the production, refinement or treatment of prescribed substances (includes deuterium, fissile and radioactive materials).

The *Atomic Energy Control Act* governed Canada's approach to regulating nuclear energy and materials. As regulatory practices evolved to keep pace with industry and to increase focus on health, safety, security and environmental protection, there were changes to the regulations made under the *Atomic Energy Control Act*, but the framework legislation itself did not change. Updated legislation was required to provide a modern statute for more explicit and effective regulation of nuclear energy. The *Nuclear Safety and Control Act (NCSA)* was passed by the Canadian Parliament in 1997 and came into force on May 31, 2000.

While the *Atomic Energy Control Act* encompassed both the regulatory and developmental aspects of nuclear activities, the *Nuclear Safety and Control Act* disconnects the two functions and provides a distinct identity to the regulatory agency. The Atomic Energy Control Board has been replaced by the Canadian Nuclear Safety Commission, underlining its separate role from that of Atomic Energy of Canada Ltd.

The authority and responsibilities of the CNSC are specified under Section 9 of the *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; and

  - 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 NSC 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; and
- establish a clearer basis for implementing Canada's policies and obligations concerning the non-proliferation of nuclear weapons.

Using its authority under the NSC Act and its regulations, the CNSC 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, devices and equipment;
- import and export of prescribed items; and
- radioisotopes.

Like its predecessor, the AECB, the Canadian Nuclear Safety Commission regulates activities involving nuclear energy or materials in Canada, from nuclear power plants and nuclear research facilities, to equipment for diagnosis and cancer treatment, the operation of uranium mines, fuel fabrication facilities, the use of radioactive sources for oil exploration, to radioisotopes used in various industries. The CNSC is also responsible for the administration and implementation of Canada's international obligations pursuant to existing bilateral and multilateral nuclear cooperation agreements, convention and undertakings.

The Commission comprises a maximum of seven members, as opposed to five under the former Atomic Energy Control Board. This provides a broader range of expertise, and permits members to sit in panels. The Commission is made a court of record with powers to hear witnesses, take evidence and control its proceedings, while maintaining the flexibility to hold informal hearings. The Nuclear Safety and Control Act sets out a formal system for review and appeal of decisions and orders made by the Commission, designated officers and inspectors.

The CNSC is responsible for subjects involving nuclear energy applications that would otherwise have been under provincial jurisdiction. Examples of such subjects include:

- occupational health and safety;
- regulation of boilers and pressure vessels; and
- 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 to 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 *Nuclear Safety and Control Act* binds the Crown, both federal and provincial, and the private sector. In addition, the *NuclearSafety and Control Act* provides authority for the Commission and the Governor in Council to incorporate provincial laws by reference.

In addition to the above, the *Nuclear Safety and Control Act* provides for the following:

- clearly defined powers for inspectors, bringing the powers into line with current legislative practices;
- provisions for certification/decertification of prescribed equipment and nuclear energy workers;
- increased penalties for non-compliance, and bringing them into line with current legislative practices;
- clear appeal provisions;
- the authority to order remedial actions in hazardous situations and to require responsible parties to bear the costs of decontamination and other remedial measures;
- the power to demand financial guarantees for operation, decommissioning and waste management;
- recovery of the costs of regulation from persons licensed under the Act; and
- the authority for:
    - strengthened security requirements at reactor sites;
    - enhanced requirements for transport of nuclear materials;
    - enhanced protection for nuclear energy workers, industrial radiographers and carriers of nuclear materials; and
    - measures requiring hospitals to provide radiation protection information to patients who undergo nuclear medicine therapy.

The CNSC regulatory regime also includes the control of certain nuclear materials and other nuclear items for purposes of non-proliferation and 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. Accordingly, the CNSC participates in activities of the International Atomic Energy Agency to ensure that Canada complies with the requirements of Article III of the *Treaty on the Non-Proliferation of Nuclear Weapons* (N.P.T.), and cooperates with other national governments in fulfilling the terms of bilateral nuclear cooperation agreements and in meeting other multilateral non-proliferation commitments.

The CNSC 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 CNSC, such as the requirements for special safety systems at nuclear power stations. Others are set by provincial authorities or national standards associations.

The CNSC'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 CNSC if there is no conflict with the provisions of the *Nuclear Safety and Control Act* and its regulations.

After a licence is issued, the CNSC carries out compliance inspections to ensure that its requirements are continually met. If the compliance inspection and assessment program identifies a non-compliance or an adverse trend that may eventually lead to a noncompliance, there is a range of possible actions that the CNSC can take. These range from a requirement for licensee action to prosecutions. Additional information on the CNSC compliance program is provided in Article 9.3.

Other legislation enacted by Parliament that pertains to control of nuclear energy are the *Nuclear Liability Act,* the *Canadian Environmental Assessment Act,* the *Nuclear Energy Act* and the proposed *Nuclear Fuel Waste Management Act*. These acts are discussed in more detail in Articles 7.2.2 through 7.2.5.

7.2 A Summary of the Laws, Regulations and Requirements Governing the Safety of Nuclear Power Stations in Canada

The CNSC operates within a legal framework that includes law and supporting regulatory documents. Law includes such legally enforceable instruments as acts, regulations, licences and orders. Regulatory documents such as policies, standards, guides, notices, procedures and information documents support and provide further information on these legally enforceable instruments. Together, law and regulatory documents form the framework for the regulatory activities of the CNSC. The *NSC Act*, as described above, is the top tier in the regulatory framework.

7.2.1 Regulations Issued under the Nuclear Safety and Control Act

In order to simplify the transition to the new regulatory system under the *NSC Act*, the number of substantive changes in the new regulations was minimized. Under the *AEC Act*, the technical requirements were specified in the *Atomic Energy Control Regulations*, the *Transport Packaging of Radioactive Materials Regulations* and the *Uranium and Thorium Mining Regulations*. Under the *NSC Act*, these requirements are specified in nine regulations. For the most part, the new regulations restate existing regulatory requirements in an updated and consistent format. Some of the new regulations subsume requirements previously imposed as licence conditions and some new regulatory requirements have been added, as discussed later in this section.

The regulations issued under the *Nuclear Safety and Control Act* are as follows:

- General Nuclear Safety and Control Regulations;
- Radiation Protection Regulations;
- Class I Nuclear Facilities Regulations;
- Class II Nuclear Facilities and Prescribed Equipment Regulations;
- Uranium Mines and Mills Regulations;
- Nuclear Substances and Radiation Devices Regulations;
- Packaging and Transport of Nuclear Substances Regulations;
- Nuclear Security Regulations; and
- Nuclear Non-proliferation Import and Export Control Regulations.

The *Cost Recovery Fees Regulations*, issued in 1996, are still in-force.

The regulations continue the practice of allowing licensees flexibility in how they comply with the requirements. With some exceptions, such as the transport packaging and licence exemption criteria for certain devices, the regulations do not specify in detail the criteria that will be used in assessing a licence application or judging compliance. The regulations provide licence applicants with general performance criteria, and lists of information and programs that they must prepare and submit to the CNSC as part of the licence application process. This information and specified programs, when referenced in the licence, become legal requirements for the licensee in question. This approach to nuclear regulation is consistent with the practice followed to date in Canada.

There are also the *Canadian Nuclear Safety Commission Rules of Procedure*, which do not impose requirements for health, safety and protection of the environment, but which set out rules of procedure for public hearings to be held by the CNSC and for certain proceedings conducted by Designated Officers of the CNSC. The Rules of Procedure place requirements on the CNSC staff and can be seen as a form of mandatory documentation that applies to the public, licensees and CNSC staff and commissioners with respect to the conduct of certain procedures.

Nuclear facilities were divided into two categories, Class I and Class II, to recognize the wide range of risks associated with different facilities, and to ensure that comparable facilities are licenced and regulated to the same standards. Class I nuclear facilities have been further subdivided to clarify the distinctions between various operations. Class IA covers facilities such as reactors, while Class IB applies to facilities such as medical isotope producers and uranium processors. Class II facilities, which present lower risks, include accelerators, and medical and industrial irradiators.

All regulations pertain to power reactor operation except the *Class II Nuclear Facilities and Prescribed Equipment Regulations* and the *Uranium Mines and Mills Regulations*. The *General Nuclear Safety and Control Regulations* and *Radiation Protection Regulations* are generally applicable to all nuclear facilities.

The new Act establishes more stringent regulations to ensure that public health and safety are better protected, for example:

- lower radiation dose limits have been established, consistent with the recommendations of the International Commission on Radiological Protection;
- updated regulations governing the transport and packaging of nuclear materials will also reduce unnecessary risks to health and safety or the environment; and
- security requirements at reactor sites have been strengthened.

The new regulations are also more transparent. For example, the security provisions for transport of certain nuclear material reflect a balance between the public's right to know about the movement of nuclear material in Canada and the need to ensure the physical security of the shipments. The Act and regulations afford the Commission discretion to ensure that those who "need to know" do know, while allowing Canada to abide by its international commitments regarding the security of certain nuclear material.

The new regulations require licence applicants to submit information on the potential effects of their operations on the environment, both for radioactive and non-radioactive hazardous substances. This information is used by the Commission, in consultation with other federal and provincial regulatory bodies, to establish the operating parameters for a nuclear facility.

A brief description of the regulations and substantive additions to the regulations are discussed in Annex 7.1.

7.2.2 Nuclear Liability Act

The *Nuclear Liability 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 makes provisions for Canada to enter into international arrangements that carry nuclear liability. At present, Canada is not a party to any such arrangement.

7.2.3 Canadian Environmental Assessment Act

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; and
- the proposal made use of lands administered by the Government of Canada.

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

In 1995, this process was succeeded by the Canadian Environmental Assessment Act (CCEA) (S.C. 1992, c. 37, as amended), 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 a screening environmental assessment. Note that should a comprehensive study be required under the CEAA, the process is somewhat different in that the Canadian Environmental Assessment Agency carries out a further public review of the completed assessment and the federal Minister of the Environment, as opposed to the Responsible Authority, makes the decision on the results of the assessment. The Minister also has the ability to appoint a review panel or mediator if he deems this appropriate.

The responsible authority has the responsibility to determine the scope of the project to be assessed and the scope of the factors to be considered; the responsible authority also directly manages the environment assessment process, ensures that the environment assessment report is prepared and is responsible for making decisions under the Act. The CNSC is a responsible authority for projects that it regulates.

In practice, the responsible authority may delegate to the project proponent the conduct of the environmental assessment and the design and implementation of mitigation measures and a follow-up program. The responsible authority alone, however, remains directly responsible for ensuring that the assessment is carried out in compliance with the Act, and for deciding on the course of action with respect to the project following the assessment.

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 Nuclear Energy Act

The *Nuclear Energy Act* came into force in May 2000, along with the *Nuclear Safety and Control Act*. This Act contains the portions of the *Atomic Energy Control Act* that pertained to the development and utilization of nuclear energy. The provisions of this Act apply to AECL as a company as defined in Section 2 of the *Nuclear Energy Act*.

7.2.5 Nuclear Fuel Waste Act

In April 2001, the Government of Canada introduced nuclear fuel waste legislation that would allow the government to ensure that fuel waste owners fully meet their financial responsibilities and carry out waste activities in a comprehensive, integrated, and economically sound manner.

The legislation, called *"An Act Respecting the Long-Term Management of Nuclear Fuel Waste"*, will require nuclear utilities to form a waste management organization as a separate legal entity. The organization would then be required to report regularly to the Government of Canada. This organization would provide recommendations to the Government on the long-term management of nuclear fuel waste. The legislation would also require the utilities to establish a segregated trust fund to finance long-term nuclear fuel waste management activities. It would also provide for a review and approval framework for government oversight, which would ensure that Canadian taxpayers are not liable for the long-term management of nuclear fuel waste.

The nuclear fuel waste legislation results from the Government of Canada's December 1998 Response to the Nuclear Fuel Waste Management and Disposal Concept Environmental Assessment Panel, also known as the Seaborn Panel. In March 1998, this Panel had made recommendations on the next steps, including disposal concepts, for the long-term management of nuclear fuel waste in Canada.

7.2.6 Regulatory Documents

In addition to the various legally binding regulations issued pursuant to the *NSC Act*, the CNSC issues documents on matters related to its regulatory mandate. These CNSC documents include policies, guides and standards on various matters or provide guidance to licensees on acceptable ways of complying with regulatory requirements. The guidance documents, by themselves, are not legally enforceable but 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 main classes of regulatory documents developed by the CNSC are:

- Regulatory Policy: a document that describes the philosophy, principles and fundamental factors used by the CNSC in regulatory programs.

- Regulatory Standard: a document that is suitable for use in compliance assessment and describes rules, characteristics or practices which the CNSC accepts as meeting the regulatory requirements.

- Regulatory Guide: a document that provides guidance or describes characteristics or practices that the CNSC recommends for meeting regulatory requirements or improving administrative effectiveness.

- Regulatory Notice: a document that provides case-specific guidance or information to alert licensees and others about significant health, safety or compliance issues that should be acted upon in a timely manner.

- Regulatory Procedure: a document that describes work processes that the CNSC follows to administer the regulatory requirements for which it is responsible.

All Regulatory Documents are developed through a consultative process. These are designated as Consultative Documents (C-XXX). Each document is made available for public comment in its draft form. At the end of the comment period, all public input is reviewed by CNSC staff. The document may then be revised, re-issued for further comment, withdrawn, or formalized as a CNSC Regulatory Document. Key regulatory documents are listed in Table 7.1.

Regulatory documents R-7, R-8, R-9 and R-10 contain the principal safety standards that the CNSC requires its licensees to meet for the containment system, shut-down system 1 (SDS1), shut-down system 2 (SDS2) and the emergency core cooling system.

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

TABLE 7.1 CNSC Regulatory Documents, Guides, Policies, Standards and Consultative Documents

Regulatory Documents	
R-7	Requirements for Containment Systems for CANDU Nuclear Power Stations (1991)
R-8	Requirements for Shutdown Systems for CANDU Nuclear Power Stations (1991)
R-9	Requirements for Emergency Core Cooling Systems for CANDU 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-91	Policy on Monitoring and Dose Recording for the Individual (1990)
R-99	Reporting Requirements for Operating Nuclear Power Facilities (1995)
R-100	The Determination of Effective Doses from the Intake of Tritiated Water (1987)
R-105	The Determination of Radiation Doses from the Intake of Tritium Gas (1988)
R-117	Requirements for Gamma Radiation Survey Meter Calibration (1995)
Regulatory Policies	
P-119	Policy on Human Factors (2000)
P-211	Compliance (May 2001)
P-223	Protection of the Environment (2001)
P-242	Considering Cost-benefit Information (2000)
Regulatory Standards	
S-106	Technical and Quality Assurance Standards for Dosimetry Services in Canada (1998)
Regulatory Guides	
G-129	Guidelines on How to Meet the Requirement to Keep All Exposures As Low As Reasonably Achievable (1997)
G-149	Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors (2000)
G-206	Financial Guarantees for the Decommissioning of Licensed Activities (2000)
G-219	Decommissioning Planning for Licensed Activities (2000)
G-228	Developing and Using Action Levels (2001)

TABLE 7.1 Continued

Consultative Documents	
C-6	Requirements for the Safety Analysis of CANDU Nuclear Power Plants
C-006	Safety Analysis of CANDU Nuclear Power Plants, Rev. 1
C-091	Ascertaining and Recording Radiation Doses to Individuals (an update of R-91)
C-98	Reliability Programs for Nuclear Power Plants (2001)
C-099	Reporting Requirements for Operating Nuclear Power Plants (an update of R-99)
C-118	Relationship Between Dose Limits for the Public and Operating Emission Levels for Nuclear Facilities
C-138	Software in Protection and Control Systems
C-144	Trip Parameter Acceptance Criteria for the Safety Analysis of CANDU Nuclear Power Plants
C-200	Radiation Safety Training for Radioisotope, Medical Accelerator and Transportation Workers
C-204	Certification of Persons Working at Nuclear Power Plants
C-205	Access Control for Protected and Inner Areas of Nuclear Facilities
C-208	Transport Security for Category I, II and III Nuclear Material
C-225	Emergency Planning at Class I Nuclear Facilites and Uranium Mines and Mills
C-229	Certification of Exposure Device Operators
C-273	Making, Reviewing and Receiving Orders Under the Nuclear Safety and Control Act
C-274	Preparing a Security Report for Licence Applications
C-276	Human Factors Engineering Program Plans
C-278	Guide to Human Factors Verification and Validation Plans

Regulatory document R-99, “Reporting Requirements for Operating Nuclear Power Facilities”, 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 “Requirements for the Safety Analysis of CANDU Nuclear Power Plants” (C-6), and “Safety Analysis of CANDU Nuclear Power Plants” (C-006 Rev. 1), contain the standards for safety analysis of CANDU nuclear power stations. Additional information regarding C-6, and C-006 Rev. 1, can be found in Article 6. Brief descriptions of the contents of the other documents listed in Table 7.1 are included in Annex 7.2.

7.3 A Description of the Licensing System for Nuclear Power Stations in Canada

The *Class I Nuclear Facilities Regulations* stipulate three formal licensing steps for nuclear power stations:

- a site preparation licence;
- a construction licence; and
- an operating licence.

The *General Nuclear Safety Control*, *Nuclear Security*, *Radiation Protection*, and *Nuclear Substance and Radiation Devices* Regulations also have requirements that must be met.

The *Class I Nuclear Facilities Regulations* now require licences for decommissioning and abandonment of a Class I nuclear facility. Requirements for abandonment of a nuclear substance, nuclear facility, prescribed equipment or prescribed equipment are included in the *General Nuclear Safety and Control Regulations*.

Requirements for a licence to site, construct, operate or decommission a Class I nuclear facility are listed below. Note that additional information is also required by Section 3 of the *General Nuclear Safety and Control Regulations*:

- a description of the site of the activity to be licensed, including the location of any exclusion zone and any structures within that zone;
- plans showing the location, perimeter, areas, structures and systems of the nuclear facility;
- evidence that the applicant is the owner of the site or has authority from the owner of the site to carry on the activity to be licensed;
- the proposed quality assurance program for the activity to be licensed;
- the name, form, characteristics and quantity of any hazardous substances that may be on the site while the activity to be licensed is carried on;
- the proposed worker health and safety policies and procedures;
- the proposed environmental protection policies and procedures;
- the proposed effluent and environmental monitoring programs;
- if the application is in respect of a nuclear facility referred to in paragraph 2(b) of the *Nuclear Security Regulations*, the information required by Section 3 of those Regulations;
- the proposed program to inform persons living in the vicinity of the site of the general nature and characteristics of the anticipated effects on the environment and the health and safety of persons that may result from the activity to be licensed; and
- the proposed plan for the decommissioning of the nuclear facility or of the site.

The CNSC's licensing system is administered with the cooperation of federal and provincial government departments in areas such as health, environment, transport and labour. Regulatory control is also achieved by setting standards and guidelines for the licensees. Some are prepared within the CNSC, and others are set by provincial authorities or national standards associations which may have specific conditions attached.

For all nuclear power stations, the seven-member Commission 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 one Commission hearing held over two days 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 Commission considers the applicant's request, recommendations from the staff of the CNSC, and any written or oral presentations from the public.

The specific stages of the licensing process are site acceptance, construction approval, commissioning and issuance of an operating licence. The site acceptance, construction approval and commissioning stages are described in Annex 7.3.

7.3.1 Operating Licence

Before it issues an operating licence, the CNSC 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:

- a description of the structures at the nuclear facility, including their design and their design operating conditions;
- a description of the systems and equipment at the nuclear facility, including their design and their design operating conditions;
- a final safety analysis report demonstrating the adequacy of the design of the nuclear facility;
- the proposed measures, policies, methods and procedures for operating and maintaining the nuclear facility;
- the proposed procedures for handling, storing, loading and transporting nuclear substances and hazardous substances;
- the proposed measures to facilitate Canada's compliance with any applicable safeguards agreement;
- the proposed commissioning program for the systems and equipment that will be used at the nuclear facility;
- the effects on the environment and the health and safety of persons that may result from the operation and decommissioning of the nuclear facility, and the measures that will be taken to prevent or mitigate those effects;
- the proposed location of points of release, the proposed maximum quantities and concentrations, and the anticipated volume and flow rate of releases of nuclear substances and hazardous substances into the environment, including their physical, chemical and radiological characteristics;
- the proposed measures to control releases of nuclear substances and hazardous substances into the environment;
- the proposed measures to prevent or mitigate the effects of accidental releases of nuclear substances and hazardous substances on the environment, the health and safety of persons and the maintenance of security, including measures to:
  - assist off-site authorities in planning and preparing to limit the effects of an accidental release,
  - notify off-site authorities of an accidental release or the imminence of an accidental release,
  - report information to off-site authorities during and after an accidental release,
  - assist off-site authorities in dealing with the effects of an accidental release, and
  - test the implementation of the measures to prevent or mitigate the effects of an accidental release;
- the proposed measures to prevent acts of sabotage or attempted sabotage at the nuclear facility;
- the proposed responsibilities of and qualification requirements and training program for workers, including the procedures for the requalification of workers; and

- the results that have been achieved in implementing the program for recruiting, training and qualifying workers in respect of the operation and maintenance of the nuclear facility.

Table 7.2 provides a partial list of the prerequisites that a licensee must submit 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 CNSC approval. Provided all has proceeded satisfactorily, a full operating licence may then be issued, usually for a term of two years. Among the terms of an operating licence is the requirement, through Regulatory document R-99, that the licensee inform the CNSC promptly of any occurrence or situation that could alter the safety of the plant. The Commission retains the right, by regulation, to impose additional conditions at any time. A sample power reactor operating licence is provided in Annex 7.5 (note that in the French language version of the report, the current operating licence for the Gentilly-2 nuclear power plant is provided as a sample of a CNSC power reactor operating licence).

TABLE 7.2 Partial List of Prerequisites for an Operating Licence

Subject	Prerequisites
Safety Analysis	• Final Safety Report • Safety Analysis 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 • Preliminary Decommissioning Plan
Nuclear Liability Insurance	• Proof of Coverage Required by the *Nuclear Liability Act*

7.4 A Description of the System of Regulatory Inspection and Assessment of Nuclear Power Stations to Ascertain Compliance with Applicable Regulations and Licences

The general philosophy applied to power reactor regulation is that the licensee has the prime responsibilityfor safety and that CNSC staff perform an oversight function. This is because the licensees must make routine safety-related decisions in the day-to-day operation of the reactors. The licensees are expected to have in place a standard set of programs and processes to provide adequate protection to the environment and the health and safety of workers and the public (a representative list of licensee programs is included as Annex 7.4).

This being said, Section 24(4) of the *NSC Act* states

"No licence may be issued, renewed, amended or replaced unless, in the opinion of the Commission, the applicant:

(a) is qualified to carry on the activity that the licence will authorize the licensee to carry on; and

(b) will, in carrying on that activity, make adequate provision for the protection of the environment, the health and safety of persons and the maintenance of national security and measures required to implement international obligations to which Canada has agreed."

This clause places the burden of responsibility on the CNSC to confirm that the applicant is qualified to carry on the licensed activity, and that programs in-place are adequate to protect the environment, the health and safety of persons and maintain national security and measures required to implement international obligations to which Canada has agreed.

Assessment of the licensees' safety performance is done primarily through the following three mechanisms:

- compliance verification activities;
- safety performance indicators; and
- the review of safety significant events occurring at nuclear power plants.

This information is integral to the operating licence renewal process. Licence renewal is discussed below.

The approach to licensing is to plan and conduct a balanced assessment of the licensee programs and activities, with priority placed on certain areas based on performance history, risk and expert judgement. The assessment is used to provide the Commission with a comprehensive review of the licensee and the facility and a supported staff recommendation for any licensing decision as well as to guide on-going regulatory activities.

Licensing plans for each station have been developed for all power reactors. These plans are intended to encompass all regulatory activities associated with each nuclear generating station and coordinate, track and integrate the activities regarding the safety surveillance of a station. The elements of the plan are aligned with the subject areas included in the licence renewal process. Assessments prepared from plan implementation are then used in the licence renewal evaluation. In developing these plans, the results of assessments during the previous licensing period will be reviewed to ensure a comprehensive assessment, while focusing on areas of concerns identified during previous licence periods.

Compliance activities have been grouped as follows: rounds (area inspections), system inspections, operating practice assessments, and audits/appraisals. CNSC staff from site and head office carry out these compliance verification activities, which are currently being revisited as part of the work on the CNSC

Compliance Program (additional details on the Compliance Program are presented in Article 9). Compliance verification activities currently defined by CNSC Regulatory Activity Management Procedures are listed in Table 7.3.

Rounds are performed 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 are usually done according to preplanned inspection guides prepared for the specific occasion. Results are normally recorded in a CNSC report which is sent to the licensee for follow-up action as necessary and retained on file.

Audits are always preplanned to a high degree of detail with the acceptance criteria spelled out in advance. The staff members who conduct the audit 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. 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 a CNSC report to the licensee and follow-up actions are recorded and assigned target dates for completion.

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

Analysis of safety significant events taking place at nuclear stations is the third component used in evaluating the safety performance of a station. The CNSC staff objective here is not to duplicate reviews done by licensees, but to ensure that they have adequate processes in place to take corrective actions when needed and to feedback into day to day operation lessons learned from past events. CNSC staff will only carry out a detailed review of the most safety significant events.

TABLE 7.3 Compliance Verification Activities - CNSC Regulatory Activity Management Proceedures

Operating Practice Assessements
- Fuel Handling
- Waste Management
- Maintenance
- Startup
- Normal Operation
- Shutdown Safety
- Heat Sinks
- Outage Management
- Effluent Monitoring

Appraisals
- Security
- Radiation Protection
- Safety Culture
- Chemistry
- Emergency Preparedness
- Fuel & Physics

Audits
- Pressure Boundary
- Prescribed Substances
- Change Control
- Quality Assurance

Rounds
- Control Room
- Reactor Building
- Turbine Hall
- Battery Room
- Control Equipment Room

System Inspections
- Containment
- Emergency Coolant Injection
- Shut-down System 1
- Shut-down System 2
- Stand-by Safety Systems
- Safety-related Systems
- Electrical Systems

Observations
- Software Maintenance
- Setback/Stepback
- Turbine testing
- Emergency Drills

Interviews
- Fuel & Physics

7.4.1 Licence Renewal

As discussed above, a key aspect of granting or renewing is a licence is the determination that the applicant is qualified to carry on the licenced activity and has made adequate provisions to protect the environment, the health and safety of persons and maintain national security and measures required to implement international obligations to which Canada has agreed. Licensee programs required to support renewal of a nuclear power reactor operating licence application are listed in Annex 7.4.

The licensee submits a license renewal applicant that meets the requirements for operating licences (these were outlined in Article 7.3.1).

As part of the licence renewal process, CNSC staff produces a Commission Member Document with the emphasis on the performance of both the licensee and the station. A standard document format is used in the licence renewal process. The CNSC's review process focuses on obtaining assurance that the risk to 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 CNSC regulatory requirements and accommodates the licence renewal cycle. CNSC 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 *Nuclear Safety and Control Regulations*;
- relevant regulatory documents;
- industry codes and standards;
- the facility operating licence; and
- pertinent station policies and procedures.

The review process considers the following aspects of plant management and operation:

- Operating Performance
  - Review of Station Operation
  - Conduct of Operations
  - Technical Surveillance
  - Reportable Events
  - Action Item Management
  - Approval Process
  - Outage Management
  - Non-radiological Health and Safety

- Safety Analysis
  - Management of Generic Safety Issues

- Equipment Fitness for Service
  - Design Adequacy
  - Maintenance
  - Periodic and In-service Inspection
  - Reliability
  - Environmental Qualification
  - Nuclear Plant Life Assurance
  - Fire Protection
  - Chemistry Control

- Performance Assurance
  - Quality Assurance
  - Human Factors
  - Organization and Management
  - Corrective Action and Feedback from Operating Experience
  - Configuration Management
  - Non-proliferation and Safeguards

- Training, Examination and Certification
  - Training
  - Examination and Certification

- Emergency Preparedness
- Environmental Performance
  - Review of Unplanned Releases

- Radiation Protection

- Nuclear Security

- Safeguards

7.4.2 Performance Indicators

To strengthen the safety review process, the CNSC has developed a set of 17 safety-related performance indicators. These indicators are 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 with respect to safety. These indicators are used in the licence renewal process, in annual reviews of station performance and in the annual report on industry performance.

The indicators are listed below and cover five performance areas of the station: operations, maintenance, public safety, worker safety and compliance. Reporting of these indicators is a licence condition in all power reactor operating licences, through the requirements of Regulatory Document R-99, “Reporting Requirements for Operating Nuclear Power Facilities, 1995.

The performance indicators are to be used in conjunction with other information gathered by the CNSC. The overall regulatory safety assessment process includes the conclusions drawn from the performance indicators, from event analysis and from compliance verification activities. Conclusions drawn from these three elements, taken alone or in combination, may result in additional regulatory inspections. The CNSC performance indicators are:

- Accident Severity Rate;
- Chemistry Index;
- Compliance Chemistry Index;
- Change Control Indices;
- Emergency Preparedness Drill Completion Ratio;
- Emergency Response Equipment Call-up Completion Ratio;
- Number of Minimum Shift Complement Non-Compliances;
- Non-Compliance Index;
- Number of Pressure Boundary Degradations;

- Preventive Maintenance Completion Ratio;
- Non-Compliances With Number of Specific Radiation Protection Procedures;
- Radiation Occurrence Index;
- Unavailability of Safety-Related Systems;
- Station Whole Body Dose;
- Number of Missed Mandatory Safety System Tests;
- Number of Unplanned Transients; and
- Unplanned Capability Loss Factor.

Some of the indicators can be used to measure the station performance as a whole, while some are more suited to measure performance in specific programs. Specification sheets that provide, among other things, the purpose and calculation method for the indicator, and data sheets have been developed to ensure standardized reporting.

These indicators have predictive or reactive attributes, or both. Predictive indicators can be used to measure trends and allow inferences to be made about any likely future deterioration in performance. These indicators can be used as an early warning to identify potential problems and take corrective action before they become safety issues. Reactive indicators prompt immediate action to correct deficiencies and prevent further deterioration.

An annual staff report on the industry, which covers all power reactors, is prepared by the CNSC. This report is the integration of information gathered through CNSC staff assessment activities of the Canadian nuclear power industry operational safety and evaluation of the station-specific performance indicators. The report makes comparisons where possible, shows trends and averages and highlights significant issues that pertain to the industry at large. The industry report addresses the subject areas evaluated in the licence renewal process, and CNSC performance indicators are used to enable comparison between the stations.

Industry reports covering 1998, 1999 and 2000 show that significant progress is still required in the areas of quality assurance, reducing maintenance backlogs, personnel continuing training and configuration management. These reports also demonstrate that the industry has consistently performed well in the areas of overall station operation, technical surveillance, outage management, non-radiological health and safety, chemistry control, environmental protection and radiation protection.

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 CNSC can take in the event of noncompliance including:

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

See Annex 6.3 for some of the major design and operational changes resulting from CNSC actions.

A licensee who is subject to enforcement action is entitled to request a hearing before the Commission to contest the action taken by CNSC 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 Commission before the action is taken. The *Nuclear Safety and Control Act* gives the Commission the authority to suspend a licence without prior notice, where it is necessary to do so in the interests of health, safety or security. In this case, the licensee may request the Commission to hold an inquiry into the reasons for the suspension. Where warranted, prosecution is also an option. Following are some examples of specific instances of noncompliance, the severity of which might lead to prosecution:

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

Licence sanctions recommended by an inspector can be imposed by the CNSC 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 CNSC is not satisfied that a licensee has the required commitment to safety, as indicated by the current compliance history, the staff may recommend to the Commission 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: CNSC staff may recommend a licence amendment to the Commission. The licensee is notified in writing of the proposed action and is given an opportunity to be heard by the Commission. 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 CNSC approval before reactor startup; and/or:
  - a requirement to appear before the Commission on a regular basis to provide status reports on progress in improvements to operation and maintenance programs.

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

  - the licensee is in serious noncompliance;
  - the licensee has been successfully prosecuted;
  - the licensee has a history of noncompliance; and/or
  - the CNSC has lost confidence in the licensee's ability to comply with the regulatory requirements.

In addition to the CNSC actions described in Article 6.2 and Annex 6.3, the CNSC 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

The history and development of the Canadian legislative and regulatory framework, and the creation of the Atomic Energy Control Board and its successor, the Canadian Nuclear Safety Commission (CNSC) were described in Article 7.

8.1 Position of the CNSC in the Government Structure

The CNSC is a departmental corporation, named in Schedule II of the *Financial Administration Act*. The *NSC Act* stipulates that the CNSC 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.

According to Section 19 of the *NSC Act*: "the Governor in Council may, by order, issue to the Commission directives of general application on broad policy matters with respect to the objects of the Commission." However, it is a constitutional convention that any political directives given to agencies such as the CNSC are of a general nature and cannot interfere with Commission decisions in specific cases.

The Commission requires the involvement and support of the Minister for special initiatives, such as the introduction of new legislation, and such as regulations.

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

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

CNSC staff routinely interacts with management and staff of Natural Resources Canada in areas of mutual interest. Natural Resources Canada formulates federal government policy with regards 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 CNSC policies and licensing matters. In addition, the CNSC routinely interacts with the management and staff of the Department of Foreign Affairs and International Trade in ensuring that Canada's international commitments pursuant to bilateral and multilateral treaties, conventions and understandings are fulfilled. CNSC licensees include several publicly-funded institutions or agents of the federal and provincial governments, as well as privately-funded organizations. These include:

- Atomic Energy of Canada Limited (the federal nuclear research and development company);
- nuclear operations of provincially owned electrical utilities: Ontario Power Generation, New Brunswick Power, and Hydro-Québec, and operations of privately-operated Bruce Power;
- Canadian universities;
- hospitals and research institutions;
- provincial government ministries; and
- federal departments.

The CNSC staff organization is described in Annex 8.1

ARTICLE 9
Responsibility of the Licence Holder

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 of the nuclear power station can be expressed as follows:

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

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 responsibility for nuclear safety during everyday conditions, as well as during any non-operational phases, 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; and
- ensuring 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:

- ensuring that the licensed activity is carried out according to the requirements of relevant acts, regulations, and licences;

- ensuring 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;

- ensuring 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 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 an officer within the organization 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 an officer within the organization to an act of sabotage or attempted sabotage anywhere at the site of the licensed activity;

- ensuring 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 releases and personnel doses shall be kept below the levels stated therein and consistent with the principle of "As Low As Reasonably Achievable" (ALARA); and

- 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).

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; and

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

See Article 10 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:

- design configuration control;
- analysis reviews;
- reliability studies;
- risk management and control;
- peer evaluation; and
- external reviews.

These activities were discussed in detail in Canada's report to the First Meeting of the Parties to the Convention on Nuclear Safety, and are included in this report as Annex 9.1.

9.2. Description of the Mechanism by Which the CNSC Ensures that the Licence Holder Meets its Primary Responsibility for Safety

The CNSC 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 CNSC approval (see Article 10.1 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 CNSC.

Events related to the following are subject to the requirements of CNSC Regulatory Document R-99, "Reporting Requirements for Operating Nuclear Facilities." (Note: this list is not all-inclusive):

- non-compliance with the *NSC Act*, regulations pursuant to the act, or an order of he Commission, Designated Officer or an inspector;
- health and safety of persons (e.g., serious injury, unplanned exposure to ionizing radiation);
- releases of nuclear substances;
- failure of process systems;
- actuation of safety systems;
- degradation of the pressure boundary of a registered system;
- reactor and turbine control;
- safeguards and security;
- emergencies and external events;
- failure perform tests as required by a licence condition;
- failure to monitor or control a release path of radioactive materials that is required to be controlled; and
- discovery of a problem that reveals a hazard to the health and safety of persons, national security , or the environment that is different in nature, greater in probability, or greater in magnitude than was previously represented in licensing documentation.

  In each case, the licensee is required to make an oral event report to the CNSC 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.

- 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; and
  - 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 have 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; and
  - 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, CNSC routinely audits the licensee's compliance with its Operating Policies and Principles document. CNSC also carries out regulatory inspections to make sure there is adherence to station procedures.

9.3 The CNSC Compliance Program

Compliance program activities verify that the licensees comply with the requirements for the safe operation of the nuclear power facilities. Furthermore, through the program, the CNSC can request from or impose upon a licensee corrective action when those requirements are not met.

The CNSC compliance program is being revised in view of the *NSC Act* and the regulations. The CNSC Compliance Program aims to provide a balance between proactive incentives to encourage compliance and reactive control measures to ensure compliance. These incentives and measures are:

- promotional activities to encourage compliance;
- verification activities to assess the actual level of compliance; and
- graduated enforcement actions in cases of non-compliance.

The goal of promotion activities is to maximize compliance levels by strengthening the factors that encourage compliance and mitigating those that hinder compliance. Compliance promotion encompasses the following types of activities:

Communication activities: Such activities include informing the regulated community of the rationale behind the regulatory regime, and disseminating information to the regulated community about regulatory requirements and standards.

Consultation activities: Such activities include consulting with regulated areas through an established process in order to design realistic and achievable requirements and standards.

Information gathered through verification activities helps to confirm whether the risks associated with nuclear technology are limited to reasonable levels and whether applicable international control measures are effectively implemented. International consultations may also be required to ensure compliance with safeguards and non-proliferation obligations. To verify compliance with the regulatory requirements, the CNSC:

- 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; and
- 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, or on the other hand, verifying high standards of safety. 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 CNSC to meet this objective are designed to help protect the environment, the health and safety of workers and the public.

The criteria that can be used to assess compliance may be derived from:

- legal requirements;
- CNSC policies, and standards or guides that clarify how the Commission intends to apply the legal requirements;
- information, supplied by licensees to the Commission, that defines how the licensees intend to meet the legal requirements in performing the licensed activity; or
- expert judgement of CNSC staff.

Programs evaluated are those included in the licensing process. In verifying that licensees abide by their programs, the CNSC will check that the licensee's activities meet CNSC acceptance criteria defined during the licensing process.

Enforcement actions are an important part of compliance. They are intended to re-establish compliance and act as a deterrent against non-compliance. Consistently applied enforcement actions also encourage prompt identification and correction of items of non-compliance.

The enforcement actions available to the CNSC include discussion, verbal or written notice, warning, increased regulatory scrutiny, publicity, issuance of an order, licensing action (i.e., amendment or suspension of part of a licence), revocation of personnel certification, prosecution, and revocation or suspension of a licence.

The CNSC uses a graduated approach commensurate with risk or regulatory significance in choosing which enforcement action to use. Initially, depending on several factors, the CNSC seeks to use the least severe enforcement action considered likely to achieve the desired outcome. Subsequently, depending on the effectiveness of the initial action, the CNSC may use enforcement measures of increasing severity. The CNSC recognizes that situations will arise in which it is appropriate to use one of the more severe enforcement measures (e.g., prosecution) as an initial response.

The CNSC Compliance Policy is implemented through a corporate-wide program that integrates all elements of compliance. At the operational level, responsibility for implementation of the corporate-wide program is shared between the various CNSC service lines and technical lines (the service lines and technical lines are listed in Figure 9.1).

CNSC service lines are units of the CNSC organization that carry out the activities needed to regulate a given type of nuclear technology application. Compliance activities in the areas of promotion, verification, and enforcement are implemented through each of the service lines.

Technical lines are centers of expertise associated with specific nuclear technology specialties. Technical programs are administered by in-house CNSC technical experts who comprise the various technical groups. One or more technical programs may fall under the responsibility of a given technical group. Technical programs provide specialized information to the service lines and help to ensure that compliance with regulatory requirements is approached in a consistent manner by all service lines.

TABLE 9.1 CNSC Service Lines and Technical Lines

• Dosimetry Services • Examination and Certification • Imports and Exports • Irradiators • Non-Power Reactors • Nuclear Fuel Facilities • Nuclear Substance Processing Facilities • Nuclear Substances and Radiation Devices • Packaging and Transport of Nuclear Substances • Particle Accelerators • Power Reactors • Research and Test Establishments • Uranium Mines and Mills • Waste Management Facilities	• Decommissioning • Emergency Preparedness • Environmental Protection • Event Analysis and Investigations • Fire Protection • Geoscience • Human Factors • Laboratory Services • Quality Management • Radiation Protection • Safety Analysis • Security • Structural Integrity • System and Component Performance • Training Program Evaluation • Non-proliferation and Safeguards

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 the Canadian utilities, Atomic Energy of Canada, Limited (AECL), and several other companies that provide design, engineering and construction services to Canadian utilities. AECL performs reactor design and project management for its clients and accepts their safety and regulatory requirements. The utilities are the design authorities for their stations and 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. The CNSC harmonizes, standardizes and monitors such safety principles, procedures and mechanisms among the various facilities and across the country.

10.1.1 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; and
- commitment to quality assurance programs that meet national and international standards.

10.1.2 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 basic safety functions:

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

Refer to Article 18 and Annex 18.1 for more information about the CANDU design safety principles.

10.1.3 Operational Safety Principles

ESTABLISHMENT OF A SAFETY CULTURE

Safety is the highest priority for the operation of nuclear power stations in Canada.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 protect 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;
- setting radiation dose action levels for works that are lower than the regulatory limits; and
- developing and delivering systematic training to employees.

SAFETY POLICIES

The major administrative control for the implementation of the Nuclear Safety Policy is the Operating Policies & Principles (OP&P) for each station.These are approved by the CNSC 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.

The governing principle promulgated by the Operating Policies & Principles is to maintain station operation within the defined boundaries of the Safe Operating Envelope.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.Such principles include:

- operating limits affecting public safety must be adhered to;
- defence-in-depth shall be maintained;
- fall back actions/countermeasures are to be established; and
- conservative decision-making for improved overall safety is to be applied.

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 approved policies and principles.

SAFETY ASSESSMENTS

Safety assessments are required to be performed regularly by 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, and Annex 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, and promotion of communications between managers and their subordinates about task assignments, deadlines, quality, resources, employee concerns;
- establishing corporate and site performance indicators that support the right behaviour for safe, reliable operations; and
- setting radiation dose limits for workers that are lower than the regulatory limits.

10.1.4 Regulatory Control - Safety Principles

The safety principles in the CNSC regulatory control are accomplished 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 CNSC deems necessary in the interests of health, safety and security;
- maintaining a person designated as the Head of the Site Office 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; and
- establishing a licence renewal practice as a mechanism to ensure that there is compliance and periodic safety review (the licence is typically issued for two years, but the Commission has the authority to issue shorter or longer term licences).

Under the *Nuclear Safety and Control (NSC) Act*, the Canadian Nuclear Safety Commission has 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) continues to apply.The current practice continues with the Commission considering the recommendations of CEAA panels as part of its licensing decision process, and regulates 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.

10.2 Safety Procedures

10.2.1 Safety Procedures at the Utilities

As required by the CNSC, 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 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; and

- 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 products;
  - maintaining design intent;
  - adherence to the operating limits that affect public safety;
  - maximization of the availability of safety systems;
  - status of the unit/system is to be known;
  - maintaining defence-in-depth;
  - establishment of fall back actions/countermeasures; and
  - conservative decision making is to be applied.

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 CNSC. 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.

10.2.2 Safety Procedures at the Designer (AECL)

The health and safety of persons, and protection of the environment is given a high priority at AECL.AECL is committed to:

- meeting or exceeding 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 the personal health of all employees through the provision of pro-active safety and health programs;
  - protecting employees from unsafe conditions in the workplace, and ensuring that those that cannot be eliminated are controlled to keep the risk as low as reasonably achievable; and
  - ensuring that any radiation exposures and emissions of radioactive materials caused by activities, products and services, are significantly below allowable levels and that they are As Low As Reasonably Achievable (ALARA);

- 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;

- to continually improve environmental performance and to contribute to improvement in the environmental performance of the nuclear industry; and

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

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.

In Canada there are publicly and privately operated electric utilities. Each has the authority to raise revenue through:

- the sale of bonds and equities in the financial market; and
- 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 reviews and approves proposed rate changes. The electricity industry in the Province of Ontario is moving towards privatization. A major step in this direction is the lease of the Bruce “A” and “B” Units to Bruce Power.

Each utility assesses its human resource requirements and recruits qualified staff on the open job market. 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; and/or
- expanding their recruitment horizons to seek qualified personnel internationally.

If human resource demands were to 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/or redeploying qualified staff internally.

A fundamental conclusion of the Independent Integrated Performance Assessment of OPG (see Article 6.5 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, OPG 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.

OPG has applied for approval to return the Pickering “A” units to service (see Article 14.4) and has been increasing staffing levels commensurate with the return-to-service schedule. Bruce Power will be retaining OPG staff and resourcing according to any Bruce “A” re-start program. NB Power, through its Performance Improvement Program, has addressed its human resources deficit by increasing staffing levels from 480 persons in 1996 to 640 persons in 2000. Staff complement requirements are listed in Section 49 of the NSC Act and in the licence conditions.

The CNSC, in the licence renewal process for re-start of the Pickering “A” and Bruce “A” reactors will assess whether OPG and Bruce Power have the necessary human resources required to support safe operation of each nuclear power station throughout its life. In addition the CNSC has requested, through a licence condition, that Bruce Power report on their financial status on a regular basis to ensure that Bruce Power has the necessary financial resources required to support safe operation of their nuclear power stations throughout the lease period.

11.2 The Financing of Safety Improvements Made to the Nuclear Power Station During its Operating Life

The Canadian utilities maintain budgets for operation and maintenance and for capital improvements. 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 in to 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, and/or result in a negative impact on the financial resources available for other purposes bearing on nuclear power station safety.

Expenditures are dictated by the company’s financial position, current and planned performance, service obligations (load forecast), and financial and business strategies. These inputs are used to develop the envelopes for ongoing operating expenditures and for capital investments. The costs of safety improvement programs/projects are part of the rate base and are recovered through rates charged to customers.

Canadian utilities place a high priority on safety-related programs/projects. This high priority ensures 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

Subsection 24(5) of the NSC Act provides the legislative basis for requiring licensees, including licensees of nuclear power stations, to provide financial guarantees in a form acceptable to the regulator for the decommissioning of nuclear power stations and management of the resulting radioactive wastes including spent fuel.

Paragraph 3(1)(l) of the General Nuclear Safety and Control Regulations states “An application for a licence shall contain a description of any proposed financial guarantee relating to the activity to be licensed.”

A Regulatory Guide for the provision of financial guarantees for decommissioning activities has been published (Regulatory Guide G-206, Financial Guarantees for the Decommissioning of Licensed Activities, June 2000). The Regulatory Guide G-219, Decommissioning Planning for Licensed Activities, provides guidance on the preparation of plans for the decommissioning of activities licensed by the CNSC.

Proponents and operators of nuclear power stations must propose decommissioning plans. The decommissioning plans must be sufficiently detailed in order to:

- demonstrate that all significant impacts and hazards to persons and the environment will be remediated;
- demonstrate compliance with all applicable requirements and criteria established in Acts, Regulations and other Regulatory Standards and Guides; and
- enable credible estimates of the amount of financial guarantees required to implement the plans.

The acceptability of any proposed financial guarantees for decommissioning will be ultimately determined by the CNSC 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 CNSC, 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.

Measures to fund decommissioning may include: cash, letters of credit, surety bonds, insurance and expressed commitments from a government (either federal or provincial).

11.4 Impact of Electricity Market Deregulation and Privatization in Canada

Until 2001, all power reactors in Canada have been operated by provincially-owned utilities. However, the government of Ontario has passed legislation aimed at introducing competition into the electricity market. The former Ontario Hydro, now Ontario Power Generation, has been split into generating and distribution companies, and the government is moving toward establishment of an open market. To introduce competition, Ontario has legislated that OPG must divest itself of a significant proportion of its generating capacity within a fixed time of the market opening.

In July 2000, Bruce Power Inc., a company largely controlled by British Energy and Cameco, submitted a licence application to operate the Bruce “A” and Bruce “B” stations. Bruce Power has agreed to lease the Bruce “A” and “B” stations from OPG for 18 years, with an option to renew for a further 25 years. The New Brunswick government is currently evaluating future options for NB Power’s Pt. Lepreau NGS. Increased economic competition is being introduced into the electricity market in Canada and these changes present a number of challenges with regards to maintaining an acceptable level of safety, for both the operator and the regulator.

Subsection 24(4) of the NSC Act requires an assessment of the qualifications of an applicant to undertake any licensed activity. With the prospect of a privately-owned licensee (Bruce Power), CNSC staff assessed the technical and financial qualifications of the applicant closely since the applicant did not have a history of nuclear station operation in Canada. The CNSC performed an overall assessment of Bruce Power’s competency to operate the Bruce “A” and Bruce “B” nuclear stations, based on a review of the licence application, the lease agreement, and a review of the applicant's operating performance in other jurisdictions.

While the CNSC has not yet developed policies in this area, Subsection 24(4) of the NSC Act requires that any applicant for a power reactor operating licence must demonstrate that it is technically and financially qualified to operate the facility safely. The main expectation of the CNSC is that licensees should have access to funds to cover extended unplanned outages, when the plant is not earning revenue.

The CNSC reviewed the lease agreement as part of the overall review of the licence application and found that Bruce Power will have the necessary authority for day-to-day control of plant operation. The lease requires Bruce Power to have adequate liability insurance, the minimum level being that required by the Nuclear Liability Act of Canada.

CNSC staff reviewed operational performance of British Energy in the United Kingdom and the United States, and recent audit and inspection reports by regulators in the United Kingdom and the United States and consulted with these regulators on the performance of British Energy nuclear reactors in their respective jurisdictions. These reviews confirmed that British Energy has significant operating experience and a proven safety track record with a range of non-CANDU nuclear reactors.

11.5 The Rules, Regulations and Resource Arrangements Concerning the Qualification, Training and Retraining of Personnel, Including Simulator Training for all Safety-related Activities for Each Nuclear Power Station

There is a hierarchy of laws, regulations and utility practices that specify the requirements for personnel who perform critical safety-related activities. The number of staff, their qualifications and their training are addressed in these documents.

Paragraphs 21(1)(i) and 44(1) (k) of the NSC Act provide the legislative basis for the certification, qualification, training and examination of personnel. Paragraphs 12(1)(a) and (b) of the General Nuclear Safety and Control Regulations specify that the licensee shall:

- ensure the presence of a sufficient number of qualified workers to carry on the licensed activity safely and in accordance with the Act, the regulations made under the Act and the licence; and
- train the workers to carry on the licensed activity in accordance with the Act, the regulations made under the Act and the licence.

The Class I Nuclear Facilities 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. Requirements are in place for the application for a licence to construct (Section 5(j)), the licence to operate (Sections 6(m) and 6(n))and in a licence to decommission (Section 7(j)).

Specific provisions are included in each operating licence which requires specific numbers of personnel, with explicit qualifications and training. Each licensee has practices that address these personnel issues.

Each operating licence contains general requirements to:

- ensure that there is a sufficient number of qualified workers to carry on the licensed activity safely and in accordance with the governing regulations; and
- 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 CNSC approval.

- have in a nuclear power station at all times the following certified positions, except as otherwise approved in writing by the CNSC (these are spelled out in the licence for each power plant and may vary depending upon the design of the plant):
  - at least four authorized nuclear operators;
  - one unit 0 supervising nuclear operator;
  - one control room shift supervisor; and
  - one shift manager.

- have in the control room sufficient persons who have been certified in writing by the CNSC 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 ("sufficient" certified persons is spelled out in each licence).

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

- are certified by the CNSC before the appointment of individuals to specific positions such as the following positions:
  - Shift Supervisor;
  - Control Room Operator; and
  - Station Health Physicist.

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

- meets the applicable qualification requirements referred to in the licence;
- has successfully completed the applicable training program and examination referred to in the licence; and
- is capable, in the opinion of the licensee, of performing the duties of the position.

A certification may be renewed after receiving from a licensee an application stating that the certified person:

- has safely and competently performed the duties of the position for which the person was certified;
- continues to receive the applicable training referred to in the licence;
- has successfully completed the applicable requalification tests referred to in the licence for renewing the certification; and
- is capable, in the opinion of the licensee, of performing the duties of the position.

Certification expires five years after the date of its issuance or renewal. The qualification process for certified 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 certified operators as part of the ongoing maintenance of their certified status.

Shift Supervisors and Control Room Operators 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, 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 (e.g., calibrate instrument loops, perform safety system tests, welding);
- identify incipient equipment failures, so that corrective action can be taken before catastrophic failures occur; and
- 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 personnel from other organizations such as:

- Engineering;
- Station Engineering Support;
- Station Health Physics;
- Maintenance Support; and
- Modifications.

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

11.6 Capability Maintenance

The Canadian nuclear industry is facing challenges similar to that of nuclear industries around the world. A potentially acute problem is the ability to support power plant operation, in terms of resolving safety-related and economic issues. This ability is referred to as capability maintenance. Elements of capability maintenance include:

- research and development program funding;
- status of facilities used in research and development programs;
- managing human resources within an organization (recruitment, succession planning, knowledge retention, maintaining expertise in core areas); and
- provision of trained personnel through university undergraduate and graduate programs.

Paragraph 24(4)(a) of the NSC Act states: “No licence may be issued, renewed, amended or replaced unless, in the opinion of the Commission, the applicant is qualified to carry on the activity that the licence will authorize the licensee to carry on.”

This places on obligation on the CNSC to ensure that applicants are competent to operate the nuclear facility. As such, the CNSC has requested power reactor licensees to address the following issues with respect to capability maintenance:

- identify all the engineering and technically based skills that are necessary to support safe operation of the plant. The inventory should include skills required for all aspects of normal operation and those anticipated to address emerging safety issues. If the skills are outside the organization and/or are shared with other organizations, this should be noted;
- for each of the skills identified, indicate the current complement and whether this is considered adequate;
- describe the means by which the inventory of skills will be maintained above or at the minimum complement; and
- describe what formal processes have been / will be put in place to deal with the issue of succession planning and workforce attrition.

In 1999, the CANDU Owners Group commissioned a report to evaluate the status of Research and Development Capabilities within Canada. In particular, the report addressed human resource and the state of R&D facilities. The report indicated that, in a number of areas, organizations that provide R&D support to the nuclear industry are operating at or near minimum capability. The study revealed that the workforce is ageing and entry rates into the nuclear industry are low. In addition, the study indicated that a number of facilities are at risk of closing in the short and medium term.

Canadian nuclear utilities and AECL, through the CANDU Owners Group, have taken steps to address capability maintenance throughout the lifetime of the plant through development of a strategic plan for maintaining the strength of R&D organizations that support the nuclear industry. This plan includes identification of the technical skills and facilities required to maintain and develop industry knowledge in key technical areas. Once the technical skills have been identified, the industry is proposing the development of CANDU-specific technical Centres of Excellence. In order to maintain expertise in the long-term, utilities propose to collaborate with the R&D organizations to ensure that appropriate succession planning is in-place.

Increased collaboration between industry and universities is a key aspect of long-term capability maintenance. Fundamental to long-term capability maintenance is the attraction, training and retention of new graduates for the Canadian nuclear industry. At the present time enrolment in nuclear-related programs and the entry rate into the industry are low. In addition, the scope of nuclear engineering education within Canada is limited. Post-graduate nuclear engineering-related programs are offered in three universities, while five universities offer undergraduate nuclear engineering-related training. The research performed at Canadian universities in support of the nuclear industry was described in Annex 1.1.

To address the issues of long-term capability maintenance, Canadian nuclear utilities have proposed the establishment of a Network of Excellence in Nuclear Engineering at Canadian universities. Stronger industry support will address two issues. Firstly, there is a need to provide undergraduate and graduate level training in nuclear engineering. In addition, there is a need to provide funding for development of programs that can be carried out at the university level. This would be part of the training for undergraduates and graduate students, and provide a means of maintaining expertise within university faculties. The CNSC is also contributing to this program.

Ontario Power Generation has committed funding towards the support of engineering programs at five Ontario universities, Queen’s, Toronto, McMaster, Waterloo and Western, in support of education and research in nuclear engineering. The funds will create five Research Chairs and sponsor up to 30 students in Master level programs. In addition, OPG has committed funding to a Natural Sciences and Engineering Research Council research chair in nuclear fuel waste research, and for nuclear engineering scholarships.

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Historically, the CNSC has recruited experienced personnel from universities and industry. However, the CNSC is facing the same human resources issues that the utilities and R&D organizations are facing. The CNSC is in the process of establishing human resources strategies which promote workforce sustainability through improvements to recruitment, retention, and succession planning. In addition, the CNSC has also developed and initiated a pilot internship program for power reactor regulation.

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. These entities use a variety of approaches to make sure that human factors expertise is available from dedicated human factors engineers/specialists on staff, and external human factors consultants.

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

All organizations recognise 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. OPG, AECL and Bruce Power have integrated this program 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 and training.

The goal of the human factors engineers/specialists is to reduce operations and maintenance errors by integrating human factors 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 and ongoing experience review.

For AECL, OPG and Bruce Power 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; and
- 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); and
- 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; and
- 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 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; and
- development of a systematic method for regulatory assessment of licensees' organization and management.

AECL also has a program that provides human factors training 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 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 has been 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:

- 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); and
- 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; and
- 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 engineers/specialists 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; and
- 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; and
- 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 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; and
- 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 (for example, laws, regulations and licence) and internal boundaries (for example, OP&Ps, Safety Reports, Radiation Protection Regulations and Quality Assurance Manuals);
- operating and maintenance practices; and
- design assumptions and intent.

12.2.4 Non-line Organizations Provide 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 Power Generation (OPG) 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 OPG Performance Assurance documents, licence requirements and commitments to the Canadian Nuclear Safety Commission (CNSC). 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 OPG 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 OPG functions; and
- initiates and coordinates internal and external nuclear safety reviews, Safety System Functional Inspections, audits and surveillance consistent with OPG performance objectives and criteria.

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

This committee accomplishes its objective by the review of:
- various safety activities;
- organizations;
- programs;
- procedures;
- requirements and results with respect to effectiveness;
- significance of occurrences; and
- 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 OPG, the Technical Advisory Panel on Nuclear Safety and the Nuclear Review Committee report directly to the OPG 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 CNSC is to ensure that licensees minimise the potential for human error in the design, operation, maintenance and decommissioning of their facilities. The CNSC Human Factors program covers a number of areas, including organization and management influences on safety, human-machine interface and workplace design, work organization and job design, procedures and job aids, operating experience and event investigation and human reliability analysis.

CNSC activities in the area of human factors include:

- review of significant design modifications and organizational changes;
- audits and evaluations of licensee programs which impact on human performance (e.g., corrective action/operating experience, engineering change control); and
- development of human factors regulatory documents.

12.3.2 Recent Human Factors Activities at CNSC

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

HUMAN FACTORS POLICY

In October 2000 the CNSC released Regulatory Policy P-119, Policy on Human Factors. The CNSC recognizes that human factors play a role in the performance of activities and facilities that it regulates.

Accordingly, policy P-119 describes how the CNSC will take human factors into account during its licensing, compliance and standards-development activities. When reviewing applications for CNSC licences the Commission will evaluate measures proposed by applicants and implemented by licensees to address human factors, and will determine whether these measures provide for the protection of the environment, the health and safety of persons, the maintenance of national security, and the implementation of international obligations to which Canada has agreed.

HUMAN FACTORS GUIDES

In early 2001 two CNSC Draft Regulatory Guides were released for public comment. These were C-276, Human Factors Engineering Program Plans and C-278, Human Factors Verification and Validation Plans. The draft regulatory guide Human Factors Engineering Program Plans guides licensees and licence applicants in their preparation of an effective Human Factors Engineering Program Plan that adequately incorporates human factors elements into licensable activities. In addition, this document supports P-119, Policy on Human Factors. The draft regulatory guide Human Factors Verification and Validation Plans assists licensees and licence applicants in preparing an effective Human Factors Verification and Validation Plan. The goal of this Plan is to establish that the human factors elements of a project or activity, that is licensed or licensable by the Canadian Nuclear Safety Commission, have been adequately addressed pursuant to document P-119, Policy on Human Factors.

DEVELOPMENTS IN SAFETY CULTURE INITIATIVES

A major goal of the CNSC was to develop and implement a systematic method for the regulatory assessment of licensees' organization and management before the year 2000. This goal has been achieved.

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. 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).

The methodology was piloted at one of the CNSC's nuclear generating stations, and has been validated at different licensee facilities, including a research reactor, particle accelerator, mine/mill facility and a conversion facility. It is now being used as a regulatory evaluation tool and to date has been applied at 11 licensee facilities. The results of the evaluations provide a descriptive profile of the organization, showing what organizational processes work and where improvements are needed. It is then used, in concert with information from a suite of audits, inspections and other evaluations, to provide comprehensive information for licensing decisions made by the Commission. At present, the CNSC is developing performance indicators, using all of the data collected to date.

HUMAN FACTORS IN ENGINEERING CHANGE CONTROL

During their life cycle, nuclear power plants undergo design modifications to address regulatory issues, and from internally driven initiatives which are intended to enhance safety and reliability, e.g., to replace ageing equipment. Design modifications frequently involve changes to the human-machine interface in the main control room and balance of plant. To ensure that safety is not adversely affected, licensees must systematically assess the impact of proposed design modifications on human performance and ensure that the design adheres to human factors principles and guidelines. In response to a CNSC action item, Ontario Power Generation has implemented a process for identifying and addressing human factors considerations as part of their overall engineering change control process. The process was rolled out at the various OPG stations in 2000.

HUMAN FACTORS IN DECOMMISSIONING

Work has recently begun on identifying the full scope of human factors issues and concerns that come into play during decommissioning. At the present, Regulatory Guide G-219, Decommissioning Planning for Licensed Activities, outlines some of the human factors issues which should be addressed in the Detailed Decommissioning Plan. Historically some of these issues have been addressed indirectly through the Health and Safety Plan, and through training provided to personnel performing decommissioning work. The CNSC has determined, however, that there are some issues that might be overlooked if a focused approach to human factors is not used. To that end the CNSC would require that licensees provide detailed plans for how they would address human factors issues through the various phases of decommissioning. To assist licensees in developing these submissions, the CNSC plans on developing a guide to human factors in decommissioning, which would identify the issues which the CNSC would consider in an application for a decommissioning license.

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 CNSC approval.

- have in a nuclear power station at all times the following certified positions, except as otherwise approved in writing by the CNSC (these are spelled out in the licence for each power plant and may vary depending upon the design of the plant):
  - at least four authorized nuclear operators;
  - one unit 0 supervising nuclear operator;
  - one control room shift supervisor; and
  - one shift manager.

- have in the control room sufficient persons who have been certified in writing by the CNSC 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 ("sufficient" certified persons is spelled out in each licence).

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

- are certified by the CNSC before the appointment of individuals to specific positions such as the following positions:
  - Shift Supervisor;
  - Control Room Operator; and
  - Station Health Physicist.

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

- meets the applicable qualification requirements referred to in the licence;
- has successfully completed the applicable training program and examination referred to in the licence; and
- is capable, in the opinion of the licensee, of performing the duties of the position.

A certification may be renewed after receiving from a licensee an application stating that the certified person:

- has safely and competently performed the duties of the position for which the person was certified;
- continues to receive the applicable training referred to in the licence;
- has successfully completed the applicable requalification tests referred to in the licence for renewing the certification; and
- is capable, in the opinion of the licensee, of performing the duties of the position.

Certification expires five years after the date of its issuance or renewal. The qualification process for certified 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 certified operators as part of the ongoing maintenance of their certified status.

Shift Supervisors and Control Room Operators 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, 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 (e.g., calibrate instrument loops, perform safety system tests, welding);
- identify incipient equipment failures, so that corrective action can be taken before catastrophic failures occur; and
- 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 personnel from other organizations such as:

- Engineering;
- Station Engineering Support;
- Station Health Physics;
- Maintenance Support; and
- Modifications.

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

11.6 Capability Maintenance

The Canadian nuclear industry is facing challenges similar to that of nuclear industries around the world. A potentially acute problem is the ability to support power plant operation, in terms of resolving safety-related and economic issues. This ability is referred to as capability maintenance. Elements of capability maintenance include:

- research and development program funding;
- status of facilities used in research and development programs;
- managing human resources within an organization (recruitment, succession planning, knowledge retention, maintaining expertise in core areas); and
- provision of trained personnel through university undergraduate and graduate programs.

Paragraph 24(4)(a) of the NSC Act states: “No licence may be issued, renewed, amended or replaced unless, in the opinion of the Commission, the applicant is qualified to carry on the activity that the licence will authorize the licensee to carry on.”

This places on obligation on the CNSC to ensure that applicants are competent to operate the nuclear facility. As such, the CNSC has requested power reactor licensees to address the following issues with respect to capability maintenance:

- identify all the engineering and technically based skills that are necessary to support safe operation of the plant. The inventory should include skills required for all aspects of normal operation and those anticipated to address emerging safety issues. If the skills are outside the organization and/or are shared with other organizations, this should be noted;

- for each of the skills identified, indicate the current complement and whether this is considered adequate;

- describe the means by which the inventory of skills will be maintained above or at the minimum complement; and

- describe what formal processes have been / will be put in place to deal with the issue of succession planning and workforce attrition.

In 1999, the CANDU Owners Group commissioned a report to evaluate the status of Research and Development Capabilities within Canada. In particular, the report addressed human resource and the state of R&D facilities. The report indicated that, in a number of areas, organizations that provide R&D support to the nuclear industry are operating at or near minimum capability. The study revealed that the workforce is ageing and entry rates into the nuclear industry are low. In addition, the study indicated that a number of facilities are at risk of closing in the short and medium term.

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Increased collaboration between industry and universities is a key aspect of long-term capability maintenance. Fundamental to long-term capability maintenance is the attraction, training and retention of new graduates for the Canadian nuclear industry. At the present time enrolment in nuclear-related programs and the entry rate into the industry are low. In addition, the scope of nuclear engineering education within Canada is limited. Post-graduate nuclear engineering-related programs are offered in three universities, while five universities offer undergraduate nuclear engineering-related training. The research performed at Canadian universities in support of the nuclear industry was described in Annex 1.1.

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ARTICLE 13
Quality Assurance

13.1 Quality Assurance Policies

It is a Government of Canada regulatory policy to use available international standards, guidelines and recommendations where those standards achieve the regulatory objective. For quality assurance, the Canadian Nuclear Safety Commission (CNSC) has given positive consideration to accepting Canadian national standards as equivalent and relied upon the use of the CSA N286 and CSA Z299 series of standards. Regulatory standards or guides on quality assurance are developed only when the available standards clearly do not achieve the regulatory objective.

The regulations that accompany the promulgation of the new Nuclear Safety and Control (NSC) Act require the implementation of quality assurance programs during the life cycle of the nuclear facility. An application to construct a nuclear facility must include the quality assurance program for its design. The licensee must also submit quality assurance 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. The nuclear regulator requires compliance with licence conditions that will correspond to each of these phases.

In 1976, the Atomic Energy Control Board (now the CNSC), proposed that standards on nuclear quality assurance be developed. The first committee was formed with participants from the industry and chaired by a member of CNSC 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 power plant. The standards cover:

- procurement;
- design;
- construction;
- commissioning;
- operating; and
- decommissioning.

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. The second edition of CSA Standard N286.7 was issued in 1999 to ensure that a complete set of quality assurance requirements is applied to computer programs in analytical, scientific and design applications. The first-tier CSA Standard N286.0, which covers general requirements for the overall quality assurance program, is under review for improvement and possible harmonization with international standards.

As a minimum, the quality assurance policies and programs that reactor 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 reactor licensee is required to develop policies for two levels of application:

- The first level policies apply to the owners 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 reactor. The “corporate” owner is then responsible for seeing that they are successfully completed. The organizations that are responsible for the work develop their own policies and practices for control purposes.

A quality assurance program is the umbrella program which assures that programs, standards, policies and procedures necessary for the safe operation of the facility exist, are documented and implemented in accordance with stated requirements. Historically poor performance in quality assurance has led CNSC staff to include a licence condition for all nuclear power plants specifying the Canadian Standards Association N286 series of standards as the regulatory requirement for quality assurance programs. This standard requires that the organization responsible for the plant establish and implement a quality assurance program for the items and services they are supplying.

13.2 Life-Cycle Application of QA Programs

Canada’s Nuclear Safety and Control (NSC) Act gives the Canadian Nuclear Safety Commission (CNSC) 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; and
- disposal.

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

The CNSC 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; and
- radiation protection personnel.

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

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 CNSC 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; and
- 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; and
- 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; and
- 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. 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 CNSC reviews the documentation that communicates the requirements of the QA program to licensee personnel in detail. When it is accepted, the CNSC 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; and
- overall compliance.

When deficiencies are detected, the licensee is notified and is required to correct them. The CNSC produces detailed reports of the audit findings and forwards them to the licensee for action and reply. The CNSC 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.

CNSC staff reviews concentrate on the licensees application of these standards and on their ability to demonstrate:

- consistent definition of roles and responsibilities for station programs;
- structured implementation of station programs;
- control of changes and program interactions; and
- internal self assessment and corrective action.

CNSC staff consider that weak performance in quality assurance is reflected in the effectiveness of station programs, and has a negative impact on safety.

Overall industry performance in quality assurance has historically been rated as conditionally acceptable by the CNSC. The extent of corporate oversight and the degree of implementation of quality assurance programs at all Canadian nuclear power plants must improve before the condition can be removed. Two major utilities initiated restructuring and enhancement initiatives.

Ontario Power Generation is continuing to make progress on their major improvement program, the Nuclear Asset Optimization Program (NAOP). One of the objectives of the program is to consolidate and simplify their core business processes in a set of governing documents. The majority of the documents have been generated and the CNSC is reviewing the governing documents to confirm that quality assurance program requirements have been adequately addressed.

New Brunswick Power is working with external consultants to define and document a consistent management structure that will satisfy the quality assurance program requirements. As a result of their new process mapping project plan, New Brunswick Power has identified the core business processes, identified the requirements for each process and has established working groups to flowchart and document each process.

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 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 CNSC staff compliance activities and change approvals are described in Article 8.2.1.

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

- a description of the design and its major safety features; and
- 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. Safety analysis requirements were discussed in Article 6. The document “Requirements for the Safety Analysis of CANDU Nuclear Power Plants” (C-6, June 1980) provides guidelines for the safety analysis. The CNSC assesses station-specific safety analysis reports and accompanying documentation. Once these reports and dcoumentation are approved by the CNSC, they become part of the basis against which verification of safety is measured.

The main features of the licensing process include the following:

- an environmental assessment is required for any new project;

- 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;

- a site licence is obtained;

- comprehensive safety evaluation is carried out by the CNSC technical staff for each stage of licensing;

- when construction approval is granted, a CNSC project office is established, with full-time resident staff, at the reactor site to enable surveillance to be maintained over safety-related activities throughout construction, commissioning and operation;

- as part of the licensing process, the CNSC 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; and
  - holding meetings to allow public participation in the review process;

- the staff of the CNSC 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; and

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

14.2 Licensee and Regulatory Control Activities Related to the Assessment and Verification of Safety

Continued monitoring and periodic safety assessments of Canadian nuclear power stations are performed by station operators and the CNSC. Activities involved in the assessment of station performance include:

- safety assessments (the safety assessments and their major results were discussed in Articles 6.2 and 6.3, and Annex 6.3);
- audits and inspections (refer to Article 7.4); and
- licence renewal assessments (refer to Article 7.4).

In addition, 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; and
- independent assessment.

Worker assessment is a continuous activity covering day-to-day, week-to-week experience, etc. 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. 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 the 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; and
- 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; and
- willingness to self-identify mistakes.

Without this basis, the ideas of performance-based assessments and self-assessment programs will not succeed.

OPG, NB Power, Bruce Power and Hydro-Quebec all have self-assessment programs in-place.

14.3 Safety Verification Programmes

Safety verification programs in effect include:

- maintenance;
- reliability;
- ageing management;
- integrity of pressure retaining components;
- fire protection; and
- environmental qualification.

Comments on these programs are provided in the following sections. Article 14.4, provides a description of licensee and regulatory activities in terms of return-to-service and refurbishment of ageing plants.

14.3.1 Maintenance Programs

At the present time, there is a license condition requiring licensees to develop and document a maintenance program. The specific licence condition regarding the requirement for a maintenance program is reproduced here.

- For the purpose of limiting, during the lifetime of the nuclear facility, the risks related to the failure or unavailability of any structure, system or component whose performance may affect the safe operation or security of the nuclear facility, the licensee shall establish, document and implement a maintenance program.

- The maintenance program shall include testing and inspection and shall be of such quality and be performed in such a manner that the availability, reliability and effectiveness of any structure, system or component remain consistent with the design and analysis documents listed in <u>licensee documents</u> (the type of document is facility specific).

An effective maintenance program should include the following features at a minimum:

- to ensure the presence of a sufficient number of qualified workers to implement the maintenance program safely;

- to train the workers to implement the maintenance program safely;

- 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 CNSC of any inspection findings that may indicate a significant flaw in a pressure-retaining component (periodic inspection program, in-service inspection program); and

- that reviews examine:

  - the adequacy of the schedule and its implementation;
  - the adequacy of past responses to operational requirements for corrective and preventive maintenance;
  - that, following maintenance, tests or in-service inspections of structures, systems and components, that the affected structures, systems and components are not returned to service until their respective configurations have been verified safe for a return to operation:
  - the adequacy of the control and management of preventive and corrective maintenance backlogs;
  - the control of radiation doses to maintenance workers;
  - compliance with quality assurance standards;
  - the adequacy of work procedures;
  - the effectiveness of monitoring, testing and assessment provisions; and
  - the detected incidence of SSC failures and any resulting impacts on health, safety, national security and the environment.

Since the mid-1980s, the CNSC 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 Reliability Programs

Safety systems must be able to reliably meet the performance requirements set by the safety analyses. Reliability requirements for the special safety systems (shutdown system 1 (SDS1), shutdown system 2 (SDS2), emergency core cooling systems (ECCS), and containment) are specified in Regulatory Guides for containment systems (R-7), shutdown systems (R-8) and emergency core cooling systems (R-9). These guides are specifically referenced in the Darlington NGS operating licence, but are not referenced in the Pickering “A”, Pickering “B”, Bruce “A”, Bruce “B”, Gentilly-2 and Pt. Lepreau operating licences.

The unavailability target for these special safety systems is 1×10^{-3}. The unavailability targets for other safety-related and safety-significant systems are specified in licensee documentation that has been approved and accepted by the CNSC.

Submission of an annual reliability report for all reactors is a requirement of CNSC Regulatory Document R-99. The reliability report is basically a summary of the reliability standard requirements. Typically, a reliability program for all safety systems (e.g. special safety systems, safety-related and safety-significant systems) of a plant will comprise a wide range of activities that are implemented at appropriate frequencies. These frequencies will vary, depending upon the nature of the activity. Some activities will be repeated frequently or on an ongoing basis over the duration of the reliability program. Other activities will be conducted less frequently - such as only once, periodically, or relatively infrequently over the duration of the program.

Accordingly, the reliability program for the safety systems of a plant, as described above, includes such infrequent actions as identifying and describing the safety systems, and defining their respective reliability targets, potential failure modes, design specifications, and performance requirements. In addition, the program includes on-going or repeat activities over its duration, such as assessments, inspections, tests, checks, monitoring activities and the recording and reporting of program performance and results.

At the end of each year, the licensee of each power reactor 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. 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; and
- 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 reliability analysis;
- any discovery of new failure modes or failure trends; and
- 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; and
- schedule for implementation of the above actions.

Canadian utilities have had reliability programs in place since the design and construction stage. A reliability standard “Reliability Programs for Nuclear Power Plants”, is under development. This standard formalizes the programs already in-place in Canadian nuclear power reactors. The public consultation process for this standard is now complete.

PROBABILISTIC SAFETY ANALYSIS

A Probabilistic Safety Analysis (PSA) is not a regulatory requirement in Canada. However, Regulatory Guide C-6, Rev. 1 (Safety Analysis of CANDU Nuclear Power Plants, September 1999) defines it as part of the Safety Analysis. Regulatory documents are under development: P-151, Policy on Risk-Informed Decision Making, G-152, Guidance for Risk-informed Decision Making and G-42, PSA Attributes for Regulatory Decision Making.

OPG has performed PSAs for Pickering “A”, Bruce “B” and Darlington. The PSA for Pickering “B” is in progress. More than 60% of Bruce “A” PSA was complete before Bruce “A” was shut down. The Darlington PSA is currently being updated, while conditions on the re-start of Pickering “A” were based on the results of the Pickering “A” PSA. OPG has begun to use probabilistic arguments in submissions to the CNSC. PSAs have not been carried out for Pt. Lepreau and Gentilly-2; however, the CNSC has indicated that a PSA would be part of the Periodic Safety Review that would be required if these reactors were to be refurbished (refer to Article 14.4 regarding refurbishment of Pt. Lepreau).

PSAs performed to date have shown that the main risk contributors differ significantly from one plant to another. This confirms the principle that PSA results cannot be used generically, but only on a plant specific basis. These PSAs have identified the importance of some plant features which, while known since commissioning, had not been deemed as safety critical. This has resulted in regulatory actions requiring plant improvements. Examples of this are the importance of the powerhouse emergency venting system at Bruce “B”, and the cross-link between Emergency Coolant Injection and the moderator at Pickering “A”. The cross-link weakness was known, but its safety significance was not recognized until the Pickering “A” risk assessment was completed. Removing the cross-link is a condition for Pickering “A” re-start (refer to Article 14.4.1 and Annex 14.1 for Pickering “A” re-start). Once OPG completed the Bruce “B” risk assessment and recognized the issue regarding the powerhouse emergency venting system and took appropriate corrective actions.

14.3.3 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 CNSC has embarked on the development of a regulatory position on requirements for the management of ageing.

In 1990, the CNSC 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; and
- the adequacy of the planned maintenance program.

In the early 1990’s, the CNSC 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; and

- ageing degradation that causes key system parameters such as flow rates, heat transfer rates and pressure drop to change to the extent that they impact on the limits assumed in the Safety Report.

Canadian nuclear power station owners have made reasonable progress in the development of ageing management programs. For some licensees, the overall life management programs provide the focus on safety with regard to identification of critical components and required performance standards needed to meet the proposed regulatory requirements. Other licensees have many of the required managed processes in place, but the programs have not been integrated into an ageing management framework.

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 licence renewal cycle, the CNSC has flexibility to adjust to these changes. However, the CNSC recognizes that the present safety review process needs to be improved in order to provide the level of assurance needed in the continued safe operation of nuclear power stations.

14.3.4 Integrity of Pressure Retaining Components (Pressure Tubes, Feeder Pipes and Steam Generators)

Licensees carry out periodic and in-service inspections on an ongoing basis to confirm that equipment important to station operation remains fit-for-service. This equipment includes steam generators, fuel channels, feeder piping and other high-pressure components.

PRESSURE TUBES

Managing aging of pressure tubes is one of the major aspects of ageing management in Canadian nuclear power reactors. Ageing can affect pressure tube integrity, and plant safety margins from the thermalhydraulics and reactor physics perspectives. Pressure tube ageing mechanisms include:

- corrosion and hydrogen/deuterium ingress;
- decreased fracture toughness;
- flaws created by fretting damage;
- hydride blister formation;
- irradiation damage and deformation (diametral and axial creep, and sag); and
- delayed hydride cracking.

Licensees have prepared, and are following life management strategies that will provide a good base on which to assess the condition of pressure tubes. Developing fitness-for-service criteria has been more difficult, despite a considerable body of data from in-plant monitoring and inspection, from examination of pressure tubes removed from reactors, and from research and development programs.

A Canadian Standards Association sub-committee was formed to prepare a consensus document that could be referenced as part of a future amendment of operating licences for Pressure Boundary Requirements. A Draft Standard CAN/CSA-N285.8-M2000, “Technical Requirements for In-service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors” is currently in preparation. The standard is based in part on fitness-for-service guidelines developed by a technical task team formed by the CANDU Owners Group. These guidelines utilized the applicable rules of Section XI of the ASME Boiler and Pressure Vessel Code. Since 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, additional rules were developed in the guidelines to address delayed hydride cracking in CANDU pressure tubes.

This standard incorporates the acceptance criteria, procedures, material property data and derived values used in the evaluations and assessments to confirm the acceptability of pressure tubes for continued service. The methods and the material property values used in the evaluations and assessments are standardized.

FEEDER PIPES

CANDU reactor inspections from 1995 onward has revealed unexpected reduction in the wall thickness of some feeder pipes that connect the reactor to the steam generators. The greatest wall thinning has been observed at the elbow and piping just downstream of the location where the outlet feeder pipe is connected to the outlet end-fitting (the end-fitting is designed to enable on-line re-fuelling, and connects the pressure tubes to the feeder pipes). The corrosion mechanism has been identified as flow-assisted corrosion. The licensees have determined the degree and extent of the wall thinning, and the acceptability of continued operation in both the short and long term. Research to identify service conditions that contribute to corrosion is ongoing and chemistry-based solutions to reduce the corrosion rate have begun.

Although feeder piping remains fit-for-service in the short term, the expected life of some feeders will be limited if the current rate of degradation continues. Licensees have developed life management programs to specify the long-term monitoring programs and are developing acceptance criteria. As part of the strategy for managing the degradation, licensees identified the relative severity and rate of degradation at their respective stations.

Ontario Power Generation has advised the CNSC that all of the Pickering, Bruce and Darlington units are experiencing flow-assisted corrosion which is thinning the wall of the feeder outlet elbow and the pipe immediately downstream. Extensive inspections of outlet feeders have been completed for three of the four units at Darlington. Monitoring inspections for Bruce “B” and Pickering “B” have been performed, and Pickering unit 4 has an inspection underway. Some short sections of outlet feeder pipes may need to be replaced in Darlington starting about 2003. Replacement of short sections of feeder pipes in the Pt. Lepreau and Gentilly-2 reactors also may be required in the next few years.

In 1997, a leak was detected in the channel S08 outlet feeder pipe in the Pt. Lepreau NGS. This was attributed to stress corrosion cracking of the carbon steel feeder elbow. At the time, it was believed that adjustments to the fuel channel fixed and floating bearings had a role to play in the failure.

Again in 2001, Point Lepreau was shutdown because of a heavy water leak that was traced to a crack in the first elbow of the outlet feeder of a fuel channel. Subsequent inspections found internal cracks in similar locations in two other outlet feeders. While the discovery of these cracks is cause for concern, the risk to the environment, the public and workers resulting from the leaks is negligible because the feeders demonstrated leak-before-break behaviour as expected. The leak rate of heavy water from the cracked feeder at the time the reactor was shutdown was about 15 kilograms/hour.

The safety analysis for CANDU reactors includes an assessment of the consequences of a complete severance of an outlet feeder as a small loss-of-coolant accident downstream of the reactor. The results of this analysis clearly show that the special safety systems will safely shutdown the reactor and maintain adequate fuel cooling without operator intervention for such an event.

The most likely mechanism for causing the damage observed in the Point Lepreau feeders in 2001was stress corrosion cracking. This was consistent with the assessments of the working group set up by the industry in 1997 to investigate the cracking in the S-08 feeder pipe. Investigation into the root cause of feeder failures is continuing.

The events at Point Lepreau clearly have potential implications for other CANDU reactors and, as a result, licensees have performed extensive inspections. Hydro Quebec inspected 197 outlet feeder elbows at Gentilly-2 with the same geometry as those that cracked at Point Lepreau during its 2001 outage. No crack indications were found in any of these elbows. Inspections performed on all other nuclear facilities have not revealed any other cracks to date.

The industry has clearly recognized the potential significance of wall thinning and cracking. A working group, with participation from all utilities and AECL, has been created under the auspices of the CANDU Owners' Group to further address feeder pipe integrity issues.

STEAM GENERATOR TUBES

The steam generators in Canadian nuclear power reactors have performed well and had few significant problems. Failure of a small number of steam generator tubes does not pose significant safety concerns but may allow a small release of radioactivity to the environment. Nevertheless, assurances that steam generator tubes do not deteriorate to the point where a large number could fail can only be obtained through detailed monitoring and management of steam generator condition.

The major steam generator degradation mechanisms include:

- divider plate leakage (all plants);
- stress corrosion cracking in the U-bend region and the top of the tubesheet (Bruce “A”) ;
- fretting in the region of the U-bend supports (Darlington, Bruce “B”);
- pitting (Pickering “B”);
- high cycle fatigue, loose parts and denting (all plants);
- wastage-type pits (Pickering “A” and “B”, Pt. Lepreau (when using phosphate chemistry)); and
- primary and secondary side fouling, which impacts on reactor inlet header temperature, heat transport system mass flow and safety margins (all plants).

Licensees have prepared, and are following life management programs that will provide a good base on which to assess continued fitness for service for steam generators throughout the industry.

14.3.5 Fire Protection

All Canadian nuclear power plant licensees have been reqested to carry out fire safety assessments (FSAs) of their facilities. A fire safety assessment generally consists of:

- Code Compliance Review - An area by area assessment of the as-found condition of the plant against applicable codes.
- Fire Hazard Analysis - An area by area assessment of plant fire risks and consequences.
- Fire Safe Shutdown Analysis - An assessment of the impact of fire on nuclear safety.

- Fire Emergency Response Capability - An assessment of the emergency response capabilities of the station and supporting local fire departments.

Significant improvements have been required for detection, automatic suppression, separation provisions, compartmentalization and fire response. All licensees are proceeding with the required fire protection improvements with notable progress at Darlington with the installation of new fire suppression systems and at Pickering with the installation of new fire detection systems. To date, Gentilly 2, Pickering “A” and Bruce “B” have completed station-specific fire hazard assessments, while the other sites are behind schedule.

The FSAs which have been completed to data, conclude that some improvements are required for detection, suppression, separation and emergency response provisions. All licensees are proceeding with assessments and/or upgrades. This is expected to result in significant improvements to safety. In general, the CNSC has been satisfied with the progress to date; however, to ensure continued progress, a standard fire protection licence condition requiring compliance with the National Fire Code, the National Building Code and with Canadian Standards Association fire protection standards for CANDU Nuclear Power Plants has been included in all power reactor licences.

14.3.6 Environmental Qualification

The environmental qualification program is intended to provide assurance that equipment needed to mitigate the consequences of an accident will function when exposed to the harsh accident conditions, and under post-accident conditions. This assurance must exist for the life of the equipment. Over the years, the environmental qualification process has not been well documented and there have been inconsistencies in the level of qualification provided.

To address these problems, licensees have been gradually reviewing and revising their programs and related documentation such as safety analysis, accident room conditions, and the environmental qualification list and assessments. The processes and procedural controls for environmental qualification are generally well developed and integrate other station activities, such as maintenance, material control and procurement, that may impact equipment qualification. Some licensees have replaced a number of power and control equipment items to enhance their environmental qualification status.

A common issue affecting Bruce “A”, Gentilly-2, Pickering “A” and Point Lepreau is the existence of polyvinyl chloride-insulated cables in special safety and support systems inside containment. Tests have shown that the cable insulation might fail in harsh environments. While this issue is being resolved at the Pickering “A” station through a cable replacement program, it remains under study, and outstanding, at the other sites.

Licensees have been requested to develop the environmental qualification programs, implement the design and equipment changes, and develop programs to maintain the stations environmental qualification status.

In general, the CNSC has been satisfied with the progress to date; however, to ensure continued progress, a standard environmental qualification licence condition has been included in all power reactor licences.

14.4 Plant Return-to-Service and Plant Refurbishment Programs

14.4.1 Return to Service of the Pickering "A" Reactors: Licensee and Regulatory Activities

Following the Integrated Independent Performance (IIP) Assessment performed by OPG, OPG shut-down the Pickering "A" and Bruce "A" reactors. Resources were directed towards the Integrated Improvement Program, and for improvement in the performance of the Pickering "B", Bruce "B" and Darlington reactors. OPG laid-up the Pickering reactors with the fuel remaining in-core, with the intent to re-start the units once the operating station improvements were well underway.

The primary objective of the Return to Service Project was to ensure that the Pickering "A" Station would operate safely when it was returned to service. In order to achieve this objective, the Return to Service Project incorporated a systematic and comprehensive review to ensure a high level of safety throughout the remaining plant service. It is important ot note that the Pickering "A" units were built in the late 1960's and early 1970's. There has been considerable development of requlatory requirements since then. It is therefore necessary to review the plant design and operation to determine whether it meets the intent of more recent requirements, and to make practical improvements to satisfy these requirements.

Several discussions and correspondence exchanges between CNSC and OPG staff were held on the requirements associated with restarting laid-up reactors before OPG formally notified CNSC of its intent to restart the units. OPG formally applied to restart the units at Pickering "A" in November of 1999. CNSC requirements regarding the restart of Pickering "A" were:

- All previously identified and committed safety upgrades should be completed.
- A comprehensive safety assessment should be completed. This should be based on either current standards, or original requirements where these can be shown to be satisfactory.
- In cases where current standards cannot be met and the original requirements are not satisfactory, a systematic review should be carried out to identify practicable upgrades.
- All upgrades essential to safety shall be installed.

Through a consultative process, OPG and the CNSC identified issues which needed to be addressed prior to returning the Pickering "A" units to service. The issues were documented in a "Basis for Return to Service" report prepared by OPG. Major work packages in the Pickering "A" return to service program were:

- Environmental Assessment;
- Shutdown System Enhancement;
- Comprehensive Safety Assessment and Systematic Review to Identify Upgrades;
- Review of Pickering A Risk Assessment Findings;
- Pickering Enhanced Assessment Findings;
- Installation of Upgrades Essential to Safety;
- Commissioning Program;
- Status of IIP Initiatives;
- Impact on Other IIP Initiatives;
- Staff Competencies;
- Core Discharge Monitors;

- Compliance with Governing Documents;
- Safety and Licensing Issues;
- Environmental Issues;
- Plant Material Condition Issues; and
- Integrated Improvement Projects.

SUMMARY OF THE PICKERING “A” RETURN TO SERVICE ASSESSMENTS

A summary of the major assessments performed in the Pickering “A” return to service project is provided below. These overlapping safety assessments ensure compliance with OPG’s Safety Policy and ensure necessary changes to improve safety are included in the Pickering A Return to Service project scope. Once the scope of work described in the Pickering A Return to Service document is complete, OPG will be able to return the Pickering A Station to power operations safely, reliably, in compliance with regulatory requirements, and it is not likely to result in any significant adverse effects on the environment.

- The Independent Integrated Performance Assessment, which was a comprehensive review of performance at OPG’s nuclear power stations in 1997. It concluded that operations were acceptable, but at the lower end of the acceptability range.

- The Pickering “A” Safety Report, which contains analyses of those single and dual failures considered most significant to public risk. The Safety Report demonstrates that the plant meets the licensing requirements.

- The Pickering “A” Risk Assessment reviewed the adequacy of the safety of the station design and operations. It concluded that the risk to the health and welfare of the population living or working in the vicinity of the Pickering “A” station was low in comparison to other risks to which they are normally exposed. Prior to the return to service of the Pickering A station, modification will be made such that the severe core damage frequency will comply with the nuclear safety policy of Ontario Power Generation.

- The Seismic Margin Assessment evaluated the ruggedness of systems, components, and structures during seismic events. It concluded that the majority of Pickering “A” success path systems, components and structures are of robust design and will safely operate as intended following a credible yet large magnitude earthquake that effects the Pickering area.

- The 1998 Environmental Review addressed thermal, chemical, metal and radiological releases, emission control measures, the results of environmental monitoring, and other key environmental issues identified by stakeholders. It concluded that Pickering management is moving towards a proactive and systematic approach to environmental management.

- The Environmental Assessment, from which the CNSC Commission decided that the proposed return to power operation of the four reactors, taking into account the mitigation measures described in the Screening Report, is not likely to cause significant adverse environmental effects.

- System Condition assessments, System Health assessments, and system inspections are being performed in order to ensure that the scope of the Pickering “A” Return to Service project includes the activities necessary to ensure the functionality of major systems.

- A "Systematic Review of Safety", which reviewed Pickering "A" against current codes, standards, and regulatory documents. The review showed that, in most cases, the Pickering "A" design meets the requirements or the intent of the requirement of the standards and regulatory documents. There were a small number of recommendations that were identified by the review. OPG committed to implement those changes that were required to meet the intent of current standards and regulatory documents to the extent practical and economical.

- A Fire Safety Assessment, which consists of:

  - Code Compliance Review - An area by area assessment of the as-found condition of the plant against applicable codes.
  - Fire Hazard Analysis - An area by area assessment of plant fire risks and consequences.
  - Fire Safe Shutdown Analysis - An assessment of the impact of fire on nuclear safety.

  The report concluded that, when the required modifications and procedures are implemented, the systems, equipment and operating procedures would assure the ability to achieve and maintain safe plant shutdown following any postulated fire in the plant.

- A review of fire protection at Pickering "A" against CSA Standard N293-95, Fire Protection for CANDU Nuclear Power Plants. The review concluded that, following the completion of the upgrades being implemented during the Pickering A Return to Service project, the design of Pickering A would meet the intent of all requirements of CSA N293-95 and that no further changes are necessary.

- A comparison between the Pickering "A" Return to Service Project and that of a Periodic Safety Review demonstrates that the previous and planned reviews of Pickering "A" have been, and will be, comprehensive and no major performance areas will be omitted.

In summary, the Pickering "A" Return to Service project demonstrates that:

- the Pickering "A" station was operated safely prior to the end of 1997;
- the units were then placed and maintained in a safe lay-up condition;
- a comprehensive review of the safety of the plant has been undertaken;
- the process for determining the scope of the return to service work is exhaustive;
- the return to service work scope is well defined;
- management processes are in place to control the work and ensure its completion, and to identify and respond to community concerns;
- sufficient staff will be available and competent to operate the station safely; and
- CNSC prerequisites for a recommendation on the return to service of the Pickering "A" station have been met.

It is important to note that prior to the restart of any of the Pickering "A" reactors, OPG will be providing a Completion Assurance Report on the installation and commissioning of the improvements and modifications specified in license renewal documentation prepared by the CNSC. The CNSC will then be undertaking verification of compliance with the Completion Assurance Report. The Commission may then authorize the re-start of the Pickering "A" units.

14.4.2 Refurbishment of the Point Lepreau and Gentilly-2 Reactors and Return to Service of the Bruce "A" Reactors

New Brunswick Power is currently evaluating the large-scale fuel channel replacement of the Point Lepreau reactor. This would involve replacement of the fuel channel assemblies, end-fittings and lower feeder pipes, which would enable operation for an additional 25 to 30 years. Other plant equipment would be replaced to allow for this extended period of operation. Components removed from the reactor, and station waste from station operation for the additional operating time will be stored in the Solid Radioactive Waste Management Facility. An environmental assessment will likely be required for the refurbishment of these reactors.

NB Power will be required to perform a safety review as part of the refurbishment project. This review will include:

- a review of the plant against current codes and standards to determine whether there are any gaps that could affect safety;
- a comparison of the scenarios in the Point Lepreau Safety Report against those contained in CNSC Consultative Document "Requirements for the Safety Analysis of CANDU Nuclear Power Plants" C-6, June 1980; and
- that factors identified in the IAEA Periodic Safety Review process are largely covered by either ongoing programs at Point Lepreau or other planned refurbishment work.

Studies planned to improve safety margins include:

- determination of upgrades to shutdown Systems to improve trip coverage;
- determination of changes to address pressure tube ejection;
- determination of changes to reduce the predicted future unavailability of ECC; and
- determination of changes to improve the moderator subcooling margin.

In addition to the items listed above, the following reviews will be performed to determine if additional safety improvements are necessary:

- review of Point Lepreau against the generic CANDU 6 PSA, and a Point Lepreau specific PSA;
- review of the ability of the safety systems, stand-by safety support and safety-related systems to meet their unavailability targets;
- review of changes to improve two-phase thermosyphoning; and
- review of fuel string gap measurement.

Currently, Hydro-Quebec is considering refurbishment of the Gentilly-2 NGS, while Bruce Power is considering the return to service of Bruce "A" units 3 and 4. The methodology developed and requirements specified for the possible refurbishment of Point Lepreau will apply for Gentilly-2 and the Bruce "A" units.

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

The NSC Act, which came into effect on 31 May 2000, resulted in a number of new sets of regulations including one set on radiation protection. The latter include many of the ICRP-60 (1991) recommendations for dose limits, as well as ICRP-65 (1994) recommendations pertaining to occupational exposure to radon progeny. The NSC Act gives the power to the Commission or a Designated Officer, inter alia, to authorize the return to work of persons who have exceeded a dose limit.

The new regulations address the following:

- implementation and requirements of licensee radiation protection programs;
- the requirements for recording of doses;
- the definition of action level and the actions to be taken when an action level has been reached;
- informing workers of the risks associated with radiation to which the worker may be exposed, and informing workers of effective and equivalent dose limits;
- the requirement to use licensed dosimetry services;
- effective and equivalent dose limits for nuclear energy workers, pregnant nuclear energy workers and persons who are not nuclear energy workers;
- dose limits that apply during the control of nuclear emergencies;
- actions to be taken when a dose limit is exceeded, and authorization of return to work;
- requirements for licenced dosimetry services;
- requirements for labelling of containers and devices; and
- requirements for posting of signs at boundaries and points of access.

15.2 Dose Limits

The CNSC prescribes dose limits for the general public and workers who may be exposed to ionizing radiation that results from the use and possession of radioactive materials and from operation of nuclear facilities.

In order to control stochastic effects, the protection quantity “effective dose” is used; it is expressed in units of mSv. To control deterministic effects the protection quantity “equivalent dose” is used; it is also expressed in units of mSv.

The new effective dose limits for workers are i) 100 mSv per five year dosimetry period, and ii) 50 mSv in any single year. The effective dose limit for a pregnant nuclear energy worker is 4 mSv, for the balance of the pregnancy. For members of the public, the effective dose limit is 1 mSv per year

To ensure that no worker exceeds the national dose limit for workers, nuclear power stations are required to establish action levels for station operation based on a fraction of the dose limits in the CNSC 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). Data on occupational dose to personnel in Canadian plants are provided in Annex 15.1.

Nuclear power stations have established effective station control for release of radionuclides to minimize the risk to the environment. Radionuclides can be released via gaseous and liquid pathways. These controls keep liquid and gaseous effluent releases well below their respective Derived Release Limits (DRL) which is an effluent release limit for a particular route of release from a particular station. If the station exceeds the DRL, the most exposed members of the public may exceed the dose limit. The releases of gaseous and liquid effluent releases from Canadian nuclear power stations in 1999 are listed in Annex 15.2.

15.3 Regulatory Control Activities and Radiation Protection

As part of its licensing and compliance program, CNSC staff continually assess nuclear facilities' performance against codes, standards, legal requirements and specific guidelines. The CNSC evaluation/ inspection program is the primary activity by which the CNSC verifies licensees' compliance with the regulatory requirements.

CNSC staff carries out many regulatory control activities related to radiation protection throughout the year. They develop and prepare regulatory documents and programs for radiation protection, environmental protection and emergency preparedness. These documents help nuclear facilities to interpret the requirements in the CNSC regulations.

The CNSC requires its licensees to keep all doses as low as reasonably achievable, social and economic factors being taken into account (ALARA). In practice, application of the ALARA principle ensures that doses actually received are much lower than the prescribed limits. The CNSC has issued Regulatory Guide G-129: "Guidelines on How to Meet the Requirements to Keep All Exposures As Low As Reasonably Achievable". The elements that the CNSC considers to be essential in the approach to ALARA are summarized as follows:

- a demonstrated management commitment to the ALARA principle;
- the implementation of ALARA through a licensee's organization and management, provision of resources, training, establishment of action levels, documentation and other measures; and
- regular operational reviews.

The CNSC has published Regulatory Policy P-223, "Protection of the Environment", which describes the principles and factors that guide the CNSC in regulating 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 in a matter that is consistent with Canadian environmental policies, acts and regulations and with Canada's international obligations. The policy will be accompanied by appropriate guides and standards to determine whether or not adequate provisions are in place for the protection of the environment.

Regulatory Guide G-228 is intended to help applicants for CNSC licenses to develop action levels in accordance with clause 3(1)(f) of the General Nuclear Safety and Control Regulations and section 6 of the Radiation Protection Regulations. Under the Radiation Protection Regulations, an action level is defined to be "a specific dose of radiation or other parameter that, if reached, may indicate a loss of control of part of a licensee's radiation protection program and triggers a requirement for specific action to be taken." The document provides guidance regarding the types of parameters that can be used in developing action levels, requirements for monitoring these parameters, and appropriate responses when an action level is reached.

To verify compliance with the requirements in the licences and the CNSC regulations, CNSC staff:

- review documentation and operational reports submitted by licensees;
- conduct radiation protection evaluations; and
- conduct evaluations of licensee environmental protection programs, emergency preparedness programs and other programs as required.

Staff also:

- monitor and evaluate radiation and environmental impacts of these licensed activities;
- review documentation and applications submitted by licensees and dosimetry service proponents;
- conduct on-site evaluations of dosimetry service applicants; and
- recommend approval of licence applications.

Another important method of assessing performance relates to unusual events. By regulation, licensees must report certain events to the CNSC. Licensees must further analyse these events to identify causes and determine trends. CNSC staff review licensees' reporting and analysis processes to verify compliance with regulatory requirements and licensees' effectiveness in correcting weaknesses. Significant events are investigated by CNSC staff.

Licensees have good control over radiation safety and dose to personnel during outages; however, issues remain regarding adherence to radiation protection procedures. OPG is currently upgrading their radiological environmental monitoring programs to improve environmental performance at their stations. New Brunswick Power has prepared a revised radiological environmental monitoring program. Hydro-Quebec's radiological environmental monitoring program is currently being revised.

Doses to plant personnel have been well below annual limits, while environmental releases have been a small fraction of derived release limits (refer to Annex 15.1 and Annex 15.2).

15.4 Environmental Radiological Surveillance

The nuclear power stations have established different programs to monitor the effect of station operations on human health and the environment. A federal agency, Health Canada, and the Province of Ontario also carry out monitoring programs around all nuclear power stations. This program was discussed in Canada's first report to the Convention on Nuclear Safety.

Health Canada established the Canadian Radioactivity Monitoring Network for environmental radioactivity in 1959. The initial role was to monitor fallout from atmospheric nuclear weapons testing. The current program is operated such that Health Canada can provide Canadians with accurate health assessments regarding existing levels of radioactivity and nuclear/radiological accidents from a national perspective. The present program consists of monitoring ambient gamma radiation at 34 sites, radioactive aerosols at 26 sites, and atmospheric tritium at 15 sites. These are augmented in a few locations with drinking water and milk sampling. The Ontario Ministry of Labour's Radiation Protection Service also monitors environmental radiation within the province of Ontario.

ARTICLE 16
Emergency Preparedness

16.1 A General Description of Laws, Regulations and Requirements for On-site and Off-site Emergency Preparedness

The CNSC requires applicants to assess the implications of their proposed activities, and to provide contingency plans to cope with potential accidents as a condition of licensing. After the plans have been reviewed and accepted by the CNSC, they become binding upon the licensee. As discussed in Article 7.4, on-site emergency procedures as well as an outline of measures to assist off-site authorities in planning and preparing to limit the effects of an accidental release are part of the regulatory requirements for the station operating licence.

Nuclear emergency preparedness and response in Canada is a multi-jurisdictional responsibility shared by all levels of government and the licensee. The Provinces have the primary responsibility for off-site nuclear emergency preparedness and response and designate municipalities within their jurisdictions that do nuclear emergency planning. The Federal Government also coordinates federal actions in support of the provinces during a nuclear emergency and has procedures to respond to emergencies with international or interprovincial implications. Potentially, this collective responsibility encompasses a wide range of contingency and response measures to prevent, correct, or eliminate accidents, spills, abnormal situations and 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 CNSC pursuant to the *Nuclear Safety and Control Act* and regulations.

- Off-site nuclear emergencies are those emergencies having an effect outside the boundaries of a Canadian nuclear facility. The Federal Government along with the facility provides support to the provinces.

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; and
- coordinating the national response to a nuclear emergency occurring in a foreign country.

The provincial governments are responsible for:

- the general health, safety and welfare of the inhabitants of their respective provinces and the protection of the environment. This includes lead responsibility for the arrangements necessary to respond to the off-site effects of a nuclear emergency;
- enacting legislation to fulfill the province’s lead responsibility for public safety;

- preparing emergency plans and procedures to ensure an appropriate response to a nuclear emergency and providing direction to municipalities so designated; and
- co-ordinating support from the nuclear facility and the Federal Government during preparedness activities and during response.

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 Federal Nuclear Emergency Plan (FNEP). This plan describes the federal government's preparedness and coordinates response to a nuclear emergency. Health Canada is the lead federal government department for the FNEP.

Under the common administrative framework of the plan, 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; and
- 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 (for example, Canada and the US) or international incident.

16.1.2 Dealing with Emergencies under the Federal Nuclear Emergency Plan and under Provincial Nuclear Emergency Plans

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; and
- 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 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; and
- 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.

Nuclear emergencies for which the FNEP could be activated cover a wide spectrum of probability and potential severity. In general, the probability that a particular emergency event will occur is inversely proportional to its potential severity. For example, there is a low probability that an emergency at a licensed nuclear power station in Canada will result in a large, uncontrolled emission of radioactive material. 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 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; and
- 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.

As the Chernobyl accident demonstrated, even a severe nuclear emergency at a major nuclear power station that is distant from Canada would have a limited effect. 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; and
- 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.

16.2 The Implementation of Emergency Preparedness Measures, Including the Role of the Regulatory Body and Other Entities

There are 19 federal departments/agencies in the 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 CNSC, in its role as a 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 to make sure that hazardous substances are transported safely and to help emergency response personnel handle related emergencies, including those involving radioactive materials. The CNSC 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. In the case of a nuclear emergency in the Province of Ontario, for example, the PNEP makes provision for the activation of a Joint Information Centre through which emergency information can be disseminated to the media and the public. This joint centre is staffed by representatives of the Province, the Federal Government, the affected municipalities, and the nuclear facility.

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.

Ontario, Quebec and New Brunswick have off-site emergency preparedness plans in place that would deal with emergencies that may occur at any of their nuclear power station installations. Summaries of the off-site emergency plans at the provincial level, the on-site emergency plans at the power stations, and the CNSC emergency preparedness and response plans are described below.

16.2.3 Province of Ontario

The Province of Ontario, Canada possesses the greatest number of commercial power reactors of any jurisdiction in Canada, and possibly North America (20 reactors). In addition, a research reactor is located at Chalk River and six U.S. nuclear facilities lie with 80 km of the province. As a result of this hazard, a nuclear emergency plan has been in place at the provincial level since 1986 (The Province of Ontario Nuclear Emergency Plan - PNEP). This plan has never been fully or partially activated, although events have occurred which resulted in formal notifications to the province, and these events were monitored until it was determined that there was no risk to the public or environment.

The legislation governing emergency preparedness and response in Ontario is the Emergency Plans Act R.S.O. 1990, c.E.9. The Act requires the government to formulate an emergency plan for emergencies arising in connection with nuclear facilities. The Act also permits the Province to designate municipalities that shall do nuclear emergency planning. Emergency Measures Ontario on behalf of the Ontario Ministry of the Solicitor General administers the PNEP and coordinates nuclear emergency preparedness and response in that province.

A nuclear emergency is defined by the PNEP as occurring when there is an actual or potential hazard to public health and property or to the environment from ionizing radiation or from a nuclear facility. The hazard may be caused by an accident, malfunction, or loss-of-control involving radioactive material or a nuclear facility.

The aim of the plan is, in the event of a nuclear emergency, to safeguard the health, safety, welfare and property of the inhabitants of the province, and to protect the environment. The PNEP, as the lead document for off-site nuclear emergency preparedness and response, coordinates the activities of Provincial ministries, nuclear facilities, the Federal Government including the CNSC, and designated municipalities in order to meet the objectives.

The PNEP details the arrangements in place for nuclear emergency planning, preparedness and response in Ontario. Various components covered in the plan include:

- aim and guiding principles;
- hierarchy of emergency plans and procedures;
- description of the hazard;
- planning basis;
- protective actions;
- concept of operations;
- emergency organization;
- operational policies;
- emergency information;
- public education;
- detailed responsibilities of the various participants; and
- committee oversight.

In the Province of Ontario, the PNEP sets out a core nuclear emergency training and exercise program for plan participants covering such components as emergency worker training, notification tests, emergency centre drills and full-scale exercises. A full-scale provincial exercise centred on a major nuclear power reactor is conducted every year with participation from the Federal Government.

16.2.4 Province of Quebec

Within the province of Quebec, the “OSCQ” has lead responsibility for emergency planning and response to nuclear emergencies regarding their impact outside the bounds of the Gentilly site. The “OSCQ” plan 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 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 venous 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.

16.2.5 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 and 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. The Point Lepreau Off-site Emergency Plan was developed by the NB EMO, 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 EMO, the Control Group coordinates implementation of the plan.

The Point Lepreau 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, 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.

16.2.6 Ontario Power Generation Nuclear Emergency Plan

The Ontario Power Generation (OPG) Emergency Plan is a corporate-level plan, which serves as the common basis of site-specific nuclear emergency preparedness and response arrangements at OPG's Bruce, Darlington and Pickering stations. (It is recognized that the Bruce stations are the responsibility of Bruce Power; however, this section describes the status of the emergency plans for the Bruce "A" and "B" units while being operated by OPG). It describes concepts, structures, roles and processes to implement and maintain an effective OPG 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 OPG 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 OPG Plan focuses on the release of radioactive materials from fixed facilities and on OPG interfaces with the Province of Ontario Nuclear Emergency Plan. The formal scope of the Plan excludes hostile (security) action incidents at OPG nuclear plants, as these incidents are dealt with in detail in other OPG 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.

The emergency plan is consistent with the corresponding OPG nuclear safety analyses and reports that were provided to the CNSC in support of individual applications for CNSC construction and operating licences.To implement its corporate Emergency plan, OPG has developed site-specific nuclear emergency preparedness and response arrangements for its stations.

In the event of an on-site nuclear emergency at an OPG power plant, OPG staff would immediately classify the nuclear emergency in accordance with criteria specified in the station emergency procedure. Should this emergency have off-site implications, OPG staff further categorizes it according to criteria contained in the Province of Ontario Nuclear Emergency Plan. To simplify this step, many events have been categorized according to the Province of Ontario notification designations. The result of this categorization exercise is Appendix D, "Notification Criteria Matrix, of OPG's corporate Emergency Plan. The site-specific emergency response procedures for Bruce, Pickering and Darlington include derivatives of Appendix D of the OPG Emergency Plan.

Emergency drills and exercises are an integral part of OPG's overall process of program assessment. These exercises are conducted periodically at all OPG power installations, in cooperation with other organizations and jurisdictions that have a role in nuclear emergency preparedness and response.

OPG maintains emergency public response capabilities within its Nuclear Public Affairs Department and the Ontario Power Generation Corporate Affairs Department. The primary targets of OPG's nuclear emergency public information program are the public who live or work near OPG nuclear power plants, and select OPG employees and contacts who need to know. In the event of a nuclear emergency involving an OPG facility, OPG 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 OPG 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, OPG issues news releases or verbal briefings to the local media, with copies to provincial and municipal officials. If the situation warrants, OPG may activate its on-site or near-site Media Centre 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, OPG'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 off-site response. OPG provides training, financial and personnel assistance to the JIC.

16.2.7 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 seventy 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", (Référence: Centrale nucléaire Gentilly 2 - Document de référence - DR-32/ Rev. 4 / Plan des mesures d'urgence (Plan de base); Octobre 1997) 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. 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.

16.2.8 Point Lepreau Nuclear Emergency Plan

New Brunswick (NB) Power includes its response arrangements for on-site nuclear emergencies at Point Lepreau in the group of documents it referred to as the Point Lepreau “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.

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 Point Lepreau “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 station boundary. The Point Lepreau 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.

An Off-site Emergency Centre is maintained by NB Power. This facility will be occupied by company staff following an incident involving abnormal radiation releases to the off-site environment. In such incidents, the Off-site Emergency Centre will be used as a communications centre, and to direct radiation monitoring programs.

16.2.9 Role of the Regulatory Body

The CNSC 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 CNSC would continue in its regulatory role, as anticipated in the FNEP and the CNSC Emergency Response Plan. The CNSC 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 CNSC'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 CNSC revised its Emergency Response Plan in May 2000. The CNSC continues to support and maintain an Emergency Operations Centre (at its headquarters in Ottawa) to enhance its ability to respond to nuclear emergencies. This facility is being used during ongoing FNEP and CNSC drills and training exercises to make sure and confirm nuclear emergency preparedness.

The CNSC Emergency Response Plan is the document that describes the strategies and guidelines that the CNSC will follow to cope with a nuclear emergency. It describes:

- emergency situations that could require CNSC involvement;
- the role of the CNSC in nuclear emergencies;
- the role of interfacing parties;
- the CNSC emergency preparedness organization;
- the concept of operations;
- the CNSC equipment infrastructure; and
- preparedness and training requirements and exercises.

The plan is issued under the authority of the President of the CNSC, 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 CNSC 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 CNSC Emergency Response Plan in the event of a declared emergency could involve:

- the CNSC emergency organization;
- CNSC employees;
- CNSC 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 Regulatory Commission; and
- the International Atomic Energy Agency.

The CNSC plan applies to all nuclear emergencies or potential nuclear emergencies that could require the regulatory or technical involvement of the CNSC. It is in effect at all times, in one of four operating modes: normal, standby, activated, or recovery.

- In the normal mode, the CNSC plans, trains and exercises to maintain its emergency preparedness. In this mode, the CNSC also responds to events which do not warrant activation of the emergency organization.

- In standby mode, the CNSC alerts responders and monitors the status of events which may require an emergency response at some stage.

- The CNSC Emergency Response Plan enters the activated mode of operations when the CNSC 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 CNSC emergency response plan, a nuclear emergency is any abnormal situation associated with a radiological activity or an CNSC-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;
- any other CNSC-licensed facility or activity; and/or
- any nuclear substance prescribed in the Nuclear Safety and Control Act.
- 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 CNSC Emergency Response Plan provides corporate guidelines for employee involvement. Emergency procedures set out the roles and responsibilities of various participants.

CNSC staff in this emergency organization are defined in the Plan, and depends upon the nature of the emergency. CNSC staff responsibilities in the event of a nuclear emergency parallel their responsibilities during routine CNSC operations. During a nuclear emergency, any member of the emergency organization or the CNSC could be called upon to perform special tasks.

The CNSC has established various technical and administrative arrangements in the interests of emergency preparedness. These arrangements form part of the CNSC's emergency response plan. They include bilateral cooperation agreements with other national and international jurisdictions, as well as CNSC 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.

16.3 International Arrangements, Including Those with Neighbouring Countries

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.

16.4 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). A summary of the CANATEX 3 exercise is provided below.

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).

16.4.1 Summary of the CANATEX 3/INEX 2 Exercise

CANATEX-3 was the third in this series, and was based on an exercise of the nuclear emergency response plans of Ontario Power Generation Nuclear, the Regional Municipality of Durham, the Province of Ontario, the Canadian Federal Government, and a joint federal plan between the United States and Canada.

Development of the exercise was coordinated through Emergency Measures Ontario, Emergency Preparedness Canada, OPG and Health Canada. The exercise took place in April 1999 and was based on a simulated accident at the Darlington nuclear power plant.

The objectives of CANATEX-3 were:

- to evaluate the Federal Nuclear Emergency Plan and its interfaces with the nuclear emergency arrangements of the Province of Ontario;
- to provide the opportunity for the Province of Ontario to evaluate the Provincial Nuclear Emergency Plan and the emergency arrangements of the affected regions, counties and municipalities;
- to test the bilateral arrangements of the Canada-United States Joint Radiological Emergency Response Plan; and
- to fulfill the requirements set out by the Nuclear Energy Agency of the Organization for Economic Cooperation and Development for the conduct of periodic nuclear emergency exercises under the INEX 2 series.

CANATEX-3 participants included, OPG, the Ontario Regional Municipality of Durham, the Province of Ontario, various ministries (lead: Emergency Measures Ontario, Ministry of the Solicitor General and Correctional Services), the Canadian Federal Government, various departments (lead: Health Canada), United States federal agencies (lead: Environmental Protection Agency (EPA)) and New York and Michigan States, various state agencies.

The CANATEX-3 exercise fulfilled Canada's commitment to hold an INEX 2 exercise in North America. INEX 2 is a series of International Nuclear Emergency Exercises sponsored by the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development. In this series of exercises, an "Accident-Host" country integrates the objectives and requirements of the INEX 2 series into a national exercise involving an accident at a nuclear facility that results in a release of radiation. In this case, Canada was the Accident-Host country, and combined the INEX 2 exercise requirements within the scope of the CANATEX-3 exercise for the pre-release, release and immediate post-release phases of a nuclear emergency.

The INEX 2 exercise objectives included in CANATEX-3 were:

- the real-time exchange of information, using actual communications hardware and software;
- decision making based on plant conditions and realistic, but limited data. Weather conditions in the accident scenario were those of the day of the exercise, and the World Meteorological Organisation participated in providing real-time information on local, regional and global weather trends; and
- development of an exercise website to post public information, including press releases, public briefings, media interactions and pressures, and the coordination of public information.

Participants for the INEX 2 aspect of the exercise included the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (NEA/OECD), the International Atomic Energy Agency (IAEA), the World Meteorological Organisation (WMO); the European Commission (EC); and various member countries of the OECD, including "neighbouring" countries (i.e.: the United-States, France (St. Pierre et Miquelon), and Denmark (Greenland)) and over thirty "far-field" countries, which simultaneously exercised portions of their nuclear emergency plans and relevant international emergency notification and communication procedures.

The exercise demonstrated the FNEP, and the FNEP and the Canada United States Joint Radiological Emergency Response Plan are generally adequate as the basis for a federal government response to a nuclear emergency in, or near, Canada. Work is on-going to achieve further improvements where required in the areas of personnel requirements, training, facilities (including communications and infomatics) information management, public communication and notification and activation.

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

Siting was fully discussed in Canada's report to the first Convention on Nuclear Safety. Article 17 from Canada's previous report is included here as Annex 17.1.

The national laws and regulatory requirements are detailed in Article 7.2. The licensing process is described in Article 7.3. The requirements for siting of a new power station are detailed in the General Nuclear Safety and Control Regulations and the Class I Nuclear Facilities Regulations.

ARTICLE 18
Design and Construction

18.1 A Description of the Licensing Process, Including a Summary of the National Laws, Regulations and Requirements Relating to the Design and Construction of Nuclear Power Stations

Design and Construction was fully discussed in Canada's report to the first Convention on Nuclear Safety. Article 18 from Canada's previous report is included in this report as Annex 18.1.

The national laws and regulatory requirements are detailed in Article 7.2. The licensing process is described in Article 7.3. The requirements for design and construction of a new power station are detailed in the *General Nuclear Safety and Control Regulations* and the *Class I Nuclear Facilities Regulations*.

ARTICLE 19
Operation

19.1 A Description of the Licensing Process, Including a Summary of the 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. 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. The operating licence renewals and the licensing requirements for continued operation are described in Article 7.4.1.

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. It is given that all regulatory requirements regarding siting, design and construction, as outlined in Articles 7, 17 and 18, have been met.

As stated under Article 7.3, before issuing an operating licence, the Canadian Nuclear Safety Commission must be assured that:

- the construction of the station conforms to the design submitted and approved;
- the safety analysis is complete; and
- the plans for operation are satisfactory.

Before a station is commissioned, several CNSC staff members are located at the station to set requirements for, observe, and report on the commissioning and start-up processes.

The CNSC staff does not attempt to participate in all aspects of the licensee's commissioning program. Reliance must be 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 program is checking the right items; and
- 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 results be documented and that any failure to meet the acceptance criteria be referred back to the design organization which decides what, if any, design changes are required. This allows the CNSC staff to do audits, at any time, to confirm:

- that the procedural requirements are being complied with; and
- that the decisions made are appropriate.

Direct involvement of CNSC staff 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. CNSC 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 (and compared to the rate assumed in safety analyses).

In other cases, a complete test is not practical, so 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 could not be justified.

The CNSC 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 CNSC staff's involvement with these commissioning tests includes reviews of the test proposals including the detailed commissioning specifications, which 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, CNSC staff reviews the test results and the commissioning reports produced.

The CNSC 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; and
- 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 CNSC site office staff attends some of these meetings. The CNSC 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, including 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).

The 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 (e.g., setpoint and other parameters limits, availability requirements);
- requirements on process systems (e.g., parameter limits, testing and surveillance principles and specifications, performance requirements under abnormal conditions); and
- 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;
- safety analysis; and
- other types of analysis (e.g. 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 CNSC. 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 and human error.

Safety analyses are performed to predict the consequences of the design-basis accidents, and compares them to regulatory documents (e.g., “Requirements for the Safety Analysis of CANDU Nuclear Power Plants”, CNSC Consultative Document C-6, 1980). 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);
- Operating Manuals; and
- the 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.

A major systematic review of each special safety system has been conducted at the Point Lepreau station to determine if the design, operation, and analysis of the station are consistent. The project is called

Determination Of Allowable operating envelope, and serves also to determine station parameters that define the allowable SOE. A 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; and
- 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 CNSC contain a series of conditions that are designed to ensure that a licensee conforms to CNSC requirements CNSC 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 as Annex 7.5 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 CNSC. 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; and
- the specific numerical limits for operating parameters that must be maintained to ensure 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 CNSC 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 CNSC. 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.

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. All power reactor operating licenses contain conditions that specifiy the requirements for a station-specific maintenance program.

Safety system testing is required on a regular basis to demonstrate that each safety function is operating correctly. The CNSC requirement is that each system must have an availability factor of 99.9%. 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 CNSC 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; and
- 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); and
- 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; and
- 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 and 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 OPG, there are three categories of procedures within the Abnormal Incident Manual:

- Abnormal State of Safety System Procedures;
- Emergency Operating Procedures; and
- 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 protect the safety of the operators and the general public in the event of a significant radiation incident. These procedures:

- direct event classification and categorization;
- make provisions for off-site notification; and
- 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 must be 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), OPG and many other Canadian companies to obtain services in many fields including research, engineering, analysis, assessment, maintenance, inspections and 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.

19.2.6 Reporting Incidents Significant to Safety

Incidents significant to safety must be reported in a timely manner by the holder of the relevant licence to the regulatory body. As discussed in Article 9.2 and Article 14, the reporting requirements in Canada are described in Regulatory Document R-99. Each licensee understands the importance of keeping the CNSC 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 ensure that the following facility reports are submitted to the CNSC 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; and
- Reliability Reports.

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. Reports submitted to the CNSC are reviewed to ensure that appropriate actions are taken. In many instances, the reports indicate that the licensee has acted appropriately and has initiated remedial measures.

In each case, the licensee is required to make an oral event report to the CNSC 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 CNSC, 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 CNSC has been conducting audits in nuclear power stations and utility corporate offices to make sure that the existing feedback systems achieve their objectives

There are also international obligations that have to be met by the CNSC. 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 the Incident Reporting Systems (IRSs), operated by both 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 CNSC 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. The Canadian INES coordinator at the CNSC is responsible for the coordination, the exchange of reports and the classification of the event severity in Canada.

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 (refer to Article 9.2 and 19.2.6 for more information on these reports). They provide information on undesirable events that are considered significant in the operation of nuclear generating units and related facilities.

Other reports include the licensees' quarterly reports, in-service reports and internal audit reports. On the regulatory side, the CNSC issues Audit Reports on operations in nuclear power stations. These reports contain the CNSC 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 CNSC to all licensees in Canada. When received by the licensees, these reports become part of their feedback systems. They should be reviewed for implications on their power stations.

CHANNELS OF FEEDBACK

There are a number of feedback channels/systems both within the utilities and the CNSC. Feedback systems at the utilities are aimed at improving both the reliable production of electricity and nuclear power station safety, while the feedback system in the CNSC is concerned with the safety of the workers, the public and the environment.

Channels within the utilities

Utilities have developed feedback systems to integrate operating experience into all aspects of station operation and management. For example, NB Power has developed the Problem Identification and Corrective Action (PICA) system and OPG has an OPEX site that incorporates Station Condition Records (SCRs), and operating experience from World Association of Nuclear Operator (WANO), Institute of Nuclear Power Operator (INPO) and CANDU Owners group (COG) sites. Similar systems exist at other Canadian utilities.

The CANDU Owners Group (COG) program involves exchange of information on operating experience between CANDU stations. Weekly OPEX screening meetings are held (via teleconference) amongst Canadian COG members and the results are provided to member utilities to determine if they are susceptible to a similar occurrence. Information is normally transmitted via the CANDU Network system.

AECL has had a feedback process for a number of years to ensure that construction, commissioning and operating experience is fed back to new designs; and that information found by AECL during the design of new plants is communicated to CANDU owners.

Channels within the CNSC

The Regulatory Event Assessment Program (REAP) was set up by the CNSC in 1998. It is a unified approach to the evaluation of events occurring at nuclear power stations and as well other nuclear facilities. Staff members from a number of specialist groups participate in the related assessments. The objectives of the system are to conduct a 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 at the CNSC. There are currently over 18,000 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 trending of certain safetyindicators and to produce periodic reviews, 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

Problems or issues that arise from event reviews that may be applicable to other stations are identified and brought to the attention of the CNSC site project officers and different specialist groups in the CNSC. They use this information to determine the appropriate course of action and assess the licensee's submissions regarding the particular event.

CNSC staff 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 CNSC site project officers review the status of corrective actions to make sure that they are completed expeditiously.

CNSC audit teams consult the operating experience in the CNSC database in planning strategies for their audits and in identifying problem areas in operation or maintenance such as procedural noncompliance, procedural deficiencies and use of nonstandard components.

Similarly, assessments conducted by CNSC specialists often utilise the operating experience recorded in the CNSC database.

19.2.8 Radioactive Waste Management

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, conditioning and long-term storage of the waste as disposal facilities are not yet available.

RESPONSIBILITY

The federal government has established a policy framework for the efficient and effective management and disposal of all radioactive wastes. 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 and cannot be processed. With the current techniques available in Canada, the radioactive waste can be reduced by about 60% of its volume prior to storage.

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.

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. 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

Considerable effort has been applied to the development of the deep geological disposal concept for nuclear fuel waste. This concept was considered by the Nuclear Fuel Waste Management and Disposal Concept Environmental Assessment Panel. In March 1998, this Panel made recommendations on the next steps, including a recommendation that consideration be given to other options for the long-term management of nuclear fuel waste in Canada.

Draft legislation has been introduced in Parliament that would lead to the selection of an option for the long-term management of nuclear fuel waste (refer to Article 7.2.5). The draft legislation also provides for funds to be set aside to implement the selected option.

ANNEXES

ANNEX 1.1
Research and Development Programs in Canada

The majority of nuclear research and development (R&D) in Canada is performed by Atomic Energy of Canada Limited (AECL), with smaller nuclear R&D programs conducted by Stern Laboratories, Kinetrics and other private companies and Canadian universities. The principal objective of the R&D program is to support the safety, licensing and design basis for Canadian nuclear installations. Building on this knowledge base, the Canadian nuclear industry invests in the applied R&D required to develop products and services for its lines of business. The knowledge base also provides the capabilities required to conduct applied R&D on a commercial basis to address issues specific to a particular nuclear facility. Lastly, the basic knowledge is required to support Canadian government policy, regulation and licensing, and international agreements.

The safety, licensing, and design basis for nuclear facilities covers those areas of R&D that impact on the radiological safety of the public, nuclear facility workers, and the environment. It covers the basic knowledge of materials, physics, chemistry, critical components, radiation, and the environment that could impact on safety. In many cases, the knowledge is articulated in mathematical models that are incorporated into computer codes to describe complex systems. R&D is used to keep the knowledge base up to date, to validate the codes and to ensure that the facilities are in place to advance the knowledge.

AECL has R&D activities in each of the key CANDU technology areas. These activities ensure that the basic science and engineering underlying each technology area is understood, and the knowledge base grows, as necessary, to address emerging issues. For example, not only must the factors affecting the performance of a component or system be understood, but the effects of aging (time, environmental exposure) must also be accounted for. This fundamental understanding is embodied in component specifications, systems operating guidelines, engineering and analysis tools, and computer programs to simulate reactor systems behaviour.

CANDU utilities fund R&D activities of direct interest to preserving the safe and effective operation of their reactors through the cost-shared programs of the CANDU Owners Group (COG). COG's R&D program is co-funded by the domestic CANDU utilities (Ontario Power Generation, Bruce Power Inc., Hydro-Québec, and New Brunswick Power) and AECL. Most of the COG-funded R&D is carried out by AECL, and the research is complementary to AECL's internally-funded R&D, providing support for the safety, licensing and design basis. A portion of the COG-funded R&D work is also carried out at Stern Laboratories, Kinetrics, other private companies and Canadian Universities.

The R&D activities supporting the safety, licensing and design basis in each of the main technology areas are described in more detail below.

SAFETY TECHNOLOGY

As its name implies, Safety Technology R&D is aimed at ensuring CANDU reactors and Canadian nuclear facilities continue to meet high safety standards. The basis is a fundamental understanding of the primary system thermalhydraulics, reactor core and fuel channel thermomechanical behaviour, fuel and fission product behaviour and containment performance. Key facilities include the NRU research reactor and hot cells for studying fuel, fission product and integrated systems behaviour, scaled thermalhydraulic loops for the primary and moderator systems, full-scale facilities for investigating high-temperature channel behaviour, hydrogen combustion facilities, and scaled representations for studying containment thermalhydraulics. The fundamental knowledge gained from Safety Technology R&D is built into models and computer codes that can be used to assess the consequences of a postulated accident sequence. In addition, generic safety issues are addressed, licensing methodologies are developed, and technologies are developed to mitigate accident consequences. Safety Technology R&D provides the technology required for regulation of nuclear installations in the form of basic understanding of phenomena and associated models and computer programs.

The current thrusts for Safety Technology R&D are in the areas of component and system performance under postulated accident conditions (fuel channel behaviour, system thermalhydraulics, fuel and fission product behaviour, and containment performance) and methodology development. An important initiative is also underway to qualify a suite of safety and licensing computer programs in cooperation with Canadian CANDU utilities.

Fuel channel R&D addresses the thermal, chemical and mechanical behaviour of CANDU fuel channels under high temperature conditions that would be typical of a loss-of-coolant accident (LOCA). Research underway is aimed at quantifying heat transfer from fuel channels to the moderator to confirm the effectiveness of the moderator as a secondary heat sink. Experiments are also in progress to verify predictions of the behaviour of molten fuel injected into the moderator at high pressure following a single-channel flow-blockage event.

System thermalhydraulics provides the experimental data required to understand and model the performance of a CANDU primary heat transport system under postulated upset and accident conditions. A current focus is the quantification of channel voiding rates during a large-LOCA accident sequence.

Research on fuel and fission products ensures that the thermal, chemical and mechanical behaviour of CANDU fuel under high temperature LOCA conditions is well understood, and the thermodynamic and kinetic behaviour of any fission products released if the fuel sheath is breached can be predicted. Currently, data from a series of integrated in-reactor fuel and fission product experiments is being analysed and used to validate associated computer models, and the retention of fission product aerosols in the primary heat transport system is being investigated.

Research is also conducted into the phenomena that might affect the ability of a reactor containment building to contain the consequences of postulated accident sequences, including containment thermalhydraulics, hydrogen combustion and the behaviour of gaseous and aerosol fission products. Work is underway to quantify fission product retention in containment leak paths, and to incorporate our understanding of hydrogen, aerosol and iodine behaviour into computer programs that model containment behaviour.

Two important aspects considered under methodology development are severe accident R&D, and development of a best-estimate plus uncertainty analysis methodology. Under the former, experiments are conducted to investigate key severe accident phenomena (e.g. core-disassembly mechanisms) and to develop severe accident models and methodology for CANDUs. For the latter, a methodology is being developed to take advantage of the qualification exercise currently underway for key safety and licensing computer programs (see below), and allow a better determination of the safety margins associated with the operation of CANDU reactors. The current methodology is based on a limit-operating envelope approach that can underestimate the actual margins that are available through the use of a series of bounding assumptions.

The cooperative initiative on computer program qualification is ensuring that the computer programs meet the recently developed Canadian Standards Association (CSA) standards for Software Quality Assurance, and that the uncertainties associated with application of these computer programs are quantified. This suite of computer programs will greatly enhance the safety, licensing and design basis of

CANDU reactors, and can be used in support of operating stations as well as licensing current generation designs.

REACTOR CORE TECHNOLOGY

Reactor Core Technology is central to nuclear reactor R&D. Knowledge of the underlying reactor and radiation physics, fuel technology, and fuel thermalhydraulics are required to design, build and operate reactors, both power reactors (CANDUs) and small reactors (MAPLEs). Key facilities include the NRU research reactor for studying fuel behaviour, metallurgical hot cells for fuel examinations, scaled thermalhydraulic loops, and the ZED-2 critical facility for studying reactor physics. The basic knowledge in reactor core technology can be applied to developing advanced fuels and fuel cycles that improve reactor operation, minimize waste and promote the efficient use of uranium resources.

Research on reactor core technology can be divided into four main areas: reactor and radiation physics, fuel technology, fuel channel thermalhydraulics, and advanced fuel designs. Since the computer programs that are developed in support of these essential disciplines are important for safety and licensing of CANDU reactors, they are included in the cooperative code qualification activity that was described under Safety Technology.

Reactor and radiation physics provides the fundamental understanding of neutron physics and radiation behaviour required to safely operate a reactor, including ensuring the nuclear reactions are well-controlled, and ensuring adequate shielding is provided. One of the current thrusts is to reduce the uncertainties in calculating the energy deposition that occurs upon voiding of a fuel channel during a LOCA.

Under fuel technology, R&D is conducted to ensure the continued excellent performance of current generation CANDU fuel. Activities include surveillance examinations of fuel removed from power reactors, investigation of the performance of CANDU fuel under load-following conditions, and investigation of fuel performance at higher burnups.

A key element of fuel design is heat transfer to the primary heat transport system. In the case of a CANDU reactor, detailed experiments and modeling are conducted for the thermalhydraulics of the fuel channel. An issue of current interest is fuel performance under conditions where there may be limited periods of poor heat transfer.

In the area of advanced fuel designs, AECL is developing new fuel bundle designs that improve the operating and safety margins of CANDU reactors. The base for these designs is the new CANFLEX bundle, which allows a combination of higher fuel burnup, higher channel powers, and increased operating margins. AECL is also exploring fuel options that can improve utilization of uranium resources.

FUEL CHANNELS

A key difference between CANDUs and LWRs is the use of an array of fuel channels to circulate coolant past the fuel rather than containing the fuel within a large pressure vessel. Each fuel channel consists of a zirconium-alloy pressure tube, with a steel end fitting at each end to connect to out-of-core piping, and separated from the heavy water moderator by a larger calandria tube. The use of fuel channels requires an in-depth understanding of the behaviour of the channel materials, including their fracture and deformation properties, the effects of radiation, and the effects of environmental degradation (corrosion and hydrogen ingress). In addition, techniques are required for non-destructively assessing the health of fuel channel materials. Key facilities include corrosion test loops, analytical microscopes and instruments, high-flux research reactors, metallurgical hot cells, and materials testing laboratories. The fundamental knowledge gained from Fuel Channels R&D is used to provide component specifications, operating guidelines and fitness-for-service guidelines for fuel channel components. Fuel Channel R&D is focussed on the areas of pressure tube fracture mechanisms, fuel channel deformation, pressure tube corrosion and inspection technology.

Zirconium alloys can be susceptible to an environmentally assisted cracking mechanism, delayed hydride cracking (DHC). R&D is conducted to ensure that pressure tube material properties, fracture properties and DHC properties, and the effects of exposure to the temperature, pressure and neutron fluence in a reactor, are understood. Experiments are underway in high-flux research reactors to ensure these properties are known to the projected end-of-life. Other experiments are underway to determine the effects of in-service flaws (e.g. debris frets) on pressure tube fitness for service.

Exposure to the high temperatures, pressures and neutron fluence in a reactor causes fuel channels to deform over time (elongation, sag, diametral expansion). R&D is conducted to ensure the underlying mechanisms are understood, and the deformation of a fuel channel can be predicted. A current focus is ensuring accurate predictions of diametral expansion, and the effects of circumferential variations in temperature and flux on local expansion. Being able to predict changes in pressure tube diameter is important to ensuring that the coolant flow rates through the fuel bundles are known with a high degree of certainty.

As pressure tubes oxidize with exposure to the primary coolant, they slowly pick up deuterium (an isotope of hydrogen). High deuterium concentrations are one factor required for the initiation and growth of DHC cracks. Experiments are underway in research reactors to ensure that the rate of deuterium pickup can be predicted with confidence. Improvements are also being made to models for oxidation and deuterium pick-up.

Fuel channels require periodic inspections to meet regulatory and operational requirements. R&D is being conducted to develop improved methods for non-destructively determining deuterium concentrations and changes in pressure tube shape. Improvements are also being made to the detection and characterization of pressure tube flaws that form during service.

COMPONENTS AND SYSTEMS

This area comprises R&D on out-of-core components and systems, notably primary and secondary side heat transport systems, moderator systems, feeders, steam generators, pumps, valves and seals. Key facilities include the NRU research reactor, corrosion test loops, activity transport loops, scaled thermalhydraulics loops, and component test beds. A basic understanding of reactor chemistry and materials behaviour is applied to developing technology to ensure component and system life. In-service inspection technology is developed to assess the health of components and systems, and improved components are developed.

Currently, R&D efforts in the Components & Systems area are focussed on understanding steam generator and heat exchanger performance and safety margins; the development of non-destructive testing technology, such as remote probes for feeder thickness measurements and for detecting defects in piping; improved seals for existing and advanced CANDU designs; and improvements to the emergency core cooling system.

In addition to component development, advances are being introduced into the methodology used for heat and mass transfer in CANDU components, for the description of vibration and fretting of components including fuel, for the transport of radioactivity in CANDU systems, and for the various mechanisms affecting the performance and lifetime of components due to chemical and mechanical degradation.

New processes are also being developed for maintenance of CANDU plants. These include methods for chemical and physical cleaning of various circuits in CANDU reactors, and the application of AECL's systems codes to plant operations. A particular focus is the development of system health monitors. One example is ChemAND (Chemistry ANalysis and Diagnostic system), which tracks system health based on on-line analysis of current and historical plant chemistry data, as well as monitoring safety-related chemistry parameters.

CONTROL AND INFORMATION TECHNOLOGY

CANDU reactors were one of the first to make extensive use of digital control systems. Real-time control, display and protection systems are an integral part of a CANDU control room, and integrated information management software is designed to facilitate station operation and enhance performance. This technology is being upgraded and extended to improve performance and safety. A current focus is the development of improved systems for monitoring and controlling reactors that can be retrofitted to existing reactors, for example as part of a refurbishment exercise.

HEAVY WATER AND TRITIUM TECHNOLOGY

One of the essential features of a CANDU reactor is the use of heavy water as an efficient moderator for neutrons. AECL is developing technologies for the production of heavy water and for heavy water management in CANDU plants, including the recycling of used heavy water, based on a proprietary wetproofed catalyst originally developed from our underlying chemistry program. The catalyst effects rapid exchange of hydrogen isotopes between chemical species, which can be used to concentrate deuterium in water, to upgrade heavy water by extracting H, and to detritiate heavy water (i.e. remove the hydrogen isotope tritium which poses a radiological concern). Technology is also developed for the management, handling and storage of tritium. Key facilities are various scaled demonstrations of heavy-water production and detritiation processes.

Heavy water management in a CANDU reactor includes upgrading (exchanging hydrogen for deuterium in the heavy water) and tritium extraction after a plant has operated for many years. Currently, heavy water recovered from the vapour recovery dryers and collection systems, or taken from the moderator and heat transport systems, is upgraded using large water distillation columns. AECL is developing a CECE (Combined Electrolysis Catalytic Exchange)-based technology for upgrading that will be considerably less expensive.

ENVIRONMENT, EMISSIONS AND WASTE MANAGEMENT

There has been a decreasing trend in the radiation doses associated with all reactor designs during the past decade. AECL is following a methodology for dose reduction that includes measurements at existing stations, examination of operational practices and data, development of improved technologies for measurement and mitigation, and rigorous review of CANDU designs to ensure that full advantage is being taken of our knowledge base on radionuclide behaviour. Targets have been adopted that include reducing the buildup of activation products, tritium and heavy water management processes that reduce tritium emissions, and improved waste management developments to reduce emissions during waste handling. As well, improved methods for the characterization of environmental pathways are being developed. The focus is on nuclides such as tritium and carbon-14, and on extending our knowledge base of the effects of these nuclides on non-human biota.

HEALTH PHYSICS AND RADIATION PROTECTION

Historically, AECL has studied the more fundamental aspects of radiation and health to ensure a sound basis for health protection programs and regulation. This R&D includes applied work in health physics and basic scientific programs to elucidate mechanisms underlying health effects of ionizing radiation at low doses and low dose rates that are of interest in occupational exposures and from routine emissions. The Biological Research Facility, used for small-mammal studies, is central to much of the research. Efforts are underway with Health Canada to establish a more broadly-based National Centre for Radiological Sciences.

Nuclear Engineering Research in Canadian Universities

Nuclear engineering research in is carried out in a number of universities throughout Canada. A brief description of the programs is provided here.

McMASTER UNIVERSITY

McMaster University has the largest university-based Nuclear Engineering program in Canada. The program is facilitated by the university's own research reactor, the McMaster Nuclear Reactor (MNR). Situated on campus, the MNR is a 5MW plate-type swimming pool reactor with irradiation, remote handling, neutron radiography, and expansive beamtube facilities. Research is being followed in the areas of reactor physics, fusion technologies, thermalhydraulic, computer simulation and computer networks. The university's facilities are complemented by joint research with Atomic Energy of Canada's Chalk River Laboratories and Sheridan Park branches.

UNIVERSITY OF TORONTO

The University of Toronto has a diverse range of research programs in the field of nuclear technology, including a study of the behaviour of fission products following reactor accidents, and the application of gamma irradiation to the elimination of organic compounds. The university also supports research into the nuclear-related disciplines of two-phase flow dynamics, radioanalysis and radioactivation, numerical simulation of hydrodynamics, heat transfer, and thermalhydraulics.

ROYAL MILITARY COLLEGE OF CANADA

The nuclear engineering group of the Royal Military College of Canada is involved in the experimentation and modeling of nuclear fuel behaviour during normal and reactor accident conditions. Of particular emphasis is the modeling of the source term in concern of the mechanisms of fission-product release from defective fuel or severely-damaged fuel. Fuel-failure monitoring techniques are being developed for normal reactor operation in both research and commercial (CANDU and LWR) power reactors. New radiation detection techniques are also being studied to determine their application in the measurement and characterization of terrestrial and space radiation.

ECOLE POLYTECHNIQUE DE MONTREAL

The Ecole Polytechnique de Montreal has been active in the field of computational reactor physics for more than 10 years. Among its research institutions are the Hydro-Québec Chair in Nuclear Engineering, which specializes in neutron transport, including the development of new numerical methods in transport and diffusion theory. The educational focus is on an in-depth understanding of neutron transport problems and the basic concepts of reactor design, as well as a practical knowledge of the modern methods of both static and time-dependent reactor analysis.

UNIVERSITY OF NEW BRUNSWICK

The nuclear program at the University of New Brunswick includes a Chair in Nuclear Engineering, under the Department of Chemical Engineering, as well as the Centre for Nuclear Energy Research (CNER). The CNER performs research and development work in the field of nuclear energy that is associated with the operation and maintenance of CANDU nuclear power stations, with expertise in corrosion science, chemical probe development and information and control systems. The Centre is staffed by a combination of persons associated with Atomic Energy of Canada (AECL), the University of New Brunswick and the New Brunswick Research and Productivity Council, and CNER employees.

ANNEX 1.2
A Brief Description of the CANDU Reactor

A brief description of the CANDU reactor and its features is presented here.

REACTOR

The reactor comprises a stainless steel horizontal cylinder (called 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.

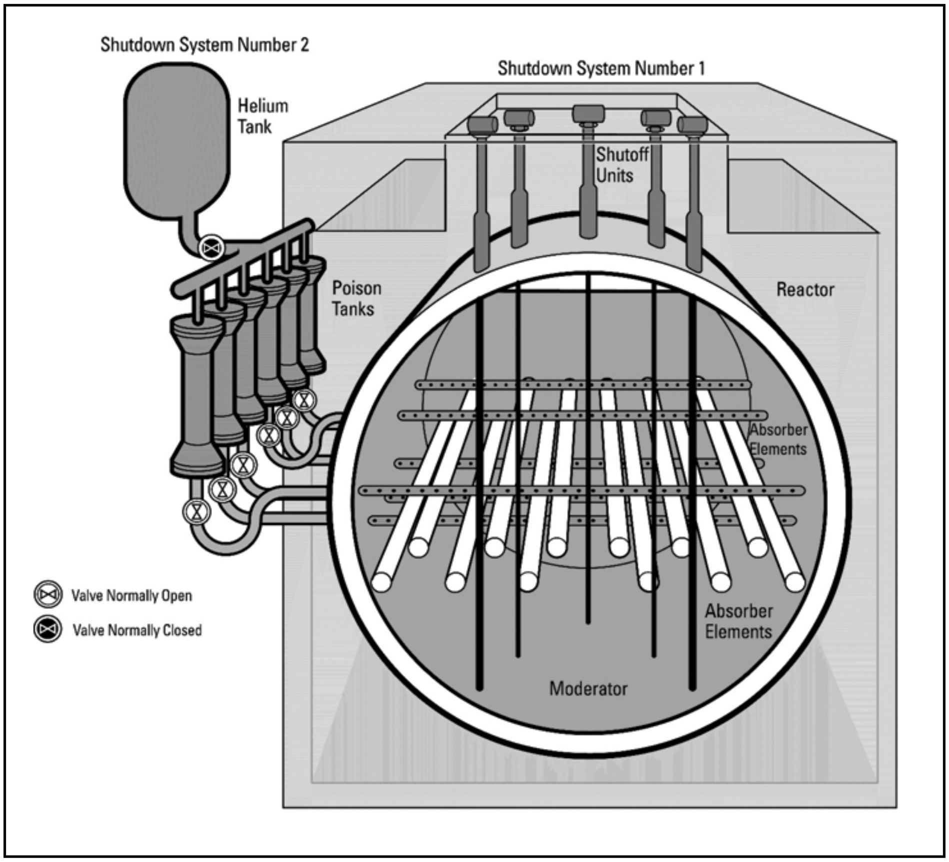
REACTOR REGULATING SYSTEM

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. Shutdown systems one and two are shown schematically in Figure A1.1. SDS1 uses shutoff rods while SDS2 uses poison injection.

FIGURE A1.1: CANDU Two Shutdown Systems



REACTOR ASSEMBLY

The CANDU reactor assembly, shown in Figure A1.2, includes several hundred channels contained in, and supported by the calandria. Figure A1.3 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 light 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.

FIGURE A1.2: CANDU Reactor Assembly

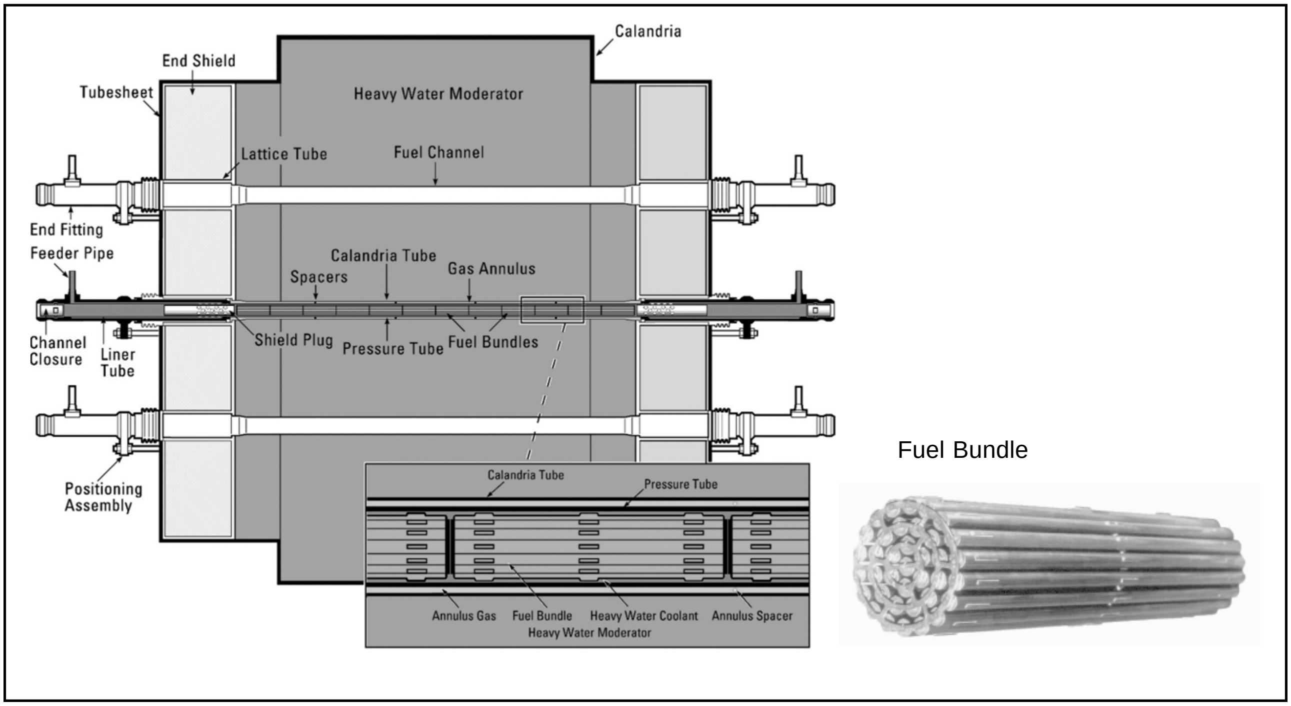
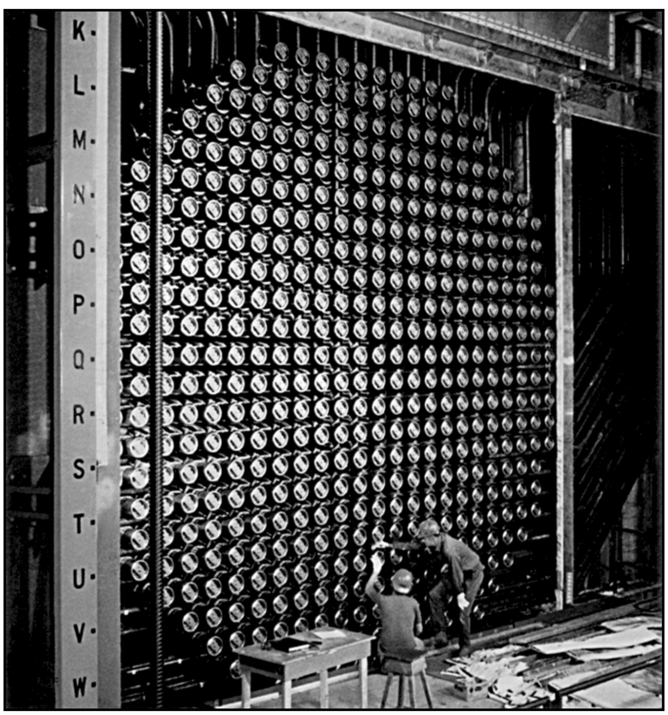


FIGURE A1.3: CANDU Reactor Face (during construction)



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

The fuel channels are also shown in Figure A1.2, 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; and
- 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 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 A1.2. 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.

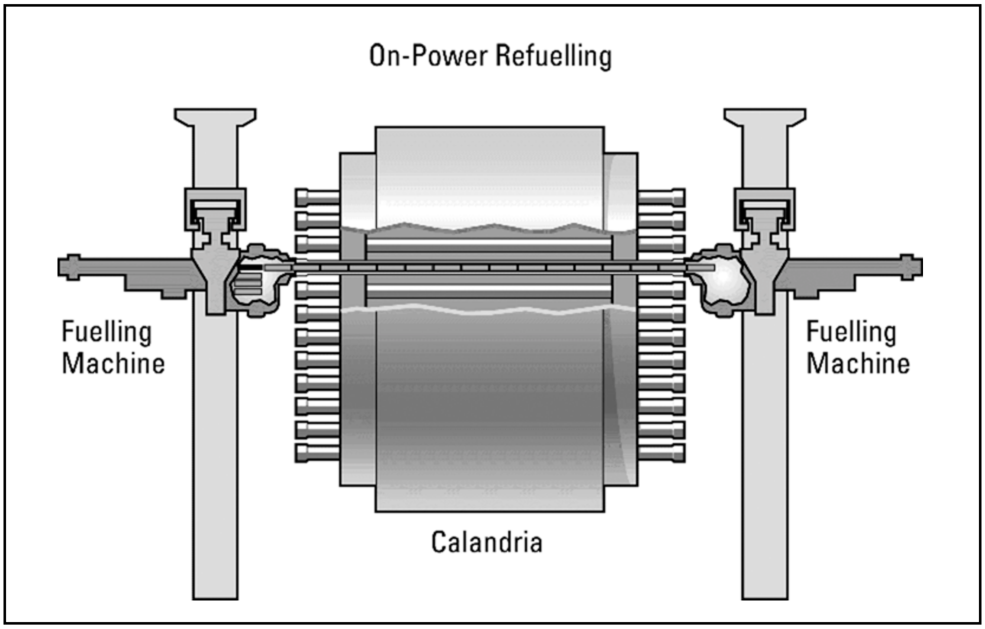
The fuel handling system:

- provides facilities for the storage and handling of new fuel;
- refuels the reactor remotely while it is operating at any level of power; and
- 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 A1.4 . New fuel bundles, from one fuelling machine, are inserted into a fuel channel in the same direction as the coolant flow 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 or the 12 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.

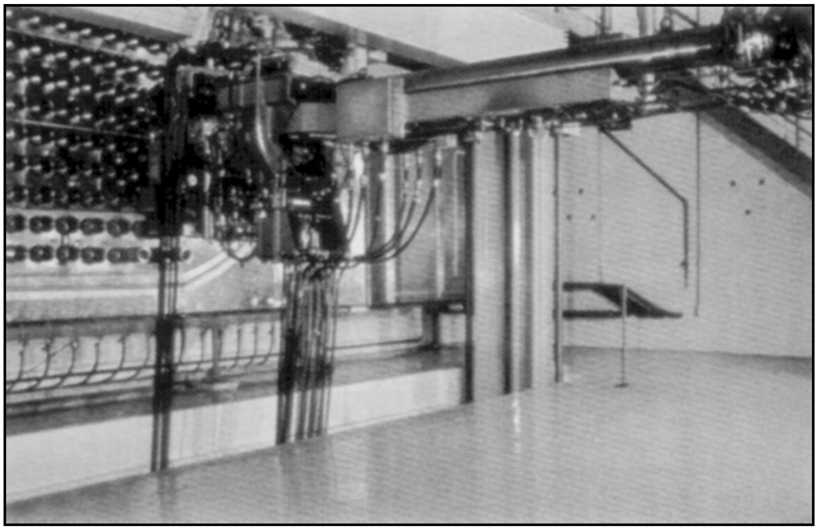
FIGURE A1.4: CANDU Power-On Fuelling



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.

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 A1.5.

FIGURE A1.5: Fuelling Machine in Operating Position at Face of Reactor



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.

ANNEX 6.1
CNSC Generic Action Items

GAIs are regulatory tools used by the CNSC to pursue concerns that may affect more than one station. The GAIs are used to monitor the progress of licensees with regards to safety issues and to provide a basis for communication of licensee progress. Many items have been raised over the years; some of them have been closed and others are still open. The following are brief descriptions of both kinds.

A. Open Generic Action

Generic Action Item 88G02
"Hydrogen Behaviour in CANDU Nuclear Generating Stations"

1. Safety Issue

The hydrogen released due to high temperature interaction between zirconium alloys and steam during certain Design Basis Accidents (DBAs) may produce flammable gas mixtures in some regions of containment, in the short term. Flammable gas mixtures may also develop due to water/steam radiolysis and metal corrosion in various containment regions such as the sump regions and the calandria, over time. The mechanical and thermal loads generated by potential ignition of these gas mixtures may then threaten the integrity and functions of the containment envelope, critical internal structures and necessary safety-related system components and instrumentation.

To mitigate the combustion behaviour of the short term releases in the multi-unit NGSs, Ontario Power Generation (OPG) has installed Hydrogen Ignition Systems (HISs) to remove the hydrogen via benign burns initiated by a number of igniters. Assessments of the short term behaviour of hydrogen with the 3-D/1-D hybrid code GOTHIC were then carried out to demonstrate the effectiveness of the HISs. Based on a derived safe load criterion, OPG has concluded that all potential hydrogen burns in its NGSs are benign, even for impaired or ineffective HISs and Local Air Cooling units (LACs).

No short term mitigating measures have been installed in the single unit NGSs. It was postulated that dilution via mixing by natural circulation can mitigate the short term hydrogen threat. Scoping analyses with the 1-D code PRESCON2 to identify the potentially limiting cases, have been completed. Assessments with the 3-D/1-D hybrid GOTHIC code to determine the bounding gas mixtures and combustion modes in certain critical containment regions, have been carried out, as well.

No analysis of the threat posed by the hydrogen released over the long term in multi and single unit NGSs, have been carried out yet because the utilities were of the opinion that effective mitigation was provided by mixing via natural circulation or by venting via the Filtered Air Discharge System (EFADS).

As requested by CNSC staff, the utilities plan to demonstrate the effectiveness of current and planned mitigation measures including Passive Auto-Catalytic Hydrogen Recombiners (PARs) in the short and long terms.

2. Closure Criteria

Licensees are required to:

- Consider the entire spectrum of design basis accidents including loss of coolant accident plus loss of emergency core cooling (LOCA + LOECC) and continue to use the methodology that maximizes the hydrogen and fission product source terms (limiting steam flows to the fuel channels), when evaluating the short-term and long-term hydrogen releases for each NGS.

- Carry out conservative, comprehensive and limiting assessments of mixing and transport for each NGS to:

  - determine the local hydrogen distributions in critical containment regions via 3-D assessments, crediting any hydrogen mitigating measures which are in place or planned; and
  - identify the envelope of hydrogen combustion and its consequences:

    - demonstrate that all local pockets of sensitive gas mixtures (combustible clouds) cannot lead upon ignition to DDTs and fast deflagrations with potentially unacceptable combustion loads in any region of containment;

    - demonstrate via well-supported calculations and/or experiments that gas mixtures of composition outside the DDT and fast deflagration envelope (i.e., in the slow deflagration domain), do not have, if ignited, consequences detrimental to the containment boundary, supporting internal structures and required safety-related equipment, taking into account turbulence and flame acceleration generated by obstacles, in particular;

    - demonstrate via well-supported calculations and/or experiments that potential standing flames do not threaten the survivability and/or function of vulnerable containment boundaries, critical internal structures and essential post-LOCA equipment;

    - ensure the survivability and/or functions of threatened essential post-LOCA equipment, vulnerable containment boundaries and critical internal structures, if any, via environmental qualification and/or augmented protection and mitigation; and

    - demonstrate, unless calandria refilling is assured, that, for LOCAs + LOECC with consequential moderator drain, the releases of hydrogen within the calandria, do not threaten the integrity of the calandria and that of necessary system components and monitoring instrumentation within it or next to it.

- Evaluate PARs for their potential to enhance the effectiveness of the short-term and long-term mitigating measures, and reduce the risk from potential hydrogen burns.

3.0 Progress to Date

Utilities have agreed to install PARS after assessing their effectiveness. Progress has been made in the following areas: analysis of LOCA + LOECC, environmental qualification of essential equipment and the validation of the computer code GOTHIC. Closure of this GAI is expected by the end of 2002.

Generic Action Item 90G02
"Core Cooling in the Absence of Forced Flow (CCAFF)"

1.0 Safety Issue

In some postulated accident scenarios, the primary heat transport pumps are assumed to fail. In this case, removal of residual heat relies on natural circulation of the coolant. Although natural circulation with the primary heat transport system full was shown to be effective, some partial inventory natural circulation experiments done at the Whiteshell RD-14M test facility have shown degraded cooling in some channels.

At issue are these unexpected results, which were observed in two phase natural circulation (thermosyphoning) experiments in the RD-14M test facility. RD-14M is a scaled representation of the major components of a CANDU reactor primary and secondary heat transport systems. A fraction of these tests resulted in fuel element simulator heatup at unexpectedly high coolant inventory.

It should also be noted that this GAI addresses only two phase thermosyphoning following faults from full power. Any related issues of faults at shutdown, e.g. loss of shutdown cooling, are to be tracked as station specific action items and are not discussed here.

2.0 Closure Criteria

In order to achieve closure, licensees are required to demonstrate, to the satisfaction of the CNSC, that experimental results do not invalidate current safety analyses that credit core cooling without forced flow. Alternatively, licensees may review the safety analyses with the new knowledge acquired from the experiments and, where necessary, implement adequate design modifications.

3.0 Progress to Date

Computer code validation work, and better modelling of the reactor and the experimental facility, are being done. One utility has implemented a design modification to maintain coolant inventory from falling to levels where channel heatup is possible. This has been achieved by maintaining D_2O feed during loop isolation that follows large LOCA. Closure of this GAI is expected by the end of March 2003.

Generic Action Item 91G01
"Post-Accident Filter Effectiveness"

1.0 Safety Issue

Following a loss of coolant accident the emergency filtered air discharge system (EFADS) in multi-unit stations would be used to maintain containment pressure below the atmospheric level in the long term. The EFADS filters are relied upon to limit radioactive releases during the venting of containment. The single-unit stations do not depend on filtered air discharge systems to ensure the effectiveness of their containment systems. However, filtered venting at these stations may be considered as a long term post-accident management option. If during venting the filters do not perform as assumed this would result in a higher than expected public hazard. It is essential, therefore, that all licensees have programs to support the credited effectiveness of filters and to ensure prompt detection of any deficiency which could prevent the filters from performing as designed.

2.0 Closure Criteria

To achieve closure of this GAI, licensees are expected to have effective programs in place which will provide a continued assurance that under all credible post-accident conditions filter effectiveness will match or exceed that credited in safety analyses. These programs may, as necessary, consist of laboratory and in-situ tests, research and analytical efforts, adequate QA, testing, maintenance, and operational procedures and should:

- ensure that there are appropriate tests for all filter elements;
- demonstrate that tests are performed under representative conditions;
- provide evidence confirming that filters are capable of performing as credited under harsh post-accident environmental conditions;
- provide an assurance that filter degradation can be reliably detected by existing tests;
- address consequences of hydrogen burns on filter performance;
- confirm that there are proper QA, maintenance and operation procedures in place
- justify frequency of tests;
- provide an assurance that instrumentation and control equipment is such that it will allow operation of filters as designed under all representative conditions;
- demonstrate the adequate availability of filters; and
- identify areas, if any, where continued design, experimental and/or analytical efforts are required.

3.0 Progress to Date

This GAI has been closed for one utility. The other utilities continue to implement their plans to demonstrate the adequacy of the filter testing program and the adequacy of procedures that ensure filter effectiveness. The expected closure date for this GAI is end of 2002.

Generic Action Item 91G02
"Operation with a Flux Tilt"

1.0 Safety Issue

The adequacy of Regional Overpower Protection (ROP) or Neutron Overpower Protection (NOP) trips for reactor operation with a flux tilt is demonstrated by analyses, which take into account different plant states for which continued operation is permitted.

ROP/NOP system design is based on information derived from simulations of certain reference and perturbed flux shapes in the reactor core. Trip setpoints are established from these simulations to prevent any channel reaching its critical power limit in case of a bulk loss of regulation. One key component in the analysis is the relationship between flux values at detector locations and channel powers for various flux shapes. The analyses assume that the ratios of changes in fluxes and channel powers due to perturbation, called simulation ratios, are invariant with respect to the reference flux shape. On this basis a limited number of combinations of credible perturbed flux shapes and an untitled reference flux shape is analyzed to derive trip setpoint values. Furthermore, these trip setpoints values provide coverage for different plant states. Differences in the reference flux shape used in the analyses and actual flux shapes are accounted for by regular detector calibration.

At issue is the adequacy of error allowances used to derive the trip setpoints values, since these can be sensitive to the accuracy of reference and perturbed flux shapes and associated simulation ratios. CNSC staff identified in 1991 an apparent deficiency in licensees' analyses and practices: while tilted-flux operations were not considered as initial conditions in ROP/NOP analyses, the reactors are permitted to operate with relatively large flux tilts. For initial flux shapes with relatively large tilts the invariance of simulation ratios needs to be demonstrated to support the adequacy of error allowances used to derive the trip setpoints values for all plant states for which continued operation is permitted. In order to demonstrate the adequacy of ROP/NOP trip setpoints, licensees were requested to:

- determine the maximum tilt permitted by the current operating procedures for prolonged operation with a flux tilt, prior to any operator's action;
- generate a steady state flux distribution, corresponding to the maximum tilt permitted by the current operating procedures, and design-basis and abnormal perturbation flux shapes, corresponding to this steady state shape; and
- assess simulation ratios (ratios of changes in fluxes and channel powers due to perturbations) for the above flux shapes, and assessment of the ROP/NOP trip coverage by determining whether the ratios are invariant within any postulated error allowance.

2.0 Closure Criteria

To achieve closure, licensees are required to perform the following:

- determine the adequacy of various error allowances and assumptions, such as the invariance of simulation ratios used in the analyses, to cover the actual maximum tilt permitted by the revised and improved operating procedures - there should be a specific error allowance for simulation ratio variation due to flux tilts in ROP/NOP error analysis, its magnitude should be sufficient to cover variations found in analyses that have been performed, as well as other cases which are not analyzed;
- determine the sensitivity of ROP/NOP analysis results to flux tilt definition; at issue is the fact that while the RRS flux tilt definition is based upon bulk regional averaged parameters, the ROP/NOP requirement is to prevent the critical power in any individual channel being reached; and
- determine the potential impact of factors not fully covered in current analyses on the effectiveness of the ROP systems; the main factors not completely covered are: the effect of transient xenon, boiling in a region of channels, and replacement of a failed detector.

3.0 Progress to Date

This GAI has been closed for two utilities. Submissions by the other utilities are under review by CNSC staff. Closure is expected shortly.

Generic Action Item 94G02
"Impact of Fuel Bundle Condition on Reactor Safety"

1.0 Safety Issue

The condition of certain fuel bundles irradiated in CANDU reactors has been observed to differ from that predicted and accounted for in design, operation, and safety analysis documentation. The fuel bundles in question have shown signs of more-than-expected degradation such as end plate cracking, spacer pad wear, element bowing, sheath wear, bearing pad wear, sheath strain, disappearance of CANLUB layer, oxidation of defective fuel, and fission product release.

Fuel bundle degradation depends on the reactor design, fuel channel, fuel design, fuel manufacturer, and operating conditions. Since theoretical models have been unable to correlate these factors adequately to the fuel condition, fuel and pressure tube inspections are necessary. Owing to the number of factors upon which the degradation depends, the inspection program must be extended beyond inspection of defective fuel to observe these changes. Such expanded inspections, when done at Bruce A and B, showed that degradation has been occurring throughout the life of Bruce B.

Fuel bundle degradation is sometimes also accompanied by fretting and scratching of the pressure tube, these being superimposed onto other changes, such as creep.

The effect(s) of some of this bundle degradation on reactor safety is not known partially because of a lack of experiments and safety analysis methods to account even for changes to pressure tube geometry as the fuel channel ages. As such, the important fuel and fuel channel parameters to measure are not known.

Although some fuel inspections have been conducted and the results submitted to the CNSC, licensees do not have a formal process to ensure that the fuel and fuel channel conditions are identified and accounted for.

2.0 Closure Criteria

To achieve closure, licensees are required to perform the following:

- Implement an action plan to eliminate excessive fuel and channel degradation in acoustically active channels (where applicable).
- Implement an effective formal and systematic process for integrating fuel design, fuel and channel inspection (in situ), fuel and channel laboratory examination, research, operating limits and safety analysis. This process must have the following features:

    - annual review (by the licensee) to demonstrate effective implementation and adequate corrective actions taken for deficiencies identified in the review;
    - sufficient resources for each participating group (design, inspection, examination, research, safety analysis, and operation) to ensure that the fuel condition is known and accounted for adequately;
    - clearly defined maximum allowable limits, under normal operation, on fuel condition in terms of sheath strain, element bowing, wear (spacer pad, bearing pad, end plate), pressure tube scratching and wear, burnup and residence time; design documentation and pressure tube fitness-for-service guidelines should be updated accordingly;
    - a determination, for the full range of the operating envelope, the power boost sheath failure threshold for CANLUB fuel and the chemistry effects of CANLUB on centerline temperature and fission product release;
    - assurance that the safety analysis accounts for the allowable fuel condition when combined with aging effects such as pressure tube creep, the effect of CANLUB in the fuel, and any chemistry effects on temperature and fission product release, including a calculation of the number of sheath failures resulting from a bounded loss of power control; and
    - a surveillance program that demonstrates compliance with identified limits, e.g., detection of significant changes in fuel condition caused by changes in fuel fabrication and factors affecting acoustic resonance.

3.0 Progress to Date

This GAI has been closed for two utilities. Other utilities plan to demonstrate the adequacy of their fuel inspection program. The closure date has not been decided yet.

Generic Action Item 95G01
"Molten Fuel/Moderator Interaction"

1.0 Safety Issue

A single channel event, such as a severe flow blockage or a stagnation break of an inlet feeder, could reduce flow through the channel without detection by the reactor regulating or shutdown systems. If the flow blockage is severe (>90%), then the coolant in the channel will boil-off, and melting of cladding, spacers, and fuel would occur. The accident proceeds rapidly because the reactor remains at full power. This situation leads to an accident scenario where the pressure tube (and subsequently the calandria tube) may fail at high pressure (10 MPa), ejecting the melt into the moderator.

The safety reports for currently operating CANDU plants account for the damage that the reactor sustains from spontaneous rupture of a fuel channel. Of concern is whether molten fuel/moderator interaction in a loss of flow event can cause additional damage that prevents the shutdown of the reactor, trigger rupture of other fuel channels, or fail the calandria vessel.

2.0 Closure Criteria

This generic action item will be closed when, following the tests:

- the dominant mode of interaction can be determined, using the resulting pressure history and particle size distribution; or

- in the event that the experimental results are inconclusive, the safety margin or potential damage resulting from the interaction, regardless of its mode, is determined, using the measured pressure transient (signature) following the pressure tube rupture as the primary tool for the evaluation.

3.0 Progress to Date

Industry plans to construct a test rig are progressing as scheduled. It will take about 5 years to complete the tests and interpret the results. The industry has agreed that the resolution of this GAI will not eliminate the need to validate the relevant computer code. The closure date has not been decided yet.

Generic Action Item 95G02
"Pressure Tube Failure with Consequential Loss of Moderator"

1.0 Safety Issue

At issue are the consequences of rupture of a pressure tube in an operating CANDU reactor, and the interpretation of fuel channel burst tests conducted by the industry.

Rupture of a pressure tube in an operating reactor could result in one or more of the following:

- a loss-of-coolant accident (LOCA) inside or outside the reactor core;
- a breach of the moderator boundary leading to a loss of moderator heavy water (LOM); and
- damage to reactor systems, structures and components, including adjacent fuel channels, reactivity control mechanisms, the calandria, and ejection of fuel bundles into the calandria and/or the reactor vault.

Analyses of such events are presented in the Safety Reports for each plant. However, the Safety Reports neglect scenarios involving a LOCA and a large loss of moderator (LOCA/LOM). Despite the fact that LOCA/LOM events are neglected, the industry approach to analysis of failures with ECI unavailable has been to credit the moderator as the ultimate heat sink for the reactor.

Events that combine a LOCA with a loss of moderator would potentially invalidate the conclusions of the safety reports. OPG has stated that “the ejection of an end fitting, in conjunction with the calandria tube failure, would seriously threaten the capability of the moderator as a heat sink if ECI is not available, because the moderator loss rate would far exceed the available makeup rate”. The absence of an effective heat sink could lead to a severe accident in which the fuel attains high temperatures and releases large quantities of radioactivity into containment. Furthermore, the results of fuel channel burst tests conducted by the industry suggest that pressure tube rupture events leading to a large loss of moderator are more probable than previously assumed. Out of about a dozen burst tests, several pressure tubes have suffered a guillotine break.

It should also be noted that similar consequences may also occur in the case of a feeder stagnation break, especially if the initial break size is increased as a result of the channel rupture. Although most of the arguments presented in this statement address tests and observations done mainly by OPG, CNSC staff considers them applicable to all currently-licensed CANDU reactors.

2.0 Closure Criteria

To achieve closure, licensees are expected to:

- demonstrate that the hydrogen mitigation measures are such that the integrity of the calandria and the containment are assured;

- submit proposals for a course of action that would result in the reduction in the risk associated with such an event; and

- submit the following, in the event that cost-benefit arguments are used in support of the proposals mentioned above:

  - a description of the cost-benefit assessment process;
  - the cost-benefit tools and associated documentation;
  - the consequence assessment methodology;
  - the consequence assessment results;
  - an examination of the various options (e.g. design, procedural,...) for event mitigation;
  - studies on pressure and calandria tube failures and end fitting ejection; and
  - the final cost-benefit analysis report.

3.0 Progress to Date

Industry plans to submit proposals to the CNSC for a course of action based on cost-benefit analyses. The closure date for this GAI is August 31, 2001.

Generic Action Item 95G03
"Compliance with Bundle and Channel Power Limits"

1.0 Safety Issue

The limiting values for bundle and channel powers are specified in the Operating Licence for each station.

Licensees ensure compliance with these limits by following operating procedures, which are based on analyses. However, current validation of the channel and bundle power analyses method is such that the errors associated with their calculations are not well-defined. If larger allowances for uncertainties were needed, channel or bundle power may become more limiting than bulk power and derating may, therefore, be required.

At issue are several areas both in the compliance analyses and procedures where, in the view of the CNSC staff, improvements are needed to ensure adequate compliance with bundle and channel power limits.

2.0 Closure Criteria

To achieve closure of the generic aspects of this issue, licensees are required to complete their related code validation program and perform additional analyses to address the identified issues related to methodology, models and computer codes. Specifically, the following should be performed:

- determining the adequacy of error allowances and level of confidence, to cover the various sources of uncertainty in codes predictions, plant data measurements and methodology; allowances should, at 98% confidence level, account for:
  - error in total reactor power normalization;
  - error in code methodology and modelling;
  - error in measurements (instrumented channels and flux detector/mapping); and
  - xenon transients initiated by fuelling.
- determining the adequacy of the validation of computer codes used for core-tracking and compliance with licence power limits, consistent with validation plans for other safety and licensing codes; and
- evaluating the acceptability of compliance procedures, addressing in particular:
  - consistency;
  - actions when limits exceeded; and
  - assurance of compliance during periods of time at which core-tracking runs are made.

3.0 Progress to Date

Licensees has performed work in the following areas: establishment of operating bundle and channel power envelops, assessment of errors in power measurements and power calculation methodologies, computer code validations and compliance procedures. CNSC staff is currently reviewing this work, with the GAI expected to be closed in the near future.

Generic Action Item 95G04
"Positive Void Reactivity - Treatment in Large LOCA Analysis"

1.0 Safety Issue

CANDU reactors have a positive void coefficient of reactivity. In the postulated event of a large LOCA there is an increase in core power, due to positive void reactivity feedback. CANDU reactors have specific engineered features designed to limit the voiding rate in the core and mitigate the power pulse. Two automatic independent shutdown systems have been designed to quickly insert negative reactivity to offset the positive void reactivity. This engineered design solution is based upon an inherent feature of CANDU heavy water-natural uranium lattice: a long prompt neutron life time. The timing and rundown characteristics of each shutdown system are expected to limit the magnitude and duration of the power pulse, and to ensure that the energy deposition in the fuel will not jeopardize the fuel and fuel channel integrity.

The safety analyses in support of the acceptability of the safety systems' performance to ensure meeting the fuel and fuel channel integrity acceptance criteria are based, to a large extent, on numerical simulations of the power pulse. It is therefore important that safety analyses account for the positive void coefficient of reactivity in a conservative manner. This requires the assessment of the accuracy in determination of this coefficient. For fresh fuel at cold conditions a significant amount of data is available from experiments performed in ZED-2 reactor at Chalk River Laboratory. However, the current validation of the theoretical models and computer codes used by the CANDU industry are such that errors associated with void reactivity calculations are not well defined due to a lack of specific experimental data at in-reactor operating conditions and fuel burnups. Although an allowance for uncertainty is included in the safety analysis, the adequacy of this allowance for power reactor conditions is not fully demonstrated, due to the lack of specific experimental data.

Initially, CNSC's staff was concerned with the adequacy of the allowance and raised GAI 95G04 requesting the licensees to increase the uncertainty allowance and to provide more information on relevant research. The licensees initiated an industry-wide experimental program carried out in the ZED-2 reactor at Chalk River Laboratories.

Subsequent developments, including new experimental results from the experimental program indicated that the interim value of void reactivity error allowance (VREA), which is applied to predictions of the design and licensing code POWDERPUFS-V (PPV), is not adequate. Furthermore, it has been recognized that PPV significantly under-predicts the void reactivity effect for conditions specific to power reactors and typical average fuel burnup.

There are several areas where, in the view of CNSC staff, specific actions are needed to ensure a high confidence level of results of large LOCA analyses. These areas are:

- the accuracy and validation of current reactor physics licensing methods and computer codes used for power pulse analyses;
- the suitability of the experimental program to support the validation of reactor physics codes and data for conditions specific to power reactors and anticipated accident conditions; and
- the acceptability of results of power pulse calculations performed with more accurate and validated methods, and adequate allowances, in support of safety system performance.

2.0 Closure Criteria

To achieve closure, licensees are required to complete a suitable experimental program and related analyses based upon more accurate methods and adequate allowances, and undertake adequate interim measures. The following specific closure criteria should be met:

- Perform a systematic and comprehensive review and assessment of the various uncertainties and sources of error involved in the large LOCA methodology regarding void reactivity;

- Provide further supporting evidence for the predictions of the void reactivity coefficient and details of the experimental program and analytical benchmarks. Ensure that the experimental program and analytical benchmarks address the following issues:

    - the effect of operating conditions, such as burnup, coolant purity, moderator poison, moderator purity, fuel temperature, and pressure tube creep;
    - the effect of uncertainties in nuclear data; and
    - the effect of uncertainties related to limitations of diffusion approximation, core voiding pattern during a LOCA accident, localized absorbers representation, and fuel burnup distribution.

- Revise the large LOCA analyses by using more accurate and validated reactor physics methods regarding void reactivity and experimentally-based allowances.

3.0 Progress to Date

An extensive series of tests have been completed and analyzed in ZED-2; a comprehensive programme of code-to-code comparisons has been done; industry has now submitted a package to CNSC covering all of the closure criteria. The closure date is the end of 2002.

Generic Action Item 95G05
"Moderator Temperature Predictions"

1.0 Safety Issue

In certain loss-of-coolant accidents (LOCA) events, the integrity of some fuel channels depends on the capability of the moderator to become the ultimate heat sink. As fuel channels heat up, pressure tubes expand and some may contact their respective calandria tubes. Fuel channels where contact has occurred will likely fail if the outside of the calandria tubes is dried out. One of the important parameters that determines calandria tube dry-out is the degree of subcooling available in the moderator. An accurate prediction of moderator temperature distribution is therefore required to demonstrate fuel channel integrity under accident conditions.

Since moderator flow is three-dimensional (3-D) in nature, licensees have developed a 3-D computer code to calculate the distributions of flow and temperature in the moderator for various reactor operating conditions. This code, MODTURC-CLAS, has been validated against data from a 2-D test facility and other separate effects experiments. The code has been validated also against in-reactor moderator temperature measurements taken for Pickering Unit 5 and Bruce NGS Unit 3. The data were very few, and the code did not predict the slow temperature oscillations that were measured in the reactor. This validation has left many questions unanswered, especially since the magnitude of the oscillations is much larger than the subcooling margins that exist for most CANDU stations.

CNSC staff has therefore concluded that there is a need for producing code validation data from a 3-D test facility that represents the geometry and operating conditions of the CANDU reactors. This generic action item was then raised, and subsequently a Moderator Test Facility was constructed and commissioned at AECL-CRL in 1999 to support the CANDU-9 development.

2.0 Closure Criteria

To bring this generic action item to closure, the licensees are required to carry out:

- 3-D integrated test program under simulated reactor conditions; test results are to be analyzed and the underlying phenomena identified; and
- code validation against the integral test data; pre-predictions should be carried out as part of code validation; validation results should be used to determine code uncertainties for reactor applications.

3.0 Progress to Date

Testing for the purpose of validating the relevant computer code continues as planned at the CRL test facility. The closure date is December 31, 2001.

Generic Action Item 98G01
"PHT Pump Operation Under Two-Phase Flow Conditions"

1.0 Safety Issue

The operation of Primary Heat Transport (PHT) pumps under low suction pressure and significant void can be detrimental to the integrity of the PHT system piping due to the generation of large-amplitude pressure pulsations and excessive pumpset vibration. In the past, the PHT piping fatigue analysis was done using a limiting forcing function (harmonic excitation) obtained from laboratory tests of full-scale PHT pumps. Given the underlying assumptions, especially the amplitude and frequency of excitation, this approach was very sensitive to interpretation of the test data and their application to the PHT system. Consequently, the assessment of the piping fatigue life may not have been conservative.

Further work was therefore required to develop a mechanistic pump model from the available data base and apply it to the PHT system piping configuration. Compared to the use of some arbitrarily assumed limiting forcing function, this additional work was expected to give a more realistic representation of the behaviour of the PHT pump and piping under various accident conditions.

2.0 Closure Criteria

To achieve closure of this action item, for stations other than Darlington, licensees are required to perform the following:

- re-assess the PHT pump behaviour under two-phase flow conditions with emphasis on the unsteady feedback from the PHT system piping (consider assessment of the latest full-scale tests involving the Darlington PHT pump); and
- update the fatigue analysis of the PHT system piping.

3.0 Progress to date

This GAI has been closed for two utilities. Analyses by the other utility are expected in the near future. Closure should be shortly thereafter.

Generic Action Item 98G02
"Validation of Computer Programs Used in Safety Analysis of Power Reactors"

1.0 Safety Issue

Safety analysis is used to establish certain safety-related information about the design and behaviour of the reactor and the safety systems under various conditions including normal operation and certain postulated events such as loss-of-coolant accidents. Such information is provided in the Safety Report, and its updates, and is a primary definer of the station's licensing basis and the bounds of the safe operating envelope. Analysis is also used to demonstrate that the station is being operated within the conditions of the operating licence.

The credibility of this safety-related information depends to a great extent on the degree of conservatism incorporated into the safety analysis and on the qualification of the individual safety analysis activities and tools such as computer programs, analysis methods, and input information. Increasing recognition is being given by both licensees and the CNSC to the importance of safety analysis qualification and one important element of this qualification is computer program validation.

In the past, CNSC staff undertook a number of assessments of licensees' computer programs and safety analysis methods and has identified a number of issues with respect to computer program validation. Issues identified include a lack of a managed process in performing validation of computer programs, insufficient documentation of computer program validation, inadequate applicability of validation due to the limited range of conditions in the validation experiments in comparison with the reactor analysis, inadequate assessment of the impact of dimensional scaling, and important phenomena for which adequate validation data does not exist. CNSC staff has concluded that these deficiencies are affecting the overall confidence in the safety analysis results.

The industry has developed a generic framework for computer program validation and CNSC staff has been relying upon this generic approach to improve the computer program validation. However, to date, the results of this generic approach have been limited, and few specific commitments have been made by the licensees. CNSC staff has concluded that licensees need to establish specific validation programs to improve computer program validation and to provide the necessary confidence in the safety analysis.

Currently, there are a number of open generic action items for which validation of computer programs is an outstanding issue. These GAIs include:

- GAI 88G02 Hydrogen Behaviour in CANDU Nuclear Generating Stations
- GAI 90G02 Core Cooling in the Absence of Forced Flow
- GAI 95G04 Positive Void Reactivity - Treatment in Large LOCA Analysis
- GAI 95G05 Moderator Temperature Predictions

2.0 Closure Criteria

To achieve closure, licensees are required to:

- undertake a code validation program;
- provide bi-annual summary reports describing the overall progress and the major milestones achieved; and
- submit the information identified below; the extent of information provided shall be sufficient to demonstrate that the expectations in this position statement have been met:

  - A list of the computer programs and corresponding applications to which the licensee considers this GAI to be applicable.
  - Description of the key elements of the validation process. (For the industry's proposed validation process, the purpose of each of the documents: technical basis document, validation matrices, and validation plans should be defined, their overall content described and the inter-relationships between the documents defined.)
  - A report for each computer program which summarizes the range of conditions over which the program is intended to be used and compares these with the ranges of conditions in the experimental database. The report will identify "gaps" in the validation experimental database and either the plans for their closure or a justification for leaving them open.

In addition, the following information is required as related to each event or group of events. The way in which the information is organized into individual reports, and the extent of information needed for a particular computer program and application, are left to the discretion of the licensee:

- Identify the inter-relationships between the type of safety analysis, the important safety limits to be met, the technical disciplines and the overall goals of the validation.
- In each of the technical disciplines, identify all phenomena for which validation is potentially required, and justification for any ranking used.
- Identify experimental facilities which can be used to validate the programs for the required phenomenon; indicate any gross physical distortions that may make the facility and the way in which it behaves atypical of the reactor: such distortions include geometric differences (including scaling), differences in material properties, differences in fluid physical properties.
- Identify specific experimental tests that will be used in the validation exercises; indicate the ranges of relevant conditions in comparison with those typical of the equivalent reactor case.
- Produce validation plans for each of the programs: the plans should include sufficient information (together with other documents) to demonstrate that once the validation is complete the programs will be adequately validated; the validation plans should include the rationale for the extent of validation.
- Produce validation reports for each of the programs; the reports should contain sufficient information to demonstrate the claimed accuracy of the program for the given application.

3.0 Progress to Date

Computer code validation continues as planned. Audits have been conducted by CNSC staff, and their recommendations are being implemented. The industry has collaborated on a joint programme of code verification and validation costing around $50,000,000, and on standardization of most safety analysis codes; and most of this work will be completed by December 2001.

Generic Action Item 99G01
“Quality Assurance of Safety Analysis”

1.0 Safety Issue

The CNSC expects power reactor licensees to conduct their operations in accordance with a Quality Assurance program (QAP). The QAP includes QA requirements for various safety-related activities. Safety analysis is one activity to which the QAP shall apply.

Safety analysis is used to establish certain safety-related information about the behaviour of the reactor and the safety-related systems (and in particular the special safety systems) under normal conditions and certain postulated events such as loss-of-coolant accidents. Such information is provided in the Safety Report, and its updates. The Safety Report is a key document that defines the station’s licensing basis and the bounds of the safe operating envelope. The acceptability of this safety-related information depends to a great extent on the degree of conservatism incorporated into the safety analysis; it also depends on the credibility of the safety analysis tools and activities (such as computer codes, analysis methods, and input information).

It is important that licensees perform safety analyses in a systematic manner so that high confidence can be attributed to the definitions of the licensing basis and safe operating envelope for each of the licensees’ stations.

In recent years CNSC staff has became aware of an increasing number of occurrences of inadequate quality assurance in safety analyses performed by licensees. These deficiencies have been identified through audits and assessments by both licensees and the CNSC. Examples are:

- inadequate control over the use of computer codes resulting in the inappropriate use of these codes beyond their originally intended application;
- basic errors in fundamental conservation equations incorporated into codes;
- inadequate validation of computer codes;
- codes for which the theoretical bases have not undergone an appropriate level of peer review or independent technical scrutiny;
- inadequate independent peer review of safety analysis methods;
- inconsistencies between safety analyses input and plant data;
- significant errors in safety report updates;
- inadequate code documentation (controlled or otherwise);
- inadequate planning of safety analysis in support of outages, resulting in analyses being carried out in an unplanned and hasty manner; and
- inconsistent reporting of safety analysis errors to the CNSC.

CNSC staff has concluded that the inadequate quality assurance with respect to safety analysis is resulting in a significant reduction in the overall confidence in the safety analysis results.

2.0 Closure Criteria

To achieve closure, licensees are required to:

- have a QAP that includes requirements for safety analysis activities, and which meets applicable QA standards;

- undertake a QAP assessment in accordance with N286.0 to determine the effectiveness of the licensees' current QAPs specifically with respect to safety analysis activities; this assessment shall include a formal program review and audits; the program review shall determine the extent to which the QAPmeets applicable QA standards and CNSC staff expectations;
- submit to the CNSC a report of the assessment identifying program deficiencies and a plan and schedule for their correction;
- implement the corrections to the QAP;
- submit to the CNSC sufficiently detailed information to demonstrate that the QAP meets the requirements of applicable QA standards and this position statement; and
- provide six-monthly progress reports on resolution of this GAI.

3.0 Progress to Date

Progress has been made on this GAI in accordance with the requirements described above under "Closure Criteria". The closure date is December 31, 2001.

Generic Action Item 99G02
"Replacement of Reactor Physics Computer Codes Used in Safety Analysis of CANDU Reactors"

1.0 Safety Issue

Licensees use reactor physics methods and computer codes to provide important safety-related information in various areas of interest, such as nuclear design, operation, and compliance with the safe operating envelope. There are high expectations concerning the accuracy and validation of these methods and codes, and their predictions for normal and accident conditions, because of their role in the fundamental support of the robustness of the design and in the confirmation of safe operation; consequently, they should reflect the current state of knowledge in the area of reactor physics.

Recent experimental data, as well as reviews of key computer codes, have identified several shortcomings in the reactor physics area. The most important are: inaccurate predictions of key parameters for accident conditions, lack of proper validation data for important phenomena and range of conditions, and a significant gap between the state of knowledge reflected in the licensees' computer codes and the current state of knowledge in this area. These shortcomings have had a negative effect on the overall confidence in predictions of the reactor physics analyses, especially, for those analyses of design-basis accidents where safety margins are small. The use of a given reactor physics computer code would impact the safety margins with respect to the derived acceptance criteria within the accuracy, and associated sensitivity, of this computer code to predicted reactor physics parameters.

Currently, the industry is planning to retire some old reactor physics computer codes, such as POWDERPUFS-V (PPV) and SMOKIN, and is carrying out a validation program of more accurate ones such as WIMS-AECL, DRAGON, and WIMS-AECL based RFSP. This validation program, however, does not address issues which are specific to the code replacement process itself.

At issue, is the apparent absence of a distinct and separate program on replacement of various reactor physics codes, which is needed to address appropriately several areas requiring specific actions. These areas are:

- the approach to replacement of specific reactor physics computer codes, that would include firm schedules and dates of replacement;

- the impact of reactor physics computer codes replacement on safety margins; and
- the impact on safety report update programs, and coverage for any interim period, where a computer code has been replaced, during which period the full impact on all safety analyses may not have been fully addressed.

This position statement therefore defines CNSC staff expectations with respect to replacement of current reactor physics computer codes. It also identifies the links between this generic action item and activities being performed under other GAIs.

2.0 Closure Criteria

To achieve closure, licensees are required to undertake a structured program of replacement of reactor physics computer codes. The program should include the following:

- implementation of a structured approach of specific code replacement, including firm schedules and dates of replacement;
- replacement of all the relevant reactor physics computer codes used in safety analysis and operation, including PPV, MULTICELL, PPV-based RFSP, and SMOKIN; PPV should also be retired from use for fuel management and core tracking simulations;
- ensuring that validation of new codes is in accordance with requirements in GAI 98G02, Regulatory Guide G-149, and CSA Standard N286.7;
- assessment of the impact of code replacement on current safety margins and identify the limiting design-basis accidents whose safety margins might be significantly affected;
- assessment of the impact on safety report updates' programs and identify ongoing and future activities that might be affected by the reactor physics codes' replacement;
- definition of adequate interim allowances for use with key reactor physics parameters provided by PPV for safety analysis, such as void reactivity, delayed neutron fraction, fuel temperature reactivity, prompt neutron lifetime, until replacement of PPV; and
- adequate coverage for any interim period, where a computer code has been replaced, during which period the full impact on all safety analyses may not have been fully addressed.

3.0 Progress to Date

The industry has committed to use the more sophisticated codes WIMS/RFSP/DRAGON; that are now industry Standard Tools; and that current LOCA analyses at AECL and the utilities use them. Some progress has been made in accordance with the requirements described above under Closure Criteria. The following work was completed by all utilities:

- Providing the framework and target dates for completion, including the list of reactor physics computer codes;

- Providing the interim allowances for use with key reactor physics parameters provided by PPV for safety analysis;

The preparation of six-monthly summary reports describing the overall progress and the major milestones achieved is in progress. The work program on this GAI should be completed by the end of December 2002.

Generic Action Item 00G01
"Channel Voiding During a Large LOCA"

1.0 Safety Issue

Reactivity effects of channel voiding in CANDU reactors are positive. When the break occurs in the primary heat transport system, the system depressurizes. Some coolant in the fuel channels associated with the broken pass turns into steam. This coolant voiding in the fuel channels results in a reactor power pulse due to a positive void reactivity feedback. The increase of reactor power must be arrested and limited by the two fast-acting shutdown systems. Safety assessments have been performed for large LOCAs to show the shutdown system effectiveness.

Two significant parameters that determine the magnitude of power pulse are:

- rate and extent of coolant voiding; and
- reactivity changes due to voiding.

These parameters are predicted by computer models. However, they have not been adequately validated as there are no direct void fraction measurements applicable to CANDU fuel channel conditions. This GAI is intended to deal with the channel voiding issue since the void reactivity issue is being dealt separately under another GAI.

2.0 CLosure Criteria

To bring this generic action item to closure, the licensees are expected to carry out:

- channel void measurements during large LOCAs relevant to reactor conditions; the effects of heat transfer rate and scaling on channel voiding should also be quantified;
- validation exercises with the relevant safety analysis computer codes against the channel void data. There should be sufficient information to demonstrate the claimed accuracy of the code for the given application; and
- an impact assessment of the safety margins in the safety report.

3.0 Progress to Date

One utility is expected to provide a report on neutron scatterometer performance, a report on test matrix for the RD-14M experiments, and a report on the analysed data from RD-14M experiments.

The other two utilities are expected to fully qualify a neutron scatterometer and to complete a comprehensive uncertainty analysis of measurements including flow regime during the 2000/2001 fiscal year, perform large LOCA tests in RD-14M during the 2001/2002 fiscal year, and perform formal code validation using some of these tests by December 2001. Work is proceeding on schedule.
In addition to the above, CNSC staff expects the industry to submit a plan addressing the scaling effects before June 2001 (report is still expected), and that validation exercises be performed as date becomes available and that final validation reports be submitted before June 2002.

B. Closed Generic Action Items:

Generic Action Item 89G03, “Ontario Hydro’s Pressure Tube Inspection Program”

In 1989, CNSC (then AECB) staff concluded that OPG’s (then OH) proposed in-service inspection program did not adequately cover all degradation mechanisms. The CANDU owners Group (COG) has since developed fitness for service guidelines that were acceptable to CNSC staff. OPG has committed to using these guidelines, and this was confirmed through ongoing licensing and surveillance activities. Surveillance programs have also served to verify the threshold for blister formation.

Generic Action Item 89G05, “Use of Mercury Wetted Relays in Safety Related Systems”

Failure rates of the 5 amp non-tin-doped mercury wetted relays was found to be unacceptably high. In 1989, CNSC (then AECB) staff had requested the licensees to ensure that the reliability of the safety related systems using such relays still met their targets, or to replace them. All licensees have replaced these relays.

Generic Action Item 92G01, “Treatment of Human Factors in Ontario Hydro Reliability Analyses”

In 1991, OPG (then OH) submitted a reliability analysis in support of a proposed design change on the Pickering filtered air discharge system. The reliability study included an assessment of human reliability. The CNSC had concerns regarding the methodology used in the human reliability assessment, and has raised them as a generic action item based on the request of the licensee. CNSC has also initiated a research project to assess OPG’s human reliability methodology called THERP. The results of this assessment confirmed the acceptability of this methodology. CNSC is also monitoring OPG human factor analyses on a case by case basis.

Generic Action Item 96G02 “Feeder Pipe Fitness for Service”

Inspections in several CANDU reactors revealed an 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. However, the expected lifetime of some of the feeders could be limited by the current rate of degradation. Licensees were asked to show that feeders are fit for service. They were also asked to show sufficient understanding of the thinning phenomenon to prevent it from threatening the integrity of the feeders.

Licensees submitted a report to the CNSC including results of their investigation of the most likely cause for feeder thinning. They have also committed to a periodic feeder inspection program.

ANNEX 6.2
Systems Included in the Configuration Management Closure Project

TABLE A6.1: Systems Included in the Configuration Management Closure Project

CMCP Systems List			
Pickering "A"	**Pickering "B"**	**Bruce "B"**	**Darlington**
SDS-A	SDS1	SDS1	SDS1
SDS-E	SDS2	SDS2	SDS2
Emergency Coolant Injection System	Emergency Coolant Injection System	Containment Systems	Containment Systems
Negative Pressure Containment	Negative Pressure Containment	Emergency Coolant Injection	Emergency Coolant Injection System
Reactor Control Units (Reactor Regulating System)	Reactor Regulating System (RRS)	Reactor Regulating System	Reactor Regulating System (RRS)
Primary Heat Transport System Auxiliaries	Main Heat Transport System Auxiliaries	Main Heat Transport System Auxiliaries	Primary Heat Transport System Auxiliaries
Stanby Cooling System /Shutdown Cooling System	Shutdown Cooling System	Shutdown Cooling System	Shutdown Cooling System
Main Moderator and Auxiliary Systems	Main Moderator and Auxiliary Systems	Main Moderator Supply System	Main Moderator and Auxiliary Systems
Boiler Steam and Water Systems	Boiler Steam System	Feed and Water Condensate Systems	Main Steam Supply System
Feedwater Heating, Condensate and Boiler Feed Systems	Feedwater Heating, Condensate and Boiler Feed Systems	Emergency Water System	Feedwater Condensate Systems
Emergency Boiler Water (Supply) System	Emergency Water System	Powerhouse Emergency Venting System	Emergency Service Water System
Boiler Emergency Cooling System (BECS)	Boiler Emergency Cooling System (BECS)	Service Water Systems	Steam Generator Emergency Cooling System
Powerhouse Emergency Venting System	Powerhouse Emergency Venting System	Electrical Power Systems	Powerhouse Ventilating System *(Portions Required for Emergency Venting)*
Service Water Systems - Emergency Low/High Pressure Service Water System	Service Water Systems		Service Water Systems
Electrical Power Systems	Electrical Systems		Electrical Systems

ANNEX 6.3
Summary of Major Design and Operational Changes Resulting from Canadian Nuclear Safety Commission Actions

This Annex provides examples of significant CNSC regulatory actions, both instructions to the licensees by the CNSC, and major undertakings or initiatives by licensees in response to CNSC insistence and inquiry. The list is not comprehensive and is intended only to illustrate the type of actions taken by the CNSC 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 very small portion of the total number of actions assigned to the licensees by the CNSC. 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 CNSC about nuclear power stations in their design or construction phases, including stations intended for offshore use, are not included.

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, then the 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 resulted in significant power derating for several months. Procedural and hardware changes are in the process of being implemented before the reactor is returned to high power operation.

BRUCE “A” and “B” POSITIVE REACTIVITY DUE TO FUEL RELOCATION FOLLOWING A LOSS OF COOLANT ACCIDENT

CNSC questions in 1993 about 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 the channel inlet would result in fuel movement which would increase the reactivity in the core. Since that time OPG implemented a number of equipment and operational changes and provide better technical support and analysis before raising power. These reactors are still derated to 90% of full power operation.

PICKERING AND BRUCE EMERGENCY CORE COOLING SYSTEM (ECCS)

Following the recognition that ECCS could not prevent fuel failures for Large Loss Of Coolant Accidents (LLOCAs), CNSC staff instructed OHN to conduct a system-by-system review of the impact, which resulted in design changes. The CNSC instructed that major shielding be added to the Bruce “A” ECCS. This instruction also led to a redesign of the Bruce “B” ECCS, then under construction.

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 CNSC, 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.

Incidents of ECCS strainer blockage in the US and Europe prompted the CNSC to request a review of strainer design in CANDU reactors. There is the possibility that piping insulation and other debris could block the strainers in the ECCS in the Point Lepreau, Gentilly-2, Pickering and Bruce reactors. Design improvements to the strainers have been improved, reviewed, and approved, and progress is being made regarding installation of the strainers at all stations.

BRUCE “A” CONTAINMENT

Tests requested by the CNSC 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 CNSC also required major improvements to the emergency filtered air discharge systems at Bruce.

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 CNSC instruction to increase the time at which venting the containment becomes necessary following an accident. The CNSC 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 “A” reactors were licensed for operation by the CNSC before the introduction of the regulatory requirement for two independent, diverse and fully capable shutdown systems. Therefore, the Pickering “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 has been a licensing issue with the CNSC 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 parameter.

Following the 1986 Chernobyl accident, CNSC staff requested OPG to reassess the safety of the Pickering "A" reactors under those dual failure assumptions that involve the failure to shut down. In 1987, OPG submitted a revised analysis of the consequences of a large-break LOCA combined with failure to shut down. The analysis performed by OPG concluded that the structural integrity of containment would be maintained and that the dual failure reference dose limits would be met. CNSC staff found this analysis to be speculative and concluded that the consequences could not be quantified with confidence. OPG decided to investigate the SDSE following discussions between CNSC and OPG staff. 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 CNSC 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 to Bank A to bring the total to 23.

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.

OPG committed to installing this enhanced shutdown system on all Pickering "A" operating reactors by the end of 1997.

SDSE was installed in Unit 4 of PNGS-A, before the plant was laid up. The commissioning was successfully completed and the system was readied for future on-power testing. Some minor installation was also done on the other units. Installation of the SDSE is a condition for returning the Pickering "A" units to service.

LARGE-BREAK LOSS OF COOLANT ACCIDENT

A recent study performed under the generic action item (GAI 99G02) on replacement of reactor physics computer codes used in safety analysis, predicted results for large break Loss of Coolant Accident (LOCA) which were more severe than those previously submitted to the Canadian Nuclear Safety Commission (CNSC). The discovery was formally reported in accordance with CNSC regulatory document R-99, in February 2001.

The large LOCA event encompasses a range of break sizes up to and including guillotine rupture of the Heat Transport System, (HTS), piping including the inlet and outlet headers. CNSC staff, applying regulatory documents R-8 and R-10, have set high level acceptance criteria for this class of event relating to:

- reference dose limits; and
- fuel channel integrity.

Typically, the reference dose limits are met with a significant degree of margin (i.e. analysis results are less than 5% of the limits), and the focus of the analysis is on demonstrating that fuel channel integrity is maintained. To demonstrate fuel channel integrity, a number of quantitative derived acceptance criteria have been defined by the industry:

a) avoidance of fuel centre-line melting (avoidance of molten material in contact with the pressure tube);
b) avoidance of fuel sheath melting (avoidance of molten material in contact with the pressure tube);
c) limit on maximum fuel bundle enthalpy (avoidance of fuel dispersal or molten material);
d) margin to constrained fuel string axial expansion (avoidance of fuel buckling, and pressure tube hot spots); and
e) avoidance of sustained calandria tube dryout.

In addition, CNSC staff has adopted three other measures of event severity:

f) margin to prompt criticality;
g) maintenance of fuel bundle geometry; and
h) number of pressure tubes strained.

It is with respect to b) and the latter three measures f), g) and h) that the new results indicated a potential increase in the event severity. The concern has been higher for Bruce reactors which are more vulnerable to large break LOCA events, due to its HTS design.

The increase in predicted severity of the overpower transient following a large break LOCA necessitated compensatory changes to certain operating limits to ensure that the reactors remained bounded by the conclusions of the current licensing basis analysis. These compensatory measures were supported by the industry assessment of:

- the impact of the discovered error on public safety;
- the analysis results using the new administrative limits; and
- historical data for relevant operational parameters.

The industry has also committed a number of programs aimed to provide further confirmation and restore operational and safety margins. These include:

- developing a process for resolution of outstanding physics uncertainties;
- developing a plan for further experimental research to determine the appropriate acceptance criterion for CANDU fuel in the prompt critical region of operation;
- reviewing the feasibility of advancing the current schedule for Bruce B core conversion;
- reviewing further design options that could be used to improve safety margins; and
- reviewing the analytical methods and methodologies available to better quantify the actual margins of safety available for large break LOCA and other design basis events.

Plans for these programs are currently being developed in consultation with CNSC staff.

MAIN STEAM LINES

CNSC 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 leak detection; and
- definition and demonstration of procedures for the secondary control room (SCR), such as the required presence of a dedicated SCR operator.

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

Other safety-related corrective actions were discussed in Canada's report to the first Convention on Nuclear Safety.

ANNEX 7.1
Description of the Nuclear Safety and Control Regulations

A brief description of the regulations and substantive additions to the regulations are discussed in this annex.

GENERAL NUCLEAR SAFETY AND CONTROL REGULATIONS

The *General Nuclear Safety and Control Regulations* contain the general requirements that apply to all licensees. They also continue the exemption for naturally occurring radioactive materials that have not been associated with the development, production or use of nuclear energy. As authorized by the *NSC Act*, a requirement to provide information on any proposed financial guarantees has been added.

Financial Guarantees

Under the *AEC Act* and regulations, only uranium mining facilities were required to provide financial guarantees for decommissioning and waste management. A possible consequence of this would be that the costs associated with these activities at other facilities could fall on the taxpayer if the licensee had not set aside sufficient funds for their completion. To address this, subsection 24(5) of the *NSC Act* provides the CNSC with the authority to include a licence condition requiring financial guarantees in a form that is acceptable to the Commission. The financial guarantees section of the *NSC Act* is being implemented by regulations and licence conditions requiring certain licence applicants to provide information on proposed financial guarantees and to describe their plans for decommissioning and waste management at the end of the life of the nuclear facility. The regulations permit substantial flexibility in the ways that licensees can meet the financial requirements.

RADIATION PROTECTION REGULATIONS

The regulations contain the radiation protection requirements and as such, they apply to all licensees and others who fall within the mandate of the Commission. Medical doses, doses to caregivers who volunteer, and doses to volunteers in biomedical research are specifically excluded from the regulations. The *Radiation Protection Regulations* represent a restatement of the regulatory requirements under the *AEC Act* with revised dose limits and the addition of action levels.

Dose Limits

The dose limits in most countries are based on the recommendations of the International Commission on Radiation Protection (ICRP). Using the most recent data on the effects of radiation, the ICRP recommended lowering the dose limits in 1991 as follows:

- for nuclear energy workers, from 50 millisievert (mSv)/year to 100 mSv for five years (i.e., an average of 20 mSv/year);
- for pregnant nuclear energy workers, from 10 mSv/year to 4 mSv/year; and
- for members of the public, from 5 mSv/year to 1 mSv/year.

The Radiation Protection Regulations incorporate these recommendations.

Action Levels

An action level is a specific dose or other parameter which, if reached, may indicate a partial loss of control of the radiation protection program. The *General Nuclear Safety and Control Regulations* require applicants to submit information on any action level they use or propose to use. If an action level is referred to in a licence, the *Radiation Protection Regulations* require the licensee to investigate, take appropriate actions and notify the Commission when an action level is exceeded.

The establishment of action levels is consistent with the recommendations of the ICRP. Most major licensees have action levels, but they may be identified as reference levels, investigation levels, etc. Reporting when one of these levels is exceeded was not a regulatory requirement under the *AEC Act* or regulations.

CLASS I NUCLEAR FACILITIES REGULATIONS

The requirements specified in the *Class I Nuclear Facilities Regulations* for major facilities such as reactors, high-energy accelerators and uranium processing facilities are essentially the same as those under the *AEC Act*, regulations and licence conditions. The impact of the new regulations on operator recertification and uranium or large radioisotope processing plants that are included as Class I nuclear facilities, are discussed below.

Operator Certification

The former AECB required the senior control room staff of nuclear power reactors to pass examinations administered by the CNSC that tested their competence to operate nuclear reactors safely. Only examinations for initial certification were required, but licensees were expected to maintain the competence of their staff through regular training. For some time, the CNSC has considered that a mechanism for verifying continuing competence is necessary and under the *AEC Act*, it began the process by adding an expiry date to all existing certifications.

Under the *Class I Nuclear Facilities Regulations,* certifications issued by the CNSC expire after five years, and in order to be recertified, senior control room staff will be required to successfully complete a continuing training program and requalification tests administered by the licensee to demonstrate continuing competence. The licensee's continuing training program and tests will be evaluated regularly by Commission staff.

NUCLEAR SUBSTANCES AND RADIATION DEVICES REGULATIONS

The *Nuclear Substances and Radiation Devices Regulations* apply to all nuclear substances, sealed sources and radiation devices not covered by other regulations. As such, they apply to the vast majority of CNSC licences (> 4000). They also contain the criteria for consumer products such as smoke detectors and safety signs using tritium. In general, these regulations reflect international practice but there are some minor variations based upon Canadian policy and circumstances.

The regulations contain the requirements under the *AEC Regulations* and licence conditions, with the addition of servicing licences similar to those described previously for Class II nuclear facilities, and audible alarming dosimeters for exposure device operators. The scheduled quantities defined in the *AEC Regulations* have also been replaced with exemption quantities. This means that the quantities of radioactive material that are exempt from licensing have generally decreased.

Exemption Quantities

The schedule to the *Nuclear Substances and Radiation Devices Regulations* contains a list of the quantities of radioactive material below which no licence is required. The *AEC Regulations* also contain exemption values called "scheduled quantities," but the exemption quantities fixed under the *NSC Act*, which are based on current radiation protection knowledge and the new dose limits, are generally smaller than those found in the *AEC Regulations*.

The *AEC Regulations* exempted from licensing most materials that contain less than one scheduled quantity per kilogram. This exemption was not included in the regulations under the *NSC Act* because of concerns about the risks posed by large volumes of materials that contain low concentrations of radioactive material.

Auditable Alarming Dosimeters

The use of radiation sources to radiograph structures such as pipeline welds, aircraft components and pressure vessels for flaws is one of the most hazardous activities licensed by the CNSC. The new regulations therefore require all exposure device operators to wear an audible alarming dosimeter to alert them to dangerous levels of radiation before significant exposures occur.

PACKAGING AND TRANSPORT REGULATIONS

All industrialized countries use the recommendations of the International Atomic Energy Agency (IAEA) to regulate the transport packaging of radioactive materials. The Canadian requirements in the *Transport Packaging of Radioactive Materials Regulations* are based on the 1973 IAEA recommendations, and the new regulations are based on the 1985 recommendations, as amended in 1990. Many countries and international organizations have already adopted the latter recommendations, so most Canadian exporters and shippers are already in compliance with the packaging requirements. Therefore, the major changes are the requirement for carriers to have a radiation protection program, the expansion of those activities that require quality assurance programs and the use of Type 2 Industrial Packages (IP-2 packages).

The CNSC has been a major participant in the development of the IAEA recommendations on the packaging and transport of nuclear materials. In developing a position on transportation issues, the CNSC has communicated regularly with Transport Canada and the major Canadian shippers. Transport Canada is normally represented at the IAEA meetings, and experts from the industry have accompanied CNSC staff to IAEA meetings when specific topics have been discussed.

Numerous changes were made to these regulations as a result of consultation. The major changes consist of the removal of the requirement for a licence to package nuclear substances for most types of shipments, allowing additional methods to demonstrate that packages comply with the performance requirements and acceptance of emergency response plans that comply with the requirements of the *Transportation of Dangerous Goods Regulations* (*TDG Regulations*). Other changes were made to improve clarity and consistency with the *TDG Regulations*.

Radiation Protection Program for Carriers

The use in Canada of nuclear materials for research, industrial applications, medicine and export is substantial and growing. It is estimated that approximately one million packages containing radioactive material are transported in Canada each year. The safety record of this industry is good because of the continued efforts of licensees, Transport Canada, the transportation industry and the CNSC to improve the packaging and safe handling of nuclear materials. However, as the number of shipments has increased, more drivers and handlers have become involved. The CNSC is aware that some of these drivers and handlers do not have adequate knowledge of radiation to protect themselves, the public and the environment in all transportation situations. In addition, some exposures will have to be reduced to comply with the new dose limits, and training in radiation protection is one of the most effective ways to achieve this.

Quality Assurance Programs

In accordance with the recommendations of the IAEA, the new regulations require every person who designs, produces, tests, uses, services or inspects a package containing radioactive material, or special form material, to have a quality assurance program. This expands the types of packages and the licensed activities that require a quality assurance program under the *AEC Act* and *Transport Packaging of Radioactive Materials Regulations*. The Commission will expect licensees to implement staff training programs and verify that work is performed according to documented procedures. The requirements, which will vary depending on the risks associated with the given activity, will be explained in guidance documents.

IP-2 Packages for Ore Samples Containing More Than 2% Uranium

The properties of high-grade Canadian ores are such that the hazard they pose is consistent with that of type 2 low specific activity (LSA-2) materials, and as such, the use of IP-2 packages is more appropriate. If the IAEA regulations had been followed, all grades of uranium ores would have been considered as LSA-1 material that could be shipped in IP-1 packages. However, this provision was developed in the 1960s, when the known ore grades were approximately 1% uranium.

IP-2 Packages

Adoption of the IAEA's definition of an (IP-2 package) will require packaging of low specific activity radioactive materials to meet new drop and puncture tests when shipped under exclusive use (i.e., when packages are not combined with cargo from other shippers). This will affect primarily waste and heavy water shipments from the power utilities. It should be noted that for shipments that are not exclusive use, there is no change to the requirements.

NUCLEAR SECURITY REGULATIONS

The three new security requirements in the *Nuclear Security Regulations* described below are considered necessary to bring Canadian nuclear facilities up to the internationally accepted recommendations of the IAEA. In developing these new requirements, the Commission has given consideration to the Canadian security context. Through the consultation process, the requirements for searching those entering or leaving a protected area have been modified.

Alarm Assessment System for Protected Areas

Major nuclear facilities in Canada have security measures that are intended to protect them from unauthorized entry. These measures include protected areas and alarm systems. At some sites, a guard is dispatched to investigate the alarm and to report on the cause. This can take some time and the delay in investigating the alarm adds to the response time to address the problem if the alarm is genuine. A new provision has therefore been included in the regulations which will require licensees to continuously maintain, and in some cases, install additional assessment equipment in order to provide accurate and timely alarm assessment.

Alarm Assessment System for Inner Areas

Only two licensees are authorized to store sensitive nuclear material in a high security installation known as an inner area. When the alarm for these areas is triggered, a security guard is dispatched to investigate the cause. The introduction of a mandatory assessment system in the inner area will facilitate the immediate assessment of the cause of the alarm.

Searches at the Perimeter of a Protected Area

Nuclear facilities in Canada are protected by security perimeters that limit access to protected areas. A new provision has been included in the regulations which will require licensees to search, or otherwise monitor, persons without a security clearance and their possessions when entering and leaving the protected area. Licensees also have the right to search, on reasonable suspicion, anyone entering or leaving a protected area. The searches can be carried out by technical means and are similar to the standard of security provided at Canadian airports.

The search procedure will deter terrorists and others from carrying weapons or explosives into protected areas or removing Category I, II or III nuclear material. The proposed regulation allows the operator to use non-intrusive technical means such as metal detectors and X-ray machines in carrying out searches.

NUCLEAR NON-PROLIFERATION IMPORT AND EXPORT CONTROL REGULATIONS

The *Nuclear Non-proliferation, Import and Export Control Regulations* apply in respect of the import and export of controlled nuclear substances, controlled nuclear equipment and controlled nuclear information. Nuclear substances, equipment and information subject to these regulations are included in the schedule to the regulations. The new regulations increase the number of items for which import licences are required so that Canada will be in a better position to implement its international obligations with respect to the control of nuclear equipment. Canada imports little of this equipment, and most companies who would import these items currently have import licences for other reasons, so the overall effect of adding items to the list is not considered to be significant.

URANIUM MINES AND MILLS REGULATIONS

The *Uranium Mines and Mills Regulations* combine the requirements contained in the former *Uranium and Thorium Mining Regulations* and certain licence conditions. The mining industry has expressed concern that some information, such as a preliminary safety analysis report, will now be required at an earlier stage in the life-cycle of a mine or mill. The CNSC believes this information is necessary at an early stage if it is to be satisfied that the operating mine or mill will be capable of meeting regulatory requirements.

CLASS II NUCLEAR FACILITIES REGULATIONS

The *Class II Nuclear Facilities Regulations* specify the requirements for nuclear facilities that pose a lower risk than Class I facilities. These include low-energy accelerators, irradiators and radiation therapy installations. These regulations introduce new requirements for servicing licences and therapy room interlocks.

ANNEX 7.2
Descriptions of CNSC Regulatory Documents

Regulatory document P-119 [Policy on Human Factors] contains a description of how the CNSC will take human factors into account during its licensing, compliance and standards-development activities. It recognizes that human factors can affect the performance of the facilities and activities that it regulates. The term “human factors” means factors that influence human performance as it relates to the safety of a nuclear facility or activity over all phases, including design, construction, commissioning, operation, maintenance and decommissioning.

Regulatory document P-223 [Protection of the Environment] describes the principles and factors that guide the CNSC in regulating 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 in a manner that is consistent with Canadian environmental policies, acts and regulations and with Canada’s international obligations.

Regulatory document P-242 [Considering Cost-benefit Information] describes how the CNSC will consider cost-benefit information in certain of its decision-making processes. It pertains specifically to the CNSC decision-making process in relation to licences and orders, as provided for by the *Nuclear Safety and Control Act*. It directs CNSC staff, and provides guidance to licence applicants, to licensees, to persons who may be required to comply with an order, and to other potential participants in the processes for related decisions.

Regulatory document G-129 [Guidelines on How to Meet the Requirement to Keep All Exposures As Low As Reasonably Achievable] contains guidelines for licensees on how to meet the CNSC requirement to keep doses received by workers and members of the public As Low As Reasonably Achievable (ALARA), social and economic factors taken into account. It provides advice on the type of action required to ensure that exposures are effectively controlled and minimized. It outlines the importance of an explicit commitment by senior management to the objective of restricting exposure as far as reasonably achievable, the need for suitable programs to achieve this objective, and the value of reviewing the work periodically to ensure that exposures continue to be adequately controlled.

Regulatory document G-149 [Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors] provides guidance to licensees involved in the development, maintenance and use of computer programs used in the design and safety analysis of nuclear power plants and research reactors so that a high degree of confidence may be placed in both the programs and the results of their application. It applies to licensees whose computer programs are used in designing or supporting the design of a nuclear power plant or research reactor and/or analyzing operational transients, incidents or accidents. It does not apply to operational control systems software.

Regulatory document G-206 [Financial Guarantees for the Decommissioning of Licensed Activities] provides guidance regarding the establishment and maintenance of measures to fund the decommissioning of activities licensed by the CNSC. It presents information for those who have incurred, or expect to incur, obligation with respect to the decommissioning of activities licensed by the CNSC. Licences issued by the CNSC may contain conditions with respect to the requirements for the submission of decommissioning plans and their associated financial guarantees.

Regulatory document G-219 [Decommissioning Planning for Licensed Activities] provides guidance regarding the preparation of decommissioning plans for activities licensed by the CNSC. It presents information of interest to those who have incurred, or expect to incur, obligations with respect to the decommissioning of activities licensed by the CNSC. It also provides the basis for calculating the financial guarantees discussed in the Regulatory Guide G-206. Generic outlines for the structure and content of both preliminary and detailed decommissioning plans are presented, emphasizing the factors that will assist in determining an appropriate work plan structure, level of detail, and flexibility in a specific plan. Key elements of the generic preliminary and detailed plan are elaborated upon.

Regulatory document G-228 [Developing and Using Action Levels] is intended to help applicants for CNSC licences develop action levels in accordance with paragraph 3(1)(f) of the *General Nuclear Safety and Control Regulations* and section 6 of the *Radiation Protection Regulations*. This document applies to all applications for a CNSC licence, other than an application for a licence to abandon. It describes how the licence applicant can develop action levels that provide for the radiation protection of workers and the public during the conduct of activities licensed by the CNSC.

Consultative document C-98 [Reliability Programs for Nuclear Power Plants] ensures that the risk-significant systems of the nuclear power plant can and do function reliably, in accordance with the relevant design, performance, and safety criteria, including any safety targets of the plant and CNSC licence requirements.

Regulatory document S-106 [Technical and Quality Assurance Standards for Dosimetry Services in Canada] provides applicants, licensees and owners of x-ray equipment with guidance on dose and exposure measurement methods that would be acceptable to the regulatory agency (either the CNSC or the appropriate provincial government agency). It addresses measures associated with assuring accurate dose or exposure measurement and assignment. In addition, it specifies the technical standards for dosimetry services and the quality assurance program which, when implemented, will provide confidence that these standards are achieved and maintained. In order to ensure that all the information that can realistically be collected is available for dose determination, this document includes specifications for the documenting and reporting of monitoring data.

Consultative document C-091 [Ascertaining and Recording Radiation Doses to Individuals (an update of R-91)] describes approaches that may be used by CNSC licensees to ascertain and record exposures and doses under the *NSC Act* and regulations. It discusses related requirements, including obligations on licensees to use licensed dosimetry services and to make information on radiation doses available to workers.

Consultative document C-099 [Reporting Requirements for Operating Nuclear Power Plants (an update of R-99)] presents the reports that are mandatory for operating nuclear power plants, as a term or condition of the operating licence. Based on pertinent sections of the *Nuclear Safety and Control Act* and the *Class I Nuclear Facilities Regulations*, the information provided herein includes frequency of submission and content requirements. This document is proposed to replace R-99, *Reporting Requirements for Operating Nuclear Power Facilities*, dated January 1, 1995.

Consultative document C-118 [Relationship Between Dose Limits for the Public and Operating Emission Levels for Nuclear Facilities]clarifies the relationship between regulatory dose limits for members of the public and operating emission levels for licensed facilities. It stipulates that operating emission levels should be determined by applying the ALARA principle and should be specified in terms of effluent emission rates or dose rates; they should not be established as an arbitrary fraction of the derived emission limits.

Consultative document C-138 [Software in Protection and Control Systems] describes what constitutes a sufficient description of software used in protection and control systems and what constitutes a sufficient demonstration that the software design is adequate. It is intended, not as a standard, but as a guide to assist CNSC staff in their assessment of software, and to guide applicants and licensees in the preparation of their submissions to the CNSC.

Consultative document C-144 [Trip Parameter Acceptance Criteria for the Safety Analysis of CANDU Nuclear Power Plants] describes the criteria to satisfy certain requirements relating to the acceptability of the trip parameters to be used in nuclear power plants to prevent fuel failures and consequential failures of the pressure tubes. These criteria are intended for application to fuel bundles that have pertinent and adequate experimental data on post-dryout heat transfer and fuel deformation.

Consultative document C-204 [Certification of Persons Working at Nuclear Power Plants] describes the qualifications, training, examinations and certifications that may be required of nuclear power plant (NPP) personnel for positions referred to in a term of condition of their operating licence. It describes the licensee's obligations concerning the certification of persons and also describes their obligations to ensure that only competent persons operate or supervise the operation of an NPP.

Consultative document C-205 [Access Control for Protected and Inner Areas of Nuclear Facilities] provides information on how to meet the regulatory requirements of the *Nuclear Security Regulations* regarding physical protection measures at nuclear facilities, specifically access control for protected and inner areas. It identifies the regulatory requirements relevant to access control for protected and inner areas, outlines the CNSC's expectations of licensees with respect to controlling access to those areas, and describes recommended practices for providing effective physical protection measures for the protected and inner areas of nuclear facilities.

Consultative document C-208 [Transport Security for Category I, II and III Nuclear Material] explains the information requirements of the transportation security plan as set out in the *Nuclear Security Regulations* and describes what the CNSC considers to be effective ways to comply with these transportation security requirements. This plan is required when applying for any licence to transport Category I, II, or III nuclear material outside of the authorized area for that nuclear material.

Consultative document C-225 [Emergency Planning at Class I Nuclear Facilities and Uranium Mines and Mills] applies to applicants for a CNSC licence to operate a Class I nuclear facility, to applicants for uranium mine and mill licences, and to CNSC staff involved in assessments of proposed and existing emergency measures. It describes and discusses the elements of emergency preparedness and response that licence applicants should typically consider when they are developing plans to prevent or mitigate the effects of accidental releases from a Class I nuclear facility or a uranium mine or mill.

Consultative document C-273 [Making, Reviewing and Receiving Orders Under the *Nuclear Safety and Control Act*] describes the roles and responsibilities of inspectors and designated officers (DOs) charged with making and reviewing orders under the *NSC Act*, and the steps required to carry out these activities. It also advises persons subject to orders about their rights and responsibilities, and the steps required to respond to an order.

Consultative document C-274 [Preparing a Security Report for Licence Applications] provides guidance for licensees when preparing and submitting documentation to meet the security requirements pursuant to the *Nuclear Safety and Control Act* and its regulations. It is intended to assist the licensee and licence applicant when preparing a Security Report for the CNSC licence application and renewal process. This guide applies to licensees and licence applicants for Category I and II nuclear materials, and facilities with a nuclear reactor that may exceed 10 megawatts thermal power during operation.

Consultative document C-276 [Human Factors Engineering Program Plans] guides licensees and licence applicants in their preparation of an effective Human Factors Engineering Program Plan that adequately incorporates human factors elements into licensable activities. It presents the technical elements that might be covered in such a plan, which would take into account the risk, complexity, and the potential impact on the health and safety of persons and the environment of the licensed activity.

Consultative document C-278 [Guide to Human Factors Verification and Validation Plans] assists licensees and licence applicants in preparing an effective Human Factors Verification and Validation Plan. The goal of this Plan is to establish that the human factors elements of a project or activity, that is licensed or licensable by the CNSC have been adequately addressed. It provides the elements of an effective plan for the verification and validation of the human factors elements of activities licensed or licensable by the CNSC.

Regulatory document R-91 [Policy on Monitoring and Dose Recording for the Individual] specifies the required conditions for the monitoring of individuals who receive, or may receive, significant radiation doses or exposures. It defines both personal and non-personal monitoring and indicates the procedure for recording a person's individual dose, under either of these two monitoring systems. It also outlines a policy statement regarding the personal monitoring techniques utilized by a licensee.

Regulatory document R-100 [The Determination of Effective Doses from the Intake of Tritiated Water] describes a CNSC decision to improve the determination of effective doses from the intake of tritiated water in accordance with a report published by the Federal-Provincial Working Group on Bioassay. This report, *Bioassay Guideline 2: Guidelines for Tritium Bioassay*, requires a detailed estimate of the concentration of tritium in soft tissue derived from urinalysis measurements.

Regulatory document R-105 [The Determination of Radiation Doses from the Intake of Tritium Gas] describes the requirements of CNSC licensees with respect to the monitoring of radiation doses from the intake of tritium gas. It outlines the factors to be included in the calculation of the effective dose equivalent for both soft tissue doses and lung doses.

Regulatory document R-117 [Requirements for Gamma Radiation Survey Meter Calibration] describes the minimum requirements for calibrating a portable gamma radiation survey meter with a beam of radiation from a know calibration source. It explains the responsibilities of a licensee who calibrates its own survey meters and/or who contracts an approved calibration agency to calibrate its survey meters. It also includes the requirements for record keeping and examples of the supporting documentation.

The following documents were discussed in Article 7.2:

- R-7 Requirements for Containment Systems for CANDU Nuclear Power Stations
- R-8 Requirements for Shutdown Systems for CANDU Nuclear Power Stations
- R-9 Requirements for Emergency Core Cooling Systems for CANDU Power Plants
- R-10 The Use of Two Shutdown Systems in Reactors
- R-77 Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems
- C-6 Requirements for the Safety Analysis of CANDU Nuclear Power Plants
- C-006 Safety Analysis of CANDU Nuclear Power Plants (Rev. 1)

ANNEX 7.3
Site Acceptance, Construction Approval and Commissioning of Power Reactors

SITE ACCEPTANCE

At the site acceptance stage, the CNSC 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 CNSC will not issue a site approval or Site Preparation Licence unless an environmental assessment has been completed as required by the CEAA. If the environmental assessment concludes that further investigation is needed, or if public concerns about the project warrant, the responsible authority 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 responsible authority for action or whether further investigation is required.

The CNSC 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 CNSC 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; and
- local geology.

During this phase, the CNSC 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 CNSC 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.

CONSTRUCTION APPROVAL

Before it grants construction approval, the CNSC must be assured that the site design will meet CNSC 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 description of the proposed design of the nuclear facility, including the manner in which the physical and environmental characteristics of the site are taken into account in the design;
- a description of the environmental baseline characteristics of the site and the surrounding area;
- the proposed construction program, including its schedule;
- a description of the structures proposed to be built as part of the nuclear facility, including their design and their design characteristics;
- a description of the systems and equipment proposed to be installed at the nuclear facility, including their design and their design operating conditions;
- 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;
- preliminary plans for operation;
- conceptual plan for decommissioning the plant;
- a description of an overall quality assurance program for the project together with specific quality assurance programs for design, procurement, manufacture, construction and installation and commissioning;
- the proposed measures to facilitate Canada's compliance with any applicable safeguards agreement;
- the effects on the environment and the health and safety of persons that may result from the construction, operation and decommissioning of the nuclear facility, and the measures that will be taken to prevent or mitigate those effects;
- the proposed location of points of release, the proposed maximum quantities and concentrations, and the anticipated volume and flow rate of releases of nuclear substances and hazardous substances into the environment, including their physical, chemical and radiological characteristics;
- the proposed measures to control releases of nuclear substances and hazardous substances into the environment;
- the proposed program and schedule for recruiting, training and qualifying workers in respect of the operation and maintenance of the nuclear facility; and
- a description of any proposed full-scope training simulator for the nuclear facility.

Construction will only be authorized after the design and safety analysis programs have progressed to the point that, in the judgement of the CNSC, 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 CNSC 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 CNSC demands analyses of enough postulated accidents in adequate detail in order to ensure that all major safety design requirements have been identified and 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 (for example, shutdown systems, emergency core cooling systems and containment system) will be met under all normal and upset conditions. These standards are defined in Regulatory Documents R-7, R-8, R-9. In practice, this requires that the applicant consider 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 need 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 CNSC reviews the information in the preliminary safety assessment report 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 CNSC relies on experience from previous licensing reviews. The CNSC also takes account of any unusual or novel design features in deciding the topics that require in-depth examination.

In addition to reviewing the design and safety analysis information included in the application, the CNSC 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 Commission who makes the final decision on approval of construction.

During construction of the plant, the CNSC 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 the primary coolant system and 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 CNSC assessment reports. The CNSC has a formal documentation system to track the licensee's response and the final disposition of directives and actions arising from these audits.

COMMISSIONING

The CNSC 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 CNSC's direct involvement in commissioning concentrates on a few major tests that are considered particularly important to safety. More information on the commissioning process is provided in Article 19.

ANNEX 7.4
List of Program Descriptions Required to Support a Nuclear Power Reactor Operating Licence Application

- Chemistry Control Program
- Community Relations Program
- Configuration Management and Change Control Program
- Corrective Action and Operating Experience Program
- Decommissioning Plan and Financial Guarantees
- Effluent and Environmental Monitoring Program
- Environmental Protection Program
- Emergency Preparedness Program
- Environmental Qualification Program
- Fire Protection Program
- Human Factors Program
- Maintenance Program
- Nuclear Substance Control Program
- Occupational Health and Safety Program
- Organization Staffing and Training Program
- Periodic and In-Service Inspection Program
- Quality Assurance Program
- Plant Life Assurance Program
- Radiation Protection Program
- Safeguards Program
- Safety Report and Safety Analysis Program
- Security Program
- Station Improvement Program
- System Testing Program
- Technical Surveillance and Reporting Program
- Waste Management Program

ANNEX 7.5
Sample Power Reactor Operating Licence

Nuclear Power Reactor Operating Licence
Pickering Nuclear Generating Station B

Unless otherwise provided for in this licence, words and expressions used in this licence have the same meaning as in the *Nuclear Safety and Control Act* and associated Regulations.

I) LICENCE NUMBER: PROL 08.00/2003

II) LICENSEE: Pursuant to section 24 of the *Nuclear Safety and Control Act* this licence is issued to:

Ontario Power Generation Inc.
700 University Avenue
Toronto, Ontario M5G 1X6

III) LICENCE PERIOD: This licence is valid from April 1, 2001 to June 30, 2003, unless suspended, amended, revoked or replaced.

IV) LICENSED ACTIVITIES:

This licence authorizes the licensee to:

(i) operate the Pickering Nuclear Generating Station B (hereinafter "the nuclear facility") at a site located in the Town of Pickering in the regional municipality of Durham, Province of Ontario, and as described in the documents listed in Part I of Appendix A to this licence;

(ii) possess, transfer, use, package, manage and store the nuclear substances, other than sealed and unsealed sources and approved devices containing nuclear substances, that are required for, associated with, or arise from the activities described in (i);

(iii) possess, use, manage and store enriched uranium as required for fission chambers for the Pickering Nuclear Generating Station A Shutdown System Enhancement, including spares; and

(iv) possess and use prescribed equipment and prescribed information that are required for, associated with, or arise from the activities described in (i).

V) CONDITIONS:

1. General

1.1 The contents of the appendices attached to this licence form part of the licence.

1.2 Unless otherwise indicated in this licence, the licensee may make any change to the licensee documents listed or referred to in Part III of Appendix A provided the changes are made in accordance with the licensee's change control procedures.

1.3 Unless otherwise indicated in this licence, operation of the facility shall conform with the documents listed in Part III of Appendix A, as revised in accordance with condition 1.2.

1.4 Unless otherwise indicated in this licence, the licensee shall not make any change to any of the documents listed in Appendix B without the prior written approval of the Canadian Nuclear Safety Commission (hereinafter «the Commission»), or a person authorized by the Commission.

Amendment of the licence is not required prior to the implementation by the licensee of a proposed change to a document listed in Appendix B that has been approved in writing by the Commission, or by a person authorized by the Commission. An approved change is deemed to be part of this licence.

1.5 The licensee shall control the use and occupation of lands situated within the exclusion zone (meaning any land within 914 metres of any reactor building) as described in the Safety Report, listed in Appendix A, and shall not change the use or occupation of those lands without the prior written approval of the Commission, or a person authorized by the Commission.

1.6 The licensee shall report in accordance with regulatory document R-99 entitled: REPORTING REQUIREMENTS FOR OPERATING NUCLEAR POWER FACILITIES, listed in Appendix C.

1.7 The licensee shall provide, at the nuclear facility in respect of which this licence is issued and at no expense to the Commission, office space for employees of the Commission who customarily carry out their functions on the premises of that nuclear facility (on-site Commission staff). The licensee shall keep the office space of on-site Commission staff separate from the remainder of the building in which it is located by walls, partitions or other suitable structures.

2. Staffing and Organization

2.1 The licensee shall not make any change to the following role documents without the prior written approval of the Commission, or a person authorized by the Commission:

Title	Document Identifier Number
Site Vice-President	N-MAN-08131-10000 AECB-001 R01
Director, Operations and Maintenance	N-MAN-08131-10000 AECB-004 R00
Shift Manager (Transitional)	N-MAN-08131-10000 AECB-007 R00
Control Room Shift Operating Supervisor (Transitional)	N-MAN-08131-10000 AECB-009 R00
Authorized Nuclear Operator	N-MAN-08131-10000 AECB-010 R00
Manager, Performance Assurance	N-MAN-08131-10000 AECB-014 R02
Section Manager, Radiation Protection Programs	N-MAN-08131-10000 AECB-016 R00
Site Radiation Protection Manager	N-MAN-08131-10000 AECB-017 R00

2.2 The licensee shall maintain the staff complement at the nuclear facility, including the minimum nuclear facility shift complement, as specified in the document entitled: PICKERING OPERATIONS DEPARTMENT PROCEDURE - STATION SHIFT COMPLEMENT, listed in Appendix B.

Unless otherwise approved in writing by the Commission, or by a person authorized by the Commission, there shall be in the nuclear facility at all times at least four authorized nuclear operators, one control room shift operating supervisor and one shift manager. At all times there shall be in the main control room a minimum of two persons who have been either certified under the *Nuclear Safety and Control Act*, or authorized prior to May 31, 2000 by the Atomic Energy Control Board (hereinafter “the Board”) or by a person authorized by the Board, and who are qualified to operate the controls of the reactors and their systems. In addition, there shall be at all times, for each reactor unit, an authorized nuclear operator or a supervised control panel operator in direct attendance at the reactor unit’s main control room control panels. Any supervised control panel operator attending a reactor unit’s control panels must be continuously under the supervision of an authorized nuclear operator present in the main control room.

The minimum personnel requirements for the main control room that this condition imposes do not apply where this minimum cannot be met due to emergency conditions that could cause an unwarranted hazard to personnel in the main control room, in which case the licensee shall place the reactors in an assured shutdown and safe condition.

2.3 Any person appointed by the licensee to the function of responsible health physicist at the nuclear facility must either hold a certification issued under the *Nuclear Safety and Control Act*, or have been approved in writing prior to May 31, 2000 by the Board or by a person authorized by the Board.

Any person appointed to this function at the nuclear facility shall not delegate the authority or responsibilities of the function, in accordance with the document entitled: ONTARIO POWER GENERATION RADIATION PROTECTION REQUIREMENTS, listed in Appendix B, except to another individual who holds either a certification issued under the *Nuclear Safety and Control Act*, or who has been approved in writing prior to May 31, 2000 by the Board or by a person authorized by the Board.

When applying for certification or renewal of certification of a person for this function, the licensee shall confirm that the person meets the applicable requirements specified in Appendix D.

2.4 Any person appointed by the licensee to any of the following operating positions at the nuclear facility must hold either a certification issued under the *Nuclear Safety and Control Act*, or an authorization issued prior to May 31, 2000 by the Board or by a person authorized by the Board:

- shift manager
- control room shift operating supervisor
- authorized nuclear operator.

When applying for certification of a person for one of these operating positions, the licensee shall confirm that the person meets the applicable requirements specified in Appendix E.

2.5 The licensee shall ensure that the incumbents of the operating positions referred to in paragraph 2.4 receive the continuing training specified in Appendix F.

2.6 The licensee shall establish and document the required initial and continuing training programs to address the training requirements referred to in Appendices D, E and F. These programs shall be in accordance with the principles of a systematic approach to training.

3. <u>Operations</u>

3.1 The licensee shall operate and maintain the nuclear facility according to the methods and procedures, for the purposes, and within the limits described in the document entitled: PICKERING NGS-B OPERATING POLICIES AND PRINCIPLES, listed in Appendix B.

3.2 The licensee shall ensure that:

a) the total power generated in any one fuel bundle does not exceed 750 kilowatts;
b) the total power generated in any fuel channel does not exceed 6100 kilowatts under normal steady-state operating conditions; and
c) the total thermal power from the reactor fuel does not exceed 1744 megawatts under normal steady-state operating conditions.

3.3 The licensee shall maintain the trip setpoints in shutdown system one and shutdown system two at values that are approved in writing by the Commission, or by a person authorized by the Commission.

3.4 The licensee shall establish and implement a quality assurance program that conforms to the requirements of the following CSA Standards:

a) CAN/CSA-N286.0-92: Overall Quality Assurance Program Requirements for Nuclear Power Plants;
b) CAN3-N286.1-00: Procurement Quality Assurance for Nuclear Power Plants;
c) CAN3-N286.2-00: Design Quality Assurance for Nuclear Power Plants;
d) CAN3-N286.3-99: Construction Quality Assurance for Nuclear Power Plants;
e) CAN/CSA-N286.4-M86: Commissioning Quality Assurance for Nuclear Power Plants;
f) N286.5-95: Operations Quality Assurance for Nuclear Power Plants; and
g) N286.7-99: Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants.

3.5 For the purpose of limiting, during the lifetime of the nuclear facility, the risks related to the failure or unavailability of any structure, system or component whose performance may affect the safe operation or security of the nuclear facility, the licensee shall establish, document and implement a maintenance program.

The maintenance program shall include testing and inspection and shall be of such quality and be performed in such a manner that the availability, reliability and effectiveness of any structure, system or component remain consistent with the design and analysis documents listed in Part III of Appendix A.

3.6 Unless otherwise approved in writing by the Commission, or by a person authorized by the Commission, the licensee shall carry out tests to measure the rate of leakage from the reactor buildings when subjected to full design pressure:

before June 30, 2006, for unit 5;
before June 30, 2005, for unit 6;
before December 31, 2006, for unit 7;
before December 31, 2005, for unit 8.

3.7 Unless otherwise approved in writing by the Commission, or by a person authorized by the Commission, the licensee shall inspect the internal structures of and components within the vacuum building at least once during any period of ten years. The next inspection shall be carried out before December 31, 2010.

4. Design Modifications and Operational Change

4.1 The licensee shall not make any change to the reactor shutdown system one, shutdown system two, the containment system, the emergency core cooling system or associated systems necessary for the proper operation of these systems, that would render inaccurate the descriptions and analyses listed in the documents in Part III of Appendix A, without the prior written approval of the Commission, or a person authorized by the Commission.

4.2 The licensee shall not make any change to any equipment or procedure that could result in hazards or risks different in nature or greater in probability or magnitude than those stated in the design and analysis documents listed in Part III of Appendix A, without the prior written approval of the Commission, or a person authorized by the Commission.

4.3 Unless otherwise indicated in writing by the Commission, or by a person authorized by the Commission, prior to loading any fuel bundle or fuel assembly into a reactor, the licensee shall obtain the approval of the Commission, or a person authorized by the Commission, for the design of that bundle or assembly.

5. Pressure Boundaries

For the purpose of the following conditions, “registered”, “accepted” and “approval” means either by the Commission, or by a person authorized by the Commission, or by an authority identified by the Commission for that purpose.

5.1 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.

5.2 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 over-pressure 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.

5.3 The licensee shall keep records of regulatory approvals and other documents required under 5.1, 5.2 and the standards applicable to the work or equipment.

5.4 In addition to the reporting requirements set out in regulatory document R-99 entitled: REPORTING REQUIREMENTS FOR OPERATING NUCLEAR POWER FACILITIES, the licensee shall report promptly to the Commission when it learns of any failure of a pressure boundary that has caused injury, death, or property damage.

6. Fire Protection

6.1 The licensee shall design, build, modify and otherwise carry out work related to the nuclear facility with potential to impact protection from fire in accordance with the National Building Code, 1995, the CSA standard N293-95: Fire Protection for CANDU Nuclear Plants, and appendices A to D of that standard, entitled:

- A: Fire Protection Plan
- B: Fire Separation for Protection of Safety Systems
- C: Fire Hazard Assessments
- D: Reviewing Fire Protection for Existing CANDU Plants

6.2 The licensee shall operate, maintain, test and inspect the nuclear facility in accordance with the National Fire Code, 1995, CSA standard N293-95 and appendices A to E of that standard, inclusive.

6.3 The licensee shall, prior to implementation, submit any proposed modification of the nuclear facility with potential to impact protection from fire, for third-party review of compliance with 6.1 and the standards listed therein. The review shall be carried out by one or more independent external agencies having specific expertise with such reviews and shall be submitted, by the licensee, to the Commission, or a person authorized by the Commission.

6.4 The licensee shall arrange for annual third-party reviews, by one or more independent external agencies having specific expertise with such reviews, to review compliance with the following:

- inspection requirements of the National Fire Code, 1995; and
- inspection and audit requirements of CSA standard N293-95, Appendix E.

6.5 In the event of any conflict or inconsistency between a nuclear safety requirement under the licence and CSA standard N293-95 or the National Building Code, 1995 or the National Fire Code, 1995, the licensee shall direct the conflict or inconsistency to the Commission, or a person authorized by the Commission, for resolution.

7. Environmental Qualification of Equipment

7.1 By June 30, 2004, the licensee shall establish, that all required systems, equipment, components, protective barriers and structures in the nuclear facility are qualified to perform their safety functions under the environmental conditions defined by the nuclear facility's design-basis accidents.

8. **Radiation and Environmental Protection**

8.1 The licensee shall operate and maintain the nuclear facility according to the methods and procedures, for the purposes and within the limits described in the documents entitled: RADIATION PROTECTION POLICIES AND PRINCIPLES and RADIATION PROTECTION REQUIREMENTS - NUCLEAR FACILITIES, listed in Appendix B.

8.2 The licensee shall have in place for the nuclear facility a radiation emergency plan, as described in the document entitled: ONTARIO POWER GENERATION INC. CONSOLIDATED NUCLEAR EMERGENCY PLAN, listed in Appendix B.

8.3 The licensee shall control, monitor and record releases of radioactive prescribed substances from the nuclear facility, and such releases shall not exceed the limits identified in the document entitled: THE INTERIM DERIVED RELEASE LIMITS FOR PICKERING NUCLEAR GENERATING STATION B, listed in Appendix B.

9. **Safeguards**

9.1 The licensee shall take all necessary measures to facilitate Canada's compliance with any applicable safeguards agreement.

9.2 The licensee shall provide the International Atomic Energy Agency, an International Atomic Energy Agency inspector, or a person acting on behalf of the International Atomic Energy Agency, with such reasonable services and assistance as are required to enable the International Atomic Energy Agency to carry out its duties and functions pursuant to a safeguards agreement.

9.3 The licensee shall grant prompt access at all reasonable times to all locations at the nuclear facility to an International Atomic Energy Agency inspector, or to a person acting on behalf of the International Atomic Energy Agency, where such access is required for the purposes of carrying on an activity pursuant to a safeguards agreement. In granting access, the licensee shall provide health and safety services and escorts as required in order to facilitate activities pursuant to a safeguards agreement.

9.4 The licensee shall disclose to the Commission, to the International Atomic Energy Agency, or to an international Atomic Energy Agency inspector, any records that are required to be kept or any reports that are required to be made under a safeguards agreement.

9.5 The licensee shall provide such reasonable assistance to an International Atomic Energy Agency inspector, or to a person acting on behalf of the International Atomic Energy Agency, as is required to enable sampling and removal or shipment of samples required pursuant to a safeguards agreement.

9.6 The licensee shall provide such reasonable assistance to an International Atomic Energy Agency inspector, or to a person acting on behalf of the International Atomic Energy Agency, as is required to enable measurements, tests and removal or shipment of equipment required pursuant to a safeguards agreement.

9.7 The licensee shall, at the request of the Commission, or a person authorized by the Commission, install safeguards equipment at the nuclear facility.

9.8 The licensee shall permit an International Atomic Energy Agency inspector, or a person acting on behalf of the International Atomic Energy Agency, to service safeguards equipment at the nuclear facility.

9.9 The licensee shall operate safeguards equipment at the nuclear facility in accordance with the methods and procedures specified by the International Atomic Energy Agency.

9.10 The licensee shall provide the services required for the operation of the safeguards equipment at the nuclear facility, in accordance with the specifications of the International Atomic Energy Agency.

9.11 The licensee shall not interfere with or interrupt the operation of safeguards equipment at the nuclear facility, or alter, deface or break a safeguards seal, except pursuant to a safeguards agreement.

9.12 The licensee shall implement measures to prevent damage to, or the theft, loss or sabotage of safeguards equipment or samples collected pursuant to a safeguards agreement or the illegal use, possession, operation or removal of such equipment or samples.

9.13 The licensee shall make such reports and provide such information to the Commission as are required to facilitate Canada's compliance with any applicable safeguards agreement.

9.14 The licensee shall not, except with the prior written approval of the Commission, or a person authorized by the Commission, make changes to any aspect of the nuclear facility, nuclear facility operation, nuclear facility equipment or procedures that would affect the implementation of safeguards measures.

9.15 The licensee shall make and submit reports to the Commission in accordance with the document AECB-1049 entitled: REPORTING REQUIREMENTS FOR FISSIONABLE AND FERTILE SUBSTANCES on the inventory and transfer of fissionable and fertile substances, as indicated in regulatory document R-99 entitled: REPORTING REQUIREMENTS FOR OPERATING NUCLEAR POWER FACILITIES.

10. Security

10.1 The licensee shall maintain the nuclear security guard staff complement at the nuclear facility as specified in the document entitled: SITE SECURITY REPORT, listed in Appendix B.

11. Nuclear Facility-Specific

11.1 If the status of Pickering Nuclear Generating Station B is such that only units 7 and 8 are operating, one emergency coolant injection pump shall be run continuously in the recirculation mode. Operation in this mode for longer than four weeks shall require the prior written approval of the Commission, or a person authorized by the Commission.

If the status of Pickering Nuclear Generating Station B is such that only one unit is operating, that unit shall be shut down in a controlled manner and placed into a guaranteed shutdown state unless a second unit can be restarted within four days. Operation in this mode beyond four days shall require the prior written approval of the Commission, or a person authorized by the Commission.

DATED at OTTAWA this _____ day of March, 2001.

Linda J. Keen
President
CANADIAN NUCLEAR SAFETY COMMISSION

APPENDIX A

Location, Description, Operation, Design and Analyses of the Nuclear Facility

PART I: LOCATION of the nuclear facility:

The nuclear facility is located in the Town of Pickering in the regional municipality of Durham, Province of Ontario and described in:

i) Pickering N.G.S. Site Plan, drawing number NK30-DOA-10200-0001, Rev. 7, September 1992; and

ii) Pickering GS Site Security Taut-Wire Fence Layout and Survey, drawing number 8690-DOH-14100-1003, Rev. 5, dated September 1992.

PART II: DESCRIPTION of the nuclear facility:

The nuclear facility is described in:

i) Pickering Generating Station B Safety Report:
   - Part 1 & 2, revised October 2000;
   - Volume 2, revised February 1999; and

ii) The application for this licence entitled: Application for Renewal of Pickering Nuclear Generating Station B Power Reactor Operating Licence, dated August 29, 2000, submitted by Ontario Power Generation.

PART III: OPERATION, DESIGN AND ANALYSIS of the nuclear facility:

The operation, design and analysis of the nuclear facility is described in:

i) Pickering Generating Station B Safety Report:
   - Part 1 & 2, revised October 2000
   - Volume 2 (safety analysis), revised February 1999

ii) The programs referred to in the application for this licence entitled: Application for Renewal of Pickering Nuclear Generating Station B Power Reactor Operating Licence, dated August 29, 2000, submitted by Ontario Power Generation.

APPENDIX B

Documents Prepared by the Licencee

Unless otherwise indicated, any change to any of the documents listed below requires the prior approval of the Canadian Nuclear Safety Commission, or a person authorized by the Commission.

Licence Condition

2.1 DOCUMENT IDENTIFIER NUMBER AND TITLE:

N-MAN-08131-10000 AECB-001 R01, Site Vice-President
N-MAN-08131-10000 AECB-004 R00, Director, Operations and Maintenance
N-MAN-08131-10000 AECB-007 R00, Shift Manager (Transitional)
N-MAN-08131-10000 AECB-009 R00, Control Room Shift Operating Supervisor (Transitional)
N-MAN-08131-10000 AECB-010 R00, Authorized Nuclear Operator
N-MAN-08131-10000 AECB-014 R02, Manager, Performance Assurance
N-MAN-08131-10000 AECB-016 R00, Section Manager, Radiation Protection Programs
N-MAN-08131-10000 AECB-017 R00, Site Radiation Protection Manager

2.2 Pickering Operations Department Procedure, P-ODP-2-7, Station Shift Complement, November 2000

3.1 Pickering NGS-B, Operating Policies and Principles, Rev. 18, April 2000

8.1 Radiation Protection Policies and Principles, N-RPP-03415.1-10000-R05, December 2000; and Radiation Protection Requirements - Nuclear Facilities, N-RPP-03415-1-10001-R06, September 2000

8.2 Ontario Power Generation Inc. Consolidated Nuclear Emergency Plan, N-PROG-RA-0001, R02

8.3 The Interim Derived Release Limits for Pickering Nuclear Generating Station B, NK30-REP-03482-10001-R00, December 1999

10.1 Pickering Nuclear - Security Report, Rev. 5, January 1999

APPENDIX C

Regulatory Documents Prepared by the Canadian Nuclear Safety Commission

Licence Condition

1.6 AECB Regulatory Document R-99: Reporting Requirements for Operating Nuclear Power Facilities, January 1995

APPENDIX D

Certification Requirements for Responsible Health Physicists

1 Qualification Requirements

1.1 A responsible health physicist shall, at the time of certification, meet the requirements specified in paragraphs 1.1.1 to 1.1.3.

1.1.1 EDUCATION: Baccalaureate degree in engineering or science from a recognized university, with additional specialized courses on current radiation protection principles, methods and practices related to the operation of a nuclear power reactor.

1.1.2 EXPERIENCE: Minimum of 4 years related experience in a nuclear facility. At least 2 years of this experience must be at a nuclear power reactor of which at least 6 months must be at the nuclear facility. At least 1 year of the total 4 years related experience must be in a supervisory position.

1.1.3 TRAINING: As specified in section 2.

2 Initial Training Requirements

2.1 A responsible health physicist shall meet the requirements specified in paragraphs 2.1.1 and 2.1.2.

2.1.1 Have successfully completed training appropriate to the knowledge requirements of the responsible health physicist covering:

- the relevant provisions of the *Nuclear Safety and Control Act* (NSC Act)
- Regulations made pursuant to the NSC Act and, specifically:
  - *General Nuclear Safety and Control Regulations*
  - *Radiation Protection Regulations*
  - *Class I Nuclear Facilities Regulations*
  - *Nuclear Substances and Radiation Devices Regulations*
  - *Packaging and Transport of Nuclear Substances Regulations*
- safety culture
- responsibilities and authority of the responsible health physicist
- responsibilities and authority of persons who interact with the responsible health physicist
- facility's operating licence, including documents referenced in the licence
- licensee's and facility's policies, standards and procedures
- facility design, operation and maintenance

2.1.2 Have successfully completed a formal interview by facility management which confirms the person's competence to perform the duties of the responsible health physicist.

3 Commission Examinations for Initial Certification

3.1 A responsible health physicist shall, at the time of certification, have successfully completed an interview by Commission staff which samples the topics specified in paragraph 2.1.1 and current radiation protection principles, methods and practices related to the operation of a nuclear power reactor.

4 Responsible Health Physicist Transferring to the Nuclear Facility

4.1 A person, who either holds a certification as responsible health physicist or who has been approved as such prior to May 31, 2000 by the Board or by a person authorized by the Board at another nuclear power reactor, seeking certification for the same function at the nuclear facility shall, at the time of certification, meet the requirements specified in paragraphs 4.1.1 to 4.1.3.

4.1.1 Have successfully completed training appropriate to the knowledge requirements of the responsible health physicist covering:

- responsibilities and authority of the responsible health physicist
- responsibilities and authority of persons who interact with the responsible health physicist
- facility’s operating licence, including documents referenced in the licence
- licensee’s and facility’s policies, standards and procedures
- facility design, operation and maintenance

4.1.2 Have successfully completed a formal interview by facility management which confirms the person’s competence to perform the duties of the responsible health physicist.

4.1.3 Have successfully completed an interview by Commission staff which samples the topics specified in paragraph 4.1.1 and current radiation protection principles, methods and practices related to the operation of a nuclear power reactor.

5 Continuing Training Requirements

5.1 A responsible health physicist shall successfully complete training appropriate to the knowledge requirements of the function covering topics identified as a result of:

- changes to facility systems and equipment
- changes to licensee’s and facility’s policies, standards and procedures
- changes to regulatory requirements
- changes to the facility’s operating licence or to documents referenced in the licence
- facility and industry experience and operating events

6 Requalification Tests

6.1 A responsible health physicist seeking renewal of certification shall have successfully completed an interview by Commission staff which samples the topics specified in paragraph 2.1.1 and current radiation protection principles, methods and practices related to the operation of a nuclear power reactor, within six months prior to the expiry date of the person’s certification.

APPENDIX E

Certification Requirements for New Authorized Nuclear Operators, Control Room Shift Operating Supervisors and Shift Managers

1 Qualification Requirements

Authorized Nuclear Operators

1.1 An authorized nuclear operator shall, at the time of certification, meet the requirements specified in paragraphs 1.1.1 to 1.1.3.

1.1.1 **EDUCATION:** High school diploma that includes credits in science and mathematics.

1.1.2 **EXPERIENCE:** Minimum of 5 years of experience at a Canadian nuclear power reactor, with at least 2 years of this experience at the nuclear facility.

1.1.3 **TRAINING:** As specified in section 2.

Control Room Shift Operating Supervisors

1.2 A control room shift operating supervisor shall, at the time of certification, meet the requirements specified in paragraphs 1.2.1 to 1.2.3.

1.2.1 **EDUCATION:** High school diploma that includes credits in science and mathematics.

1.2.2 **EXPERIENCE:** Minimum of 4 years of experience as authorized nuclear operator at the nuclear facility. At least 1 year of the experience as authorized nuclear operator must be immediately prior to the selection for training for the position.

1.2.3 TRAINING: As specified in section 2.

Shift Managers

1.3 A shift manager shall, at the time of certification, meet the requirements specified in paragraphs 1.3.1 to 1.3.3.

1.3.1 **EDUCATION:** Baccalaureate in engineering or science from a recognized university. A certificate of qualification as stationary engineer second class, or certification as reactor operator with 4 years of experience in this position at a Canadian nuclear power reactor, are acceptable alternatives to a university degree.

1.3.2 **EXPERIENCE:** Minimum of 6 years experience at a Canadian nuclear power reactor, with at least 2 years of this experience at the nuclear facility.

1.3.3 **TRAINING:** As specified in section 2.

2 Initial Training Requirements

Authorized Nuclear Operators

2.1 An authorized nuclear operator shall meet the requirements specified in paragraphs 2.1.1 to 2.1.8.

2.1.1 Have successfully completed training appropriate to the knowledge requirements of the position covering:

- science fundamentals relevant to the operation of the facility
- principles of operation of facility equipment

This training shall be followed by a comprehensive written examination set by the licensee. Credit for this examination must be obtained before the person may take the Commission examination referred to in paragraph 3.1.1.

2.1.2 Have successfully completed training appropriate to the knowledge requirements of the position covering:

- radiation fundamentals
- radiation hazards
- radiation protection theory and practices
- radiation protection procedures used during normal, abnormal and emergency operation of the facility

This training shall be followed by a comprehensive written examination set by the licensee. Credit for this examination must be obtained before the person may take the Commission examination referred to in paragraph 3.1.2 and, subject to the provisions of section 4, within three years prior to certification.

2.1.3 Have successfully completed training appropriate to the knowledge requirements of the position covering:

- design and operation of facility systems
- facility systems integrated operation, including interaction between unit systems and those of the other units
- administrative procedures related to facility operation and maintenance
- principles of reactor fuelling, fuelling limitations, fuel handling and storage, irradiated fuel cooling
- principles of nuclear safety
- responsibilities of the authorized nuclear operator

This training shall be followed by a comprehensive written examination set by the licensee. Credit for this examination must be obtained before the person may take the Commission examination referred to in paragraph 3.1.2.

2.1.4 Have successfully completed, on the facility full scope replica simulator, training appropriate to the knowledge and skill requirements of the position covering:

- operation of unit systems and equipment under normal, abnormal and emergency conditions and the effects that unit operation may have on the other units
- interactions with other members of the shift crew

This training shall be followed by a comprehensive simulator-based examination set by the licensee.

2.1.5 Have successfully completed on-the-job training appropriate to the knowledge and skill requirements of the position covering:

- standard control room operating practices
- maintenance and repair of unit systems and equipment
- operations in the control equipment room

- operation of unit systems from the unit emergency control centre

This training shall include job performance measures to confirm that the person has the required knowledge and skills.

2.1.6 Have satisfactorily performed the duties of an authorized nuclear operator under the supervision of a certified or authorized incumbent of the position for a minimum of 480 hours on shift after having met the requirements specified in paragraphs 2.1.1 to 2.1.5.

2.1.7 Have successfully completed a formal interview by facility management which confirms the person's competence to perform the duties of the authorized nuclear operator.

2.1.8 The person shall meet the requirements specified in paragraphs 2.1.1 to 2.1.7 before taking the Commission examination referred to in paragraph 3.1.3.

Control Room Shift Operating Supervisors

2.2 A control room shift operating supervisor shall meet the requirements specified in paragraphs 2.2.1 to 2.2.5.

2.2.1 Have successfully completed training appropriate to those knowledge requirements of a control room shift operating supervisor which are in addition to those of an authorized nuclear operator, covering:

- fuelling strategies, properties of irradiated fuel and physics of fuel failures
- primary and back-up heat sinks
- conventional and radiation hazards to facility personnel and to the public, including radiological hazards from postulated accident conditions
- handling of conventional and radiation emergencies
- expected response of facility systems and units to equipment failures and accident conditions
- operating strategies
- configuration of systems and equipment isolation required for maintenance activities
- design and operation of facility systems for which the authorized nuclear operators do not have direct operational control
- facility's operating licence and documents referenced in the licence
- licensee's policies, standards and procedures
- situations that may result in the violation of conditions in the facility's operating licence and Operating Policies and Principles
- requirements pertaining to facility operation in Federal and Provincial Acts and Regulations, and in relevant standards and codes
- responsibilities and authority of a control room shift operating supervisor and of other facility personnel who report to or interface with the control room shift operating supervisor
- qualification requirements of facility personnel who report to the control room shift operating supervisor

This training shall be followed by a comprehensive written examination set by the licensee. Subject to the provisions of section 4, credit for this examination must be obtained within 1 year prior to certification.

2.2.2 Have successfully completed, on the facility full scope replica simulator, training appropriate to the knowledge and skill requirements of the position, covering:

- operation and monitoring of facility systems and equipment for which the authorized nuclear operators do not have direct operational control, under normal, abnormal and emergency conditions
- independent monitoring of facility systems and equipment under normal, abnormal and emergency conditions

- independent diagnosis and decision making
- supervision and direction of facility operations under normal, abnormal and emergency conditions
- interaction with other members of the shift crew

This training shall be followed by a comprehensive simulator-based examination set by the licensee. Subject to the provisions of section 4, credit for this examination must be obtained within 1 year prior to certification.

2.2.3 Have successfully completed on-the-job training appropriate to the knowledge and skill requirements of the position, covering:

- operation and monitoring of systems and equipment performed by the fuel handling operators under normal, abnormal and emergency conditions
- supervision and direction of operations in the control room, in the control equipment rooms, in the unit emergency control centres and in the field under normal, abnormal and emergency conditions
- maintenance and repair of facility systems and equipment

This training shall include comprehensive job performance measures to confirm that the person has the required knowledge and skills.

2.2.4 Have satisfactorily performed the duties of a control room shift operating supervisor under the supervision of a certified or authorized incumbent of the position for a minimum of 480 hours on shift after having met the requirements specified in paragraphs 2.2.1 to 2.2.3.

2.2.5 Have successfully completed a formal interview by facility management which confirms the person’s competence to perform the duties of the control room shift operating supervisor.

Shift Managers

2.3 A shift manager shall meet the requirements specified in paragraphs 2.3.1 to 2.3.7.

2.3.1 Have successfully completed the training and the examinations for authorized nuclear operators specified in paragraphs 2.1.1, 2.1.2 and 2.1.3.

2.3.2 Have successfully completed training appropriate to those knowledge requirements of the shift manager which are in addition to those of an authorized nuclear operator covering:

- reactor physics, principles of reactor operation and fuelling strategies
- phenomena that may significantly affect core reactivity and flux shape
- properties of irradiated fuel, principles of fuel cooling and physics of fuel failures
- operating constraints and limits associated with reactor fuelling and irradiated fuel cooling
- reactor safety, heat transfer mechanisms and fluid mechanics
- primary and back_up heat sinks
- conventional and radiation hazards to facility personnel and to the public, including radiological hazards from postulated accident conditions
- handling of conventional and radiation emergencies
- design requirements of safety related equipment and systems
- design features and limitations of facility equipment and systems
- chemical control of systems
- diagnosis of equipment failures and assessment of abnormal facility conditions
- expected response of facility systems and units to equipment failures and accident conditions
- operating strategies
- major assumptions in the facility accident analyses and technical bases for emergency operating procedures
- configuration of systems and equipment isolation required for maintenance activities

- design and operation of facility systems for which the authorized nuclear operators do not have direct operational control
- facility’s operating licence and documents referenced in the licence
- licensee’s policies, standards and procedures
- situations that may result in the violation of conditions in the facility’s operating licence and Operating Policies and Principles
- requirements pertaining to facility operation in Federal and Provincial Acts and Regulations, and in relevant standards and codes
- responsibilities and authority of a shift manager and of other facility personnel who report to or interface with the shift manager
- qualification requirements of facility personnel who report to the shift manager

This training shall be followed by a comprehensive written examination set by the licensee. Credit for this examination must be obtained before the person may take the Commission examination referred to in paragraph 3.2.3.

2.3.3 Have successfully completed, on the facility full scope replica simulator, training appropriate to the knowledge and skill requirements of the position covering:

- operation and monitoring of facility systems and equipment performed by the operators under normal, abnormal and emergency conditions
- independent monitoring of facility systems and equipment under normal, abnormal and emergency conditions
- independent diagnosis and decision making
- supervision and direction of facility operations under normal, abnormal and emergency conditions
- interactions with other members of the shift crew

This training shall be followed by a comprehensive simulator-based examination set by the licensee.

2.3.4 Have successfully completed on-the-job training appropriate to the knowledge and skill requirements of the position covering:

- standard control room operating practices
- operation and monitoring of systems and equipment performed by the fuel handling operators under normal, abnormal and emergency conditions
- supervision and direction of operations in the control room, in the control equipment rooms, in the unit emergency control centres and in the field, under normal, abnormal and emergency conditions
- maintenance and repair of facility systems and equipment

This training shall include job performance measures to confirm that the person has the required knowledge and skills.

2.3.5 Have satisfactorily performed the duties of a shift manager under the supervision of a certified or authorized incumbent of the position for a minimum of 480 hours on shift after having met the requirements specified in paragraphs 2.3.1 to 2.3.4.

2.3.6 Have successfully completed a formal interview by facility management which confirms the person’s competence to perform the duties of the shift manager.

2.3.7 The person shall meet the requirements specified in paragraphs 2.3.1 to 2.3.6 before taking the Commission examination referred to in paragraph 3.2.4.

3 Commission Examinations for Initial Certification

Authorized Nuclear Operators

3.1 An authorized nuclear operator shall, at the time of certification, meet the requirements specified in paragraphs 3.1.1 to 3.1.3

3.1.1 Have successfully completed the Commission examination for reactor operators which samples topics covered in the training referred to in paragraph 2.1.1. Credit for this examination must be obtained before taking the examination referred to in paragraph 3.1.2 and, subject to the provisions of section 4, within three years prior to certification.

3.1.2 Have successfully completed the Commission examination for reactor operators which samples topics covered in the training referred to in paragraph 2.1.3 and those aspects of unit operation, both normal and abnormal, which may result in the discharge of radioactivity to the environment, or which could affect the safety of facility personnel or of members of the public. Credit for this examination must be obtained before taking the examination referred to in paragraph 3.1.3 and, subject to the provisions of section 4, within two years prior to certification.

3.1.3 Have successfully completed the Commission simulator-based examination for reactor operators covering unit operations under abnormal and emergency conditions. Subject to the provisions of section 4, credit for this examination must be obtained within six months prior to certification.

Shift Managers

3.2 A shift manager shall, at the time of certification, meet the requirements specified in paragraphs 3.2.1 to 3.2.4.

3.2.1 Have successfully completed the Commission examination for reactor operators specified in paragraph 3.1.1. Credit for this examination must be obtained before taking the examination referred to in paragraph 3.2.2 and, subject to the provisions of section 4, within three years prior to certification.

3.2.2 Have successfully completed the Commission examination for reactor operators specified in paragraph 3.1.2. Credit for this examination must be obtained before taking the examination referred to in paragraph 3.2.3 and, subject to the provisions of section 4, within two years prior to certification.

3.2.3 Have successfully completed the Commission examination for shift supervisors which samples topics covered in the training referred to in paragraph 2.3.2. Credit for this examination must be obtained before taking the examination referred to in paragraph 3.2.4 and, subject to the provisions of section 4, within two years prior to certification.

3.2.4 Have successfully completed the Commission simulator-based examination for shift supervisors covering:

- independent monitoring of facility systems and equipment under abnormal and emergency conditions
- independent diagnosis and decision making
- supervision and direction of control room operations under abnormal and emergency conditions

Subject to the provisions of section 4, credit for this examination must be obtained within six months prior to certification.

4 Extending the Time Limits of Examination Credits

Where a person cannot be certified within the time limit for a given examination credit specified in sections 2 and 3, the extension of the time limit of the examination credit that may be approved will not exceed one year. The licensee shall submit information pertaining to one or more of the following items in support of a request for an extension:

- the opinion of the licensee that the person needs to perform the duties of the position under the supervision of a certified or authorized incumbent of the position for longer than the minimum times specified in section 2 before taking the final simulator-based examination
- the person has failed one examination for which a credit must be obtained within a specified time prior to certification
- the person's training has been delayed due to sickness or injury of the person
- the person's family related responsibilities justify the extension

5 Advancement From Authorized Nuclear Operator or Control Room Shift Operating Supervisor to Shift Manager

Authorized Nuclear Operator to Shift Manager

5.1 An authorized nuclear operator at the nuclear facility to obtain a certification as shift manager at the facility shall, at the time of certification, meet the requirements specified in paragraphs 5.1.1 to 5.1.8.

5.1.1 Have a minimum of 4 years of experience as authorized nuclear operator at the nuclear facility.

5.1.2 Have successfully completed the training specified in paragraph 2.3.2.
This training shall be followed by a comprehensive written examination set by the licensee. Credit for this examination must be obtained before the person may take the Commission examination referred to in paragraph 3.2.3.

5.1.3 Have successfully completed, on the facility full scope replica simulator, the components of the training specified in paragraph 2.3.3 covering:

- operation and monitoring of facility systems and equipment for which the authorized nuclear operators do not have direct operational control under normal, abnormal and emergency conditions
- independent monitoring of facility systems and equipment under normal, abnormal and emergency conditions
- independent diagnosis and decision making
- supervision and direction of facility operations under normal, abnormal and emergency conditions
- interactions with other members of the shift crew

This training shall be followed by a comprehensive simulator-based examination set by the licensee.

5.1.4 Have successfully completed the on-the-job training specified in paragraph 2.3.4.

5.1.5 Have satisfactorily performed the duties of a shift manager under the supervision of a certified or authorized incumbent of the position for a minimum of 480 hours on shift after having complied with the requirements specified in paragraphs 5.1.1 to 5.1.4.

5.1.6 Have successfully completed a formal interview by facility management which confirms the person's competence to perform the duties of the shift manager.

5.1.7 The person shall comply with the requirements specified in paragraphs 5.1.1 to 5.1.6 before taking the Commission examination referred to in paragraph 3.2.4.

5.1.8 Have successfully completed the Commission examinations for shift supervisors referred to in paragraphs 3.2.3 and 3.2.4.

5.2 The person shall meet the requirements specified in paragraphs 5.1.2 to 5.1.6 and in paragraph 5.1.8 within 2 years of the person's selection for the shift manager training.

Control Room Shift Operating Supervisors to Shift Managers

5.3 An control room shift operating supervisors at the nuclear facility to obtain a certification as shift manager at the facility shall, at the time of certification, meet the requirements specified in paragraphs 5.3.1 to 5.3.7.

5.3.1 Have successfully completed the components of the training specified in paragraph 2.3.2 that were not covered in the control room shift operating supervisor training referred to in paragraph 2.2.1.
This training shall be followed by a comprehensive written examination set by the licensee. Credit for this examination must be obtained before the person may take the Commission examination referred to in paragraph 3.2.3.

5.3.2 Have successfully completed, on the facility full scope replica simulator, the component of the training specified in paragraph 2.3.3 covering:

- supervision and direction of facility operations under normal, abnormal and emergency conditions
- interaction with other members of the shift crew
  This training shall be followed by a comprehensive simulator-based examination set by the licensee.

5.3.3 Have successfully completed the on-the-job training specified in paragraph 2.3.4.

5.3.4 Have satisfactorily performed the duties of a shift manager under the supervision of a certified or authorized incumbent of the position for a minimum of 480 hours on shift after having met the requirements specified in paragraphs 5.3.1 to 5.3.3.

5.3.5 Have successfully completed a formal interview by facility management which confirms the person's competence to perform the duties of the shift manager.

5.3.6 The person shall meet the requirements specified in paragraphs 5.3.1 to 5.3.5 before taking the Commission examination referred to in paragraph 3.2.4.

5.3.7 Have successfully completed the Commission examinations for shift supervisors referred to in paragraphs 3.2.3 and 3.2.4.

5.4 The person shall meet the requirements specified in paragraphs 5.3.1 to 5.3.5 and in paragraph 5.3.7 within 2 years of the person's selection for the shift manager training.

APPENDIX F

Requirements for Continuing Training for Certified or Authorized Persons in Operating Positions

1 Continuing Training Requirements

1.1 During the term of their certification or authorization, authorized nuclear operators, control room shift operating supervisors and shift managers shall meet the requirements specified in paragraphs 1.1.1 to 1.1.3.

1.1.1 Participate, on a regular basis and over a cycle not exceeding three years, in continuing training appropriate to the knowledge and skill requirements of their position, covering:

- a review of the knowledge, learned in their initial training referred to in section 2 of Appendix E, that is not maintained through day-to-day operation of the facility and that is required to work competently in their position
- simulator-based exercises that cover infrequent normal manoeuvres
- simulator-based exercises that cover a sufficiently varied number of situations that challenge the diagnostic and problem solving abilities of the certified or authorized persons and ensure that they are, at all times, proficient in selecting and using abnormal and emergency operating procedures
- exercises and drills conducted at the facility on a regular basis throughout the program to practise the response to accidents and emergencies

1.1.2 Participate in training appropriate to the knowledge and skill requirements of their position covering topics identified as a result of:

- changes to facility systems and equipment
- changes to licensee's and facility's policies, standards and procedures
- changes to regulatory requirements
- changes to the facility's operating licence or to documents referenced in the licence
- facility or industry experience and operating events

1.1.3 Successfully complete written and simulator-based tests that confirm that the person possesses the knowledge and the skills covered in each training session.

ANNEX 8.1
Canadian Nuclear Safety Commission (CNSC) Staff Organization

The CNSC consists of a President, a federally appointed Commission, and staff of the Commission. This general structure is defined by the existing legislation:

- The *NSC Act* establishes a seven-member Commission, appointed by the federal government through the Governor-in-Council (Cabinet of the Government of Canada).

- The *NSC 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 Commission. Sub-section 12(1) of the *NSC Act* states that the President "has supervision over and direction of the work of the Commission, and of the officers, technical and otherwise, employed for the purpose of carrying on the work of the Commission".

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

The major divisions of the CNSC are:

- the President's Office;
- the Office of Regulatory Affairs;
- the Office of International Affairs;
- the Directorate of Corporate Services;
- the Directorate of Reactor Regulation;
- the Directorate of Fuel Cycle and Materials Regulation; and
- the Directorate of Environmental and Human Performance Assessment.

Other groups in the CNSC organizational structure include the Legal Services Unit, the Audit and Evaluation Group, the Secretariat, the Advisory Committee on Radiation Protection and the Advisory Committee on Nuclear Safety.

The CNSC staff organization chart is shown in Figure 8A.1.

PRESIDENT'S OFFICE

The President's Office provides administrative support services directly to President. The Secretariat ensures that the seven-member Commission has the administrative and technical support it needs to function efficiently and effectively. It also provides the following functions:
- manages the Commission meeting process;
- provides recording services at Commission meetings;
- drafts minutes of Commission meetings;
- drafts and coordinates responses to submissions to the Commission;

- prepares announcements of Commission deliberations and decisions;
- provides scientific and administrative support to the Advisory Committee on Radiation Protection and the Advisory Committee on Nuclear Safety; and
- drafts policies, procedures and rules on Commission operations.

OFFICE OF REGULATORY AFFAIRS

The Office of Regulatory Affairs is responsible for a number of organization-wide programs, initiatives and actions that enhance the CNSC's regulatory effectiveness, efficiency and overall operation. In addition, the Office of Regulatory Affairs includes the Research and Support Group and the Regulatory Documents Group.

The Regulatory Documents Group designs and implements the framework and work processes to produce and manage corporate regulatory documents, and coordinates the preparation and management of CNSC corporate documents.

The Research and Support Group (RSG) is responsible for management of the Research and Support Program (the Program). Each year the CNSC funds the Program to facilitate work on mission-related activities. 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.

OFFICE OF INTERNATIONAL AFFAIRS

The Office of International Affairs:

- ensures that the CNSC 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; and
- ensures there is compliance with the *Nuclear Security Regulations* and the *Nuclear Non-Proliferation Import and Export Control Regulations*.

DIRECTORATE OF CORPORATE SERVICES

The Directorate of Corporate Services manages the CNSC's human, financial, physical and information resources. It makes sure that the CNSC 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 CNSC'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, the External Relations and Documents Division, the Communications Division and the Information Management Division.

The Human Resources Division provides specialist support to the CNSC in all areas of the human resources field, including planning, policy development, staffing, compensation and staff relations. The division also represents the CNSC 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 CNSC staff:

- assesses staff training needs, and recommends programs to meet these needs;
- provides related advice and information to CNSC staff;
- administers the corporate budget for non-technical training;
- coordinates the delivery of non-technical training programs; and
- maintains lists of trainers and records of employees' training.

The Finance Division provides financial services on behalf of all employees and units. The division participates in development or revision of financial policies, produces reports for CNSC staff and management, interfaces with other central agencies, such as Treasury Board, on financial matters. The Finance Division also:

- administers the CNSC's corporate security program on behalf of the President;
- obtains security clearances for CNSC staff in accordance with federal government policies;
- makes sure that physical security systems to protect CNSC property and information systems are developed and maintained; and
- investigates potential or actual breaches of security within the CNSC.

The External Relations and Documents Division manages CNSC interactions with the Minister's Office and manages CNSC relations, agreements, cooperation and involvement with various international organizations, agencies and governments The division administers the CNSC nuclear emergency preparedness plan in cooperation with federal, provincial and municipal agencies, and CNSC compliance with the federal *Access to Information* and *Privacy Acts*.

The Communications Division:

- provides information and publishing services on behalf of the CNSC;
- responds on an as-required basis to oral and written enquiries and requests from CNSC staff, the public, and the news media;
- issues news releases, notices and information bulletins concerning regulatory developments, nuclear safety, and Commission decisions; and
- generates, publishes, stocks and distributes documents that describe nuclear technology, the nature and effects of radiation, the organization, mandate and activities of the CNSC, Commission policies and decisions, the conclusions and recommendations of CNSC Advisory Committees, nuclear legislation and CNSC regulatory requirements and expectations.

The Information Management Division:

- operates the CNSC Records Office and the CNSC Library;
- administers and maintains the CNSC's electronic systems for information management and exchange; and
- provides technical support services on related equipment and systems to CNSC staff and work units.

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, Safety Evaluation Division - Analysis, and Safety Evaluation Division - Engineering.

The Power Reactor Operations Division regulates the construction, commissioning, and operation of nuclear power station installations, on a day-to-day basis. To promote or verify compliance with applicable legislation, licences, and standards, staff:

- assess applications, reports and submissions;
- conduct or participate in surveys, inspections, audits, reviews, and investigations; and
- recommends appropriate licensing and enforcement actions.

The Power Reactor Evaluation Division evaluates the performance of nuclear power reactor installations in Canada. Staff:

- manage technical reviews of issues that the CNSC considers pertinent to the safe design, construction, commissioning, operation, and maintenance of Canadian power reactors;
- coordinate review activities;
- report findings; and
- recommend follow up activities.

Safety Evaluation Division - Analysis participates in safety evaluations of nuclear power reactors and other facilities regulated by the CNSC. The division provides technical expertise in the areas of thermal hydraulics, reactor physics, reactor protection and plant behaviour. Staff conduct and participate in assessments, reviews, inspections and audits involving their areas of expertise.

Safety Evaluation Division - Engineering also participates in safety evaluations of nuclear power reactors and other facilities regulated by the CNSC. The division provides technical expertise in the areas of reliability and risk assessment, control and instrumentation, electrical systems, civil engineering and integrity of pressure-retaining components and systems. Staff conduct and participate in assessments, reviews, inspections and audits involving their areas of expertise.

DIRECTORATE OF FUEL CYCLES 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—and radioactive waste management facilities. The Directorate provides laboratory and compliance services for various CNSC 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 *Nuclear Safety and Control 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 ensures that uranium mines and uranium processing facilities are safely designed, developed, constructed, operated, maintained and decommissioned. To promote compliance with applicable legislation, licences, and standards, staff:

- assess applications, reports and submissions;
- conducts surveys, inspections, audits, reviews, and investigations; and
- recommends appropriate licensing and enforcement actions.

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:

- assess applications, reports and submissions;
- conducts or participates in surveys, inspections, audits, reviews, and investigations; and
- 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 protection of the environment. MRD staff: assess licence applications, prepare and issue licences, inspect uses of radioactive materials, and draft regulatory standards.

The Research and Production Facilities Division supplies laboratory services for all CNSC 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. Finally, this division carries out the CNSC role of regulating the safe transport of radioactive materials in Canada, and is involved with national and international organizations on matters related to international movements of radioactive material.

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 CNSC licensees and by applicants for CNSC licences. The Directorate also conducts audits of the radiation training and safety programs that are in place at CNSC-licensed facilities, certifies key operating personnel at nuclear power reactor installations, administers the CNSC's mission-oriented research and support program, and delivers the CNSC 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, and the Technical Training Group.

As requested by the Commission or CNSC 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 CNSC licensees;
- assesses the emergency preparedness plans of CNSC licensees;
- participates in external federal and provincial environmental assessment projects; and
- 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 CNSC licensees, and the adequacy of the corresponding employment standards and employee training programs. In cooperation with other CNSC 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 CNSC-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 CNSC units, the Performance Evaluation Division develops regulatory guidance on human factors policies and programs, operational processes and methods and personnel systems.

The Technical Training Group develops, delivers and reports upon, training programs to meet the technical needs of CNSC staff and the needs of other nuclear regulatory agencies.

Subject to federal policies and applicable legislation, CNSC employees are hired by the Commission to perform assigned duties. Under the supervision of the President, they perform various tasks that are essential to the functioning of the CNSC, and to the effective discharge of the Commission's responsibilities. See Annex 8.1 for a current CNSC staff organization chart. The job-related performance of all CNSC staff is formally evaluated each year in accordance with CNSC administrative policies and procedures. CNSC staff must maintain and demonstrate required skills and satisfactory performance. Issues regarding maintenance of technical capabilities and succession planning were discussed in Article 11.

CNSC has approximately 450 employees including:

- administrators;
- financial officers;
- auditors;
- scientists;
- engineers;
- chemists;
- biologists;
- mathematicians;
- health and nuclear physicists;
- accountants;
- technicians;
- electronic data processing experts;
- safeguards experts;
- nuclear non-proliferation experts;
- security experts;
- information processing and management specialists;
- support staff;
- maintenance personnel; and
- other specialists in a wide variety of fields and disciplines essential to effective discharge of the Commission's responsibilities and daily operation.

The tasks of CNSC staff are to:

- evaluate and process applications for CNSC licences;
- develop and prepare licensing recommendations;
- administer CNSC 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;
- ensure that the commitments pursuant to relevant bilateral and multilateral international agreements, conventions and understandings are fulfilled; and

- assist the CNSC in discharging its mandate to disseminate objective information regarding nuclear energy.

The CNSC 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 1999-2000, a total of $2,146,000 was spent on research and support work in the Program. A total of 79 projects were worked on during the year. The activities included projects in the fields of nuclear reactors; fuel cycle facilities; uranium mines and mills; waste management; dosimetry; health physics; regulations; and regulatory process development. Sub-program topics were: Environmental Protection; Radiation Protection and Health Effects; Safety and Reliability of Nuclear Facilities; Structural Integrity; Physics and Fuel; Human Performance; Waste Management and Decommissioning; Regulatory Processes and Corporate Affairs; and Special Services

In addition to the support provided by CNSC staff, the President and the Commission receive input from external sources. Advisory Committees provide specialist advice on radiological protection, nuclear safety and medical matters. 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 Commission's policies on appeals and representations allow licensees and the public the right and opportunity to be heard on nuclear matters of concern.

The CNSC is funded by Parliamentary appropriations. Its total expenditure for the fiscal year ending March 31, 2000 was $58,869,183. During this period, the CNSC collected $42,312,325 or approximately 70% of its total expenditure, through fees charged for licences and permits. The funds recovered are credited directly to the Consolidated Revenue Fund (treasury) of the federal government.

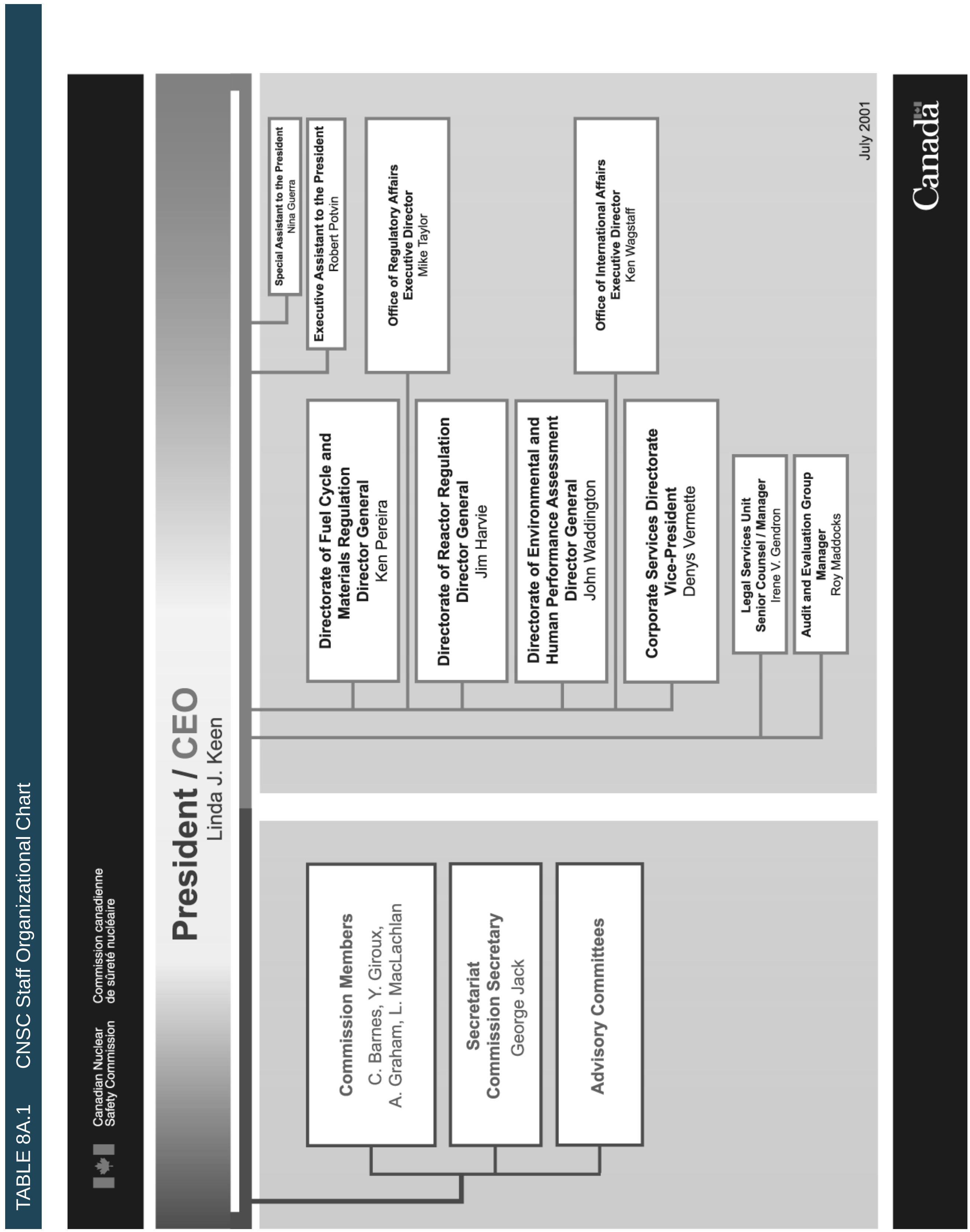


TABLE 8A.1 CNSC Staff Organizational Chart
Canadian Nuclear Safety Commission
Commission canadienne de sûreté nucléaire
President / CEO
Linda J. Keen
Commission Members
C. Barnes, Y. Giroux, A. Graham, L. MacLachlan
Secretariat
Commission Secretary
George Jack
Advisory Committees
Special Assistant to the President
Nina Guerra
Executive Assistant to the President
Robert Potvin
Office of Regulatory Affairs
Executive Director
Mike Taylor
Office of International Affairs
Executive Director
Ken Wagstaff
Directorate of Fuel Cycle and Materials Regulation
Director General
Ken Pereira
Directorate of Reactor Regulation
Director General
Jim Harvie
Directorate of Environmental and Human Performance Assessment
Director General
John Waddington
Corporate Services Directorate
Vice-President
Denys Vermette
Legal Services Unit
Senior Counsel / Manager
Irene V. Gendron
Audit and Evaluation Group
Manager
Roy Maddocks
July 2001
Canada

ANNEX 9.1
Activities of the Licence Holder Related to Safety Enhancement

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

- design configuration control;
- analysis reviews;
- reliability studies;
- risk management and control;
- peer evaluation; and
- external reviews.

DESIGN CONFIGURATION CONTROL

A configuration control process for the existing design includes:

- the design audit process (vertical/horizontal slice);
- the process of periodically ensuring that the system is designed to meet relevant standards and has not been inadvertently modified outside of its specifications (for example, by addition, removal or replacement of components, or other changes to its operating environment); and
- 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 ensure 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.

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. 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”

The design control and modification process fully complies with Canadian Standards Association (CSA) N286.2, “Design Quality Assurance for Nuclear Power Plants.”

ANALYSIS REVIEWS

Nuclear Safety Analysis reviews are undertaken periodically to account for utility operating experience, account for improved analytical techniques and 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 ensure 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; and
- 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 orgAanization 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; and
- 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; and
- correct documentation of results.

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 ensure 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; and
- 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 station would benefit from improvement; and
- 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 plant operations; and
- evaluate:
    - the knowledge and performance of station personnel;
    - the condition and performance of systems and equipment; and
    - the quality of programs and procedures; and
    - 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 following areas are evaluated:

- Organization and Administration;
- Operations;
- Maintenance;
- Engineering Support;
- Radiological Protection;
- Chemistry; and
- 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 and the evaluator’s experience.

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.

ANNEX 14.1
Key Issues for Return of the Pickering “A” Reactors to Service

Some key aspects of the Pickering “A” return to service project were:

- enhancement of the Pickering “A” shutdown system;
- a systematic review of safety;
- a review of the Pickering “A” design against applicable regulatory documents, codes and standards;
- a review of the Pickering “A” risk assessment; and
- an environmental assessment;

A summary of these issues is presented below.

SHUTDOWN SYSTEM ENHANCEMENT

When Pickering NGS-A was originally licensed for operation in 1971, the regulatory requirements of the CNSC did not include the current requirement that the design include two fully independent, diverse shutdown systems meeting regulatory document R-8, “Requirements for Shutdown Systems in CANDU Reactors”. Note that only Darlington is required to meet R-8 requirements. Pickering NGS-A was constructed with two shutdown mechanisms: shutoff rods which could be inserted quickly and a moderator dump system which was relatively slow. These two mechanisms were not independent - for example, they shared the same instrumentation to trigger their actuation. As well, moderator dump was not effective in shutting down the reactor under all accident conditions.

All other nuclear power stations in Canada meet the requirements of the regulatory policy. They have two distinct and separate shutdown systems, shutoff rods and a liquid injection system that injects neutron absorbing liquid into the moderator.

OPG proposed to enhance the performance of the existing shutdown system by adding diverse and independent neutron overpower and high log rate trip parameters. The CNSC accepted OPG’s proposal with three additional requirements. These requirements were: two additional process trip parameters, the addition of two more shutoff rods, and ensuring that the shutdown system enhancement met the $1x10^{-3}$ unavailability target. While this solution would not bring the shutdown capability into full compliance with regulatory document R-8, it would greatly improve the reliability of the shutdown function. The shutdown system enhancement had been installed, but not commissioned, in unit 4 when the Pickering NGS-A units were shutdown in 1997.

The installation of shutdown system enhancement makes the station sufficiently safe for any postulated accident condition. Furthermore, the “Pickering A Risk Assessment” indicated that core damage as a result of loss of regulation accidents will not occur at frequencies higher than 10^{-6} events per year. OPG has committed to install and commission the shutdown system enhancement in all units which is required by a condition of the current licence.

SYSTEMATIC REVIEW OF SAFETY

CNSC staff requested OPG perform a systematic review of safety as part of the return to service project to provide assurance that the Pickering NGS-A station could operate safely. CNSC staff made this request because the Pickering NGS-A had originally been designed and built to the standards of the late 1960s and there has been evolution of safety standards since that time. While the CNSC had continuously assessed the safety of Pickering NGS-A and the impact of newer standards on operating stations, there had not been a comprehensive review of the station design since the initial licensing of the station. CNSC and OPG agreed that a process similar to that provided in the International Atomic Energy Agency safety guide on periodic safety reviews [IAEA Safety Series No. 50-SG-012, “Periodic Safety Review of Operational Nuclear Power Plants”] should be followed.

OPG compared their assessments with the IAEA guide on periodic safety reviews and demonstrated that key features of a periodic safety review have been conducted as part of the return to service project or the Integrated Improvement Projects. As is the case with most safety reviews of this type, the comprehensive review made use of previous assessments such as the Pickering NGS-A seismic margin assessment finished in 1998.

The objectives of the Pickering “A” Return to Service Project and that of a Periodic Safety Review are the same, ensuring a high level of safety throughout the plant service life. OPG and the Pickering “A” Return to Service Project have programs and assessments to address the specific areas targeted by a Periodic Safety Review. The eleven specific safety factors included in a Periodic Safety Review have been or are being addressed by the Pickering “A” Return to Service project. Table A14.1 provides a summary of the reviews done and the resulting improvements undertaken, categorized by the factors listed in IAEA Safety Series document 50-SG-012. From the above comparison between the Pickering “A” Return to Service Project and that of a Periodic Safety Review, it can be concluded that all major performance areas have been addressed in completed and planned reviews of Pickering “A”.

TABLE A14.1 Comparison of Ontario Power Generation (OPG) Assessments with the International Atomic Energy Agency (IAEA) Periodic Safety Review Attributes

Safety Factor	Review and Assessments	Improvements Initiatives
Physical Condition	IIPA SSFI System Condition Assessments Equipment Tests and Inspections Seismic Assessment Fire Safety Assessment Review Against Current Standards CSA M293-95 Review System Health Reports`	IN-OE-004 Configuration Management N-EN-001 Engineering Leadership and Management N-0P-003 Plant Status Control N-MA-006 Preventive Maintenance Optimization N-EN-004 Engineering Programs N-EN-007 Steam Generator/Pressure Tube Inspections Major Equipment Maintenance and Replacement Reduce Maintenance Backlogs
Safety Analysis	PARA Reduction of Core Damage Frequency Confirmation Review of Safety Report Analysis Sections Large LOCA Analysis Moderator System and End-shield System Analyses Seismic Assessment	Modifications to Reduce Probability of Severe Core Damage Licensing Basis Project
Equipment Qualification	EQ Project	N-EN-009 Environmental Qualification
Management of Aging	IIPA System Condition Assessments EQ Project System Health Reports Inspections	N-EN-004 Configuration Management N-EN-007 Steam Generator/Pressure Tube Inspections N-EN-009 Environmental Qualification N-EN-006 Preventive Maintenance Optimization
Safety Performance	IIPA SSFI Environmental Review Environmental Assessment	IIP N-OE-001 Performance Monitoring N-EN-002 Operability Determinations Emission Reduction Initiatives
OPEX and Research	IIPA	N-PA-005 OPEX Project Mangement
Procedures	IIPA Review of RP Procedures Conduct of Operations Conduct of Maintenance	N-OP-002 Conduct of Operations N-MA-002 Conduct of Maintenance N-EN-006-3 Safe Operating Envelope N-EN-001 Engineering Governing Documents Standardize RP Procedures EOP for Uninhabitable Main Control Room Upgrade AIM
Organization/ Administration	IIPA	25 IIP Projects

TABLE A14.1 Continued

Safety Factor	Review and Assessments	Improvements Initiatives
Human Factors	IIPA Self Assessment Program Human Factors Review of Pickering “A” Control Room	N-OP-002 Conduct of Operations N-MA-001 Maintenance Management N-MA-002 Conduct of Maintenance N-EN-004 System Engineering & Engineering Programs N-EN-005 Engineering Resource Management N-RP-001 Radiation Protection Resource Management N-RA-005 Regulatory Affairs Staff Augmentation N-OE-010 Nuclear Mentoring Program N-TR-001 Authorization Training N-TR-002 Non-licenced Training Recovery Project N-TR-003 Training Facilites & Support Services N-TR-004 Training Procedures and Processes N-TR-005 Building Management Leadership SNPM FLM Academy SSPDS ESPDS MSPDS MARC
Emergency Planning	IIPA SSFI for Fire Protection	N-EP-001 Emergency Preparedness Program N-EN-008 Fire Protection Upgrade Program
Environmental Impact	IIPA OPG Review of Environmental Events Pickering Environmental Review (1998) Environmental Assessment	N-EV-001 Environmental Management System N-EV-005 Radiation Emmission Reduction N-EV-006 Upgrade Environmental Impact Monitoring N-EV-007 Conventional Fuel Oil Storage Tank Maintenance N-EV-008 MISA Compliance Strategy N-EV-009 Contaminated Land Numerous Environmental Improvements in the Scope of the Pickering Return to Service Projects

Review of Pickering “A” against Applicable Regulatory Documents, Codes and Standards

Pickering “A” was designed in the late 1960s and early 1970s in conformance with codes, standards and regulatory requirements of that period of time. More recent plants have been designed to more recent codes, standards and regulatory requirements. The fundamental difference between the application of old design standards and current standards led OPG to review the Pickering “A” design against the relevant, current standards. OPG identified modifications that could be performed to ensure conformance to the extent practical or developed justification to support the remaining differences.

A list of CNSC Regulatory Documents and Consultative Documents was reviewed. Pickering “A” was reviewed against those CNSC documents classified as having “a direct, immediate effect on installed design features”, were reviewed, except for CNSC Regulatory Document R-10 [The Use of Two Shutdown Systems in Reactors], (installation of the enhanced shutdown system was the result of a lengthy process to determine how Pickering “A” should comply with the intent of R-10. The requirement to install SDSE prior to the return to service is a Condition of the Pickering “A” Power Reactor Operating Licence.).

In additon, a total of 41 codes and standards were identified by OPG. Accordingly, Pickering “A” was reviewed against those CSA Standards classified as having “a direct, immediate effect on installed design features” were marked for review, except for the following three departures from this rule:

- Pickering “A” was not be reviewed against CSA CAN3-N289.1-80, because completion of the Seismic Margin Assessment made such a review unnecessary.
- Pickering “A” was not be reviewed against CAN/CSA-N289.5-M91, because installation of seismic instrumentation at the Pickering site made such a review was unnecessary.
- Pickering “A” was reviewed against CSA-N293-95, Fire Protection for CANDU Nuclear Power Plants, as part of a separate effort.

Regulatory documents, codes and standards that the Pickering “A” design was reviewed against are listed below (Table A14.2)

There are certain areas where the review acknowledged that Pickering “A” does not meet modern standards. These include shutdown systems, fire protection, main control room design, and seismic design. OPG had already committed to the required upgrades in these areas and, therefore, they were not addressed by the systematic review.

The review showed that, in most cases, the Pickering “A” design meets the requirements or the intent of the requirement of the standards and regulatory documents. There were a small number of recommendations that were identified by the review and are being addressed by OPG.

TABLE A14.2 Regulatory Documents, and Codes and Standards used in the Pickering “A” Design Review

CNSC Regulatory Documents	CSA Standards
R- 7 (1991), Requirements for Containment Systems for CANDU Nuclear Power Plants R- 8 (1991), Requirements for Shutdown Systems for CANDU Nuclear Power Plants R- 9 (1991), Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants R- 77 (1987), Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems	CSA- N285.0 (1995), General Requirements for Pressure- Retaining Systems and Components in CANDU Nuclear Power Plants CAN/ CSA- N285.2, Requirements for Class 1C, 2C, and 3C Pressure- Retaining Components and Supports in CANDU Nuclear Power Plants CAN/ CSA- N285.3- 88, Requirements for Containment Systems Components in CANDU Nuclear Power Plants CAN/ CSA- N285.6 Series- 88 (series of 9 standards), Material Standards for Reactor Components for CANDU Nuclear Power Plants CSA- N287.1- 93, General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants CAN/ CSA- N287.2- M91, Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants CSA- N287.3- 93, Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants CAN/ CSA- N287.4- 92, Construction, Fabrication, and Installation Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants CAN3- N288.3.2- M85, High Efficiency Air- Cleaning Assemblies for Normal Operation of Nuclear Facilities CAN3- N290.1- 80, Requirements for the Shutdown Systems of CANDU Nuclear Power Plants CAN3- N290.4- M82, Requirements for the Reactor Regulating Systems of CANDU Nuclear Power Plants CAN/ CSA- N290.5- M90, Requirements for the Support Power Systems of CANDU Nuclear Power Plants CAN3- N290.6- M82, Requirements for Monitoring and Display of CANDU Nuclear Power Plant Status in the Event of an Accident

REVIEW OF THE PICKERING “A” RISK ASSESSMENT

OPG completed and submitted the Pickering “A” Risk Assessment (PARA) to the CNSC in 1995. A subsequent review of PARA was completed to identify the improvements to be made prior to returning the Pickering “A” units to service. The PARA was conducted with the following key objectives:

- to review the adequacy of the safety of the station design and operation by means of preparing a risk model for the station; and
- to prepare a risk model in a form that could be used to assist some aspects of the safety-related decision-making process at the station.

The Pickering “A” risk assessment was based upon 34 detailed fault tree models of plant systems having safety significance. The integration of these models for an extensive range of initiating events, in conjunction with the assessment of consequences, allows the quantifiable risks associated with the operation of Pickering “A” to be calculated.

The scope of the PARA included the assessment of the public health and economic risk arising from the potential for radiological releases following initiating events internal to the station and caused by the failure of plant systems (including loss of off-site power), which can lead to damage to fuel within the reactor. The consequences of concern in the study were the radiation dose to the public beyond the site boundary, and economic losses arising from both damages to the plant and the release of radioactivity beyond the site boundary. Risks estimated are public health, on-site and off-site economic impact. The calculated public health risks were compared against the currently used OPG risk-based safety goals as a test of safety design adequacy.

The principal results of the study were:

- The risk to the health and welfare of the population living or working in the vicinity of the Pickering “A” station was low in comparison to other risks to which they are normally exposed.
- The likelihood of an accident that could cause severe damage to the reactor core was similar to that for other contemporary reactor designs, but higher than that for more recent CANDU reactors such as Darlington. This was largely due to a lack of independence between the emergency coolant injection system function and the moderator system as a heat sink function.
- The likelihood of the occurrence of a catastrophic accident that could cause acute radiation effects beyond the site boundary was sufficiently small to be considered negligible for all practical purposes. Features unique to the CANDU pressure-tube design and the multi-unit, negative-pressure containment contribute to a very low overall risk to public health and welfare.
- External economic risks from the accidental release of airborne radioactivity off-site were correspondingly low. The internal economic risk from an accident involving fuel damage was comparable to that from other stations, with the dominant contribution arising from the relatively more likely, low consequence events.
- When the report was completed, no major design or procedural changes were indicated as required as a result of this study.

The nuclear safety policy for OPG has been revised since the Pickering “A” risk assessment was issued. The revised policy specifies the following OPG risk goals and limits are listed inTable A14.3.

TABLE A14.3 OPG Nuclear Safety Goals and Limits

	Average Risk	
	Goal	**Limit**
Individual Early Fatality (Per Site)	1×10^{-6}	1×10^{-5}
Individual Delayed Fatality (Per Site)	1×10^{-5}	1×10^{-4}
Large Release (Per Unit)	1×10^{-6}	1×10^{-5}
Severe Release (Per Unit)	1×10^{-7}	1×10^{-6}
Severe Core Damage (Per Unit)	1×10^{-5}	1×10^{-4}

In order to comply with the new limits, a review of the risk assessment was performed to identify what improvements can be made prior to returning the Pickering “A” units to service.

The PARA methodology could be improved, its results are useable for correcting Pickering “A” weaknesses. In particular, correction of the cross links (notably between the Moderator and ECI recovery) and the single failure vulnerabilities (i.e., an initiating event followed by either a single mechanical failure or a failure to perform a single operator action).

The following actions were taken in response to the OPG PARA goal and limits and the CNSC concerns:

- In order to prioritize the areas where the frequency of severe core damage can be reduced, events were ranked in order of greatest potential risk reduction and greatest potential risk increase. A risk reduction factor was used which represents the decrease in the severe core damage frequency should the event be assumed not to occur (i.e., the probability for occurrence is zero). Those events that represent a greater than one percent risk reduction factor were being identified for additional review.

- Similarly, a risk increase factor represents the increase in severe core damage frequency should an activity be assumed always to fail. The top fifty activities ranked in terms of risk increase were reviewed to identify areas for additional review.

- A list of single component/operator action failures that contribute to the frequency of severe core damage (fuel damage categories FDC1 and FDC2) was prepared and reviewed to identify any practical design modifications.

- An option assessment identified ways of improving the long-term ECI reliability. The option assessment formed the basis for improving the long-term ECI function and reducing the frequency of severe core damage accidents and minimizing the cross link between moderator cooling and ECI recovery.

The risk reduction worth of each proposed change was determined. In addition, the overall reduction of severe core damage frequency was calculated. The PARA review report indicates that, by implementing the selected design modifications, the severe core damage frequency will be reduced from 1.3×10^{-4} to below 3×10^{-5} occurrences per year. In addition, the identified design modifications will eliminate the majority of the identified single failure vulnerabilities and mitigate the identified critical operator actions. Prior to the return to service of the Pickering A station, the severe core damage frequency will comply with the nuclear safety policy of OPG.

ENVIRONMENTAL ASSESSMENT

Information from the following programs was used in the systematic review of the environmental performance of Pickering:

- radioactive effluent monitoring;
- radiological environmental monitoring;
- program for estimating doses to members of the public;
- conventional effluent monitoring;
- conventional environmental monitoring, biodiversity and land use programs;
- pollution prevention programs;
- radioactive waste management; and
- conventional waste management.

These categories were further examined with the focus on using statistical methods to identify trends and unusual years.

The report summarized a number of issues associated with the potential effects of Pickering operations on the biophysical environment and provided a preliminary analysis of the key issues. While the environmental effects of most current Pickering activities were addressed by existing or planned programs, several informational gaps were identified that precluded conclusions regarding the effects for some issues. The resulting plan addressed issues of concern to the community as well as those with scientific or technical significance. The Environmental Action Plan identified the following ten areas in which improvements could be made on key environmental issues to ensure that the valued ecosystem components near the station were protected or enhanced:

- improve environmental monitoring / data systems;
- actions to control emission releases;
- improve waste management operations;
- measures to enhance area valued ecosystem components;
- introduce new risk management tools;
- improve public communication / participation;
- raise level of environmental awareness among Pickering staff;
- develop five-year plan to address biophysical issues;
- develop plan to address outstanding issues; and
- develop environmental management system.

The Environmental Review project recommended actions were integrated into an overall station Environmental Management System. Pickering developed its Environmental Management System during 1998 and the issues identified by the Environmental Review project were incorporated.

The CNSC determined that an Environmental Assessment of Pickering “A” was required under the *Canadian Environmental Assessment Act (CEAA)* as a prerequisite for consideration by the CNSC of OPG’s application for approval to return Pickering “A” to service upon completion of a specified work program. The key elements of the Commission’s decision were as follows:

- The project, taking into account the mitigation measures proposed in the Environmental Assessment Report, is not likely to cause significant adverse environmental effects.

- Therefore, consistent with paragraph 20(1)(a) of the *Canadian Environmental Assessment Act*, the CNSC may now proceed with consideration of OPG’s Operating Licence application under the *Nuclear Safety and Control Act*. The Operating Licence application will be considered under the CNSC’s normal public hearing process.

As part of its decision on the environmental assessment, the CNSC identified a follow-up and monitoring program that would be required if the CNSC approves the return to service. The purpose of the follow-up and monitoring program was two-fold:

- to assist in determining if the environmental effects of the project are as predicted in the environmental assessment; and

- to confirm if the proposed impact mitigation measures are effective.

Requirements specified by the CNSC as part of the decision were:

- The specific mitigation measures proposed in the Environmental Assessment must be implemented.

- The Commission noted that, for the purpose of this environmental assessment, the upgrades and improvements necessary for the return to service were assumed to have been completed. The Commission noted that, if the licensing process proceeds, it would need to include a mechanism to require these upgrades and improvements to be put in place; otherwise, the conclusions of the environmental assessment would not be valid for the purposes of a licensing decision.

- The environmental and safety upgrades and improvements, which were assumed to be in place for the purposes of the Environmental Assessment, are listed in Annex 14.2.

- The follow-up and monitoring program outlined in the Environmental Assessment Report needs to be further detailed through a consultation process (as planned in the Environmental Assessment) and, subject to Commission approval of the proposed return to service of Pickering “A”, integrated into the CNSC licensing and compliance process. The purpose of the follow-up program is to evaluate the accuracy of impact predictions and determine the effectiveness of the mitigation measures implemented.

- The CNSC recognizes the importance of sustained, effective and meaningful public consultation between major facility operators and the public and also OPG’s commitment to continue to improve its public involvement program. CNSC will continue to follow-up with OPG on this issue as part of the CNSC licensing and compliance process.

ANNEX 14.2
Required Improvements and Modifications for Restarting the Pickering “A” Reactors

SAFETY RELATED IMPROVEMENTS AND MODIFICATIONS

Safety-related improvements and modifications required for restart of the Pickering “A” reactors are listed below. The CNSC has indicated that this work must be completed or the information provided for each unit prior to restart of that unit and reported on in the completion assurance report, except as indicated below. “Completed” includes completion of any commissioning activities possible or practicable before restart. The remaining commissioning activities must be completed prior to return to service.

- Identify and install any changes to reactor operation required to conform with the resubmitted large loss-of-coolant analysis;
- Confirm the integrity of the calandria vault seals and penetrations for all design basis faults;
- Complete installation of shutdown system enhancements as described in licence condition 10.2 of PROL 4.00/2003;
- Complete installation of seismic upgrades;
- Complete installation of post LOCA improvements;
- Complete installation of the Emergency Coolant Injection screens and strainers;
- Complete replacement of Emergency Coolant Injection shutdown cooling isolating valve actuators;
- Complete work on position assured components;
- Complete improvements to Class III service water;
- Complete modifications to calandria inlet valves;
- Complete installation of Chameleon programmable controllers;
- Complete improvements to auxiliary feedwater system;
- Complete inspection and repair of the boiler isolating Velan valves in the heat transport system;
- Submit report on potential flooding of site (for unit 4 only);
- Complete upgrade of feedwater piping;
- Complete modifications to power house venting;
- Submit report on status of maintenance program including the status of Integrated Improvement Program projects;
- Complete inspection of feeders for flow-assisted corrosion, outlet elbow cracking and submit results;
- Complete measurement of pressure tubes axial elongation as described in OPG Fuel Channel Aging and Life Cycle Management Strategy and Plan, N-PLAN-01060-10002;
- Complete “CIGAR” inspections on 20 fuel channels and submit results;
- Complete inspection of primary heat transport pressure retaining components and metallic containment components;
- Complete reactor building pressure tests;
- Submit report on reassessment of calandria vault thermal stresses;
- Submit evidence that all block valves meet the overpressure standard requirements;
- Submit pressure vessel registrations for all out of date registrations;
- Complete modifications to reduce the core damage frequency;
- Complete the improvements to reduce the calculated unavailabilities of containment and emergency core cooling;

- Implement the proposed mechanisms and actions to mitigate control room uninhabitability or unavailability and provide supporting documentation;
- Complete installation of the committed fire suppression systems (the portion of these improvements related to the turbine hall only need to be completed prior to the turbines being returned to service);
- Complete installation of the fire alarm and annunciation system, including the committed incipient detection systems (the portion of these improvements related to the turbine hall only need to be completed prior to the turbines being returned to service);
- Submit evidence that adequate arrangements are in place with the City of Pickering Fire Department for mutual aid and coordinated fire response arrangements and hold a joint drill;
- Complete site implementation of the approved OPG fire protection program;
- Complete implementation of the upgrades identified in the Fire Protection Code Compliance Review and the Fire Hazard Assessment;
- Submit analysis of the fire survivability of the turbine building;
- Complete upgrades to minimize oxygen in the condensate;
- Complete implementation of the human factors engineering program plan;
- Complete implementation of the human factors verification and validation plan;
- Complete the configuration management Master Equipment List, the operational flowsheets, flow diagrams, on-line wiring and electrical wiring diagrams for all safety-related systems;
- Complete a detailed walkdown of the safety-related systems;
- Complete correction and updating of the operations and maintenance procedures to be technically correct and consistent with the physical plant;
- Submit evidence that a sufficient number of workers qualified to operate the unit safely under all operating conditions is available;
- Submit evidence that all workers have successfully completed refresher training and upgrade training appropriate to the knowledge and skill requirements of their position to regain their full competence. This evidence shall include such test results necessary to confirm that those persons have the knowledge and skills required to fulfill the responsibilities of their position under all unit operating conditions; and
- Complete the pre-restart portion of the Environmental Assessment Follow-up and Monitoring Program.

* The portion of these improvements related to the turbine hall only need to be completed prior to the turbines being returned to service.

REQUIRED IMPROVEMENTS AND MODIFICATIONS FROM THE ENVIRONMENTAL ASSESSMENT

For the purposes of the Environmental Assessment, it was assumed that a number of improvements and modifications were in place. For the Environment Assessment to remain valid, these improvements and modifications must be in place prior to a unit being returned to service. The remaining requirements are listed below. OPG has committed to have most of these in place prior to restart except those indicated below with an asterisk (*). These items will be completed prior to the turbines being returned to service.

- Rehabilitation of fire protection equipment;
- Upgrade of vapour recovery;
- Periodic inspection program of expansion joints and piping supports;
- Spills containment and prevention;
- Replacement or repair active liquid waste piping;
- Outfall effluent sampling;
- Condenser tube replacement;
- Improved automatic intermittent injection and dechlorination for control of zebra mussels;
- Moderator system improvements;

- Replacement of stack monitors;
- Replacement of PCB filled components;
- Replacement of irradiated fuel bay heat exchangers;
- Repair of irradiated fuel bay liner;
- Recondition shutdown cooling system;
- Replacement of turbine seal oil and lube oil heat exchangers*;
- Other small oil/water interface heat exchangers will be replaced;
- Groundwater and inactive drainage system sump sampling to identify sources of tritium in groundwater;
- Reduction of carbon-14 emissions from ion exchange columns;
- Review database of preventive and corrective maintenance activities to achieve necessary quality standards;
- Upgrading equipment to minimize oxygen in condensate and feedwater;
- Steam generator remediation;
- Rehabilitation of essential reactor building air conditioning units;
- Inspection of the Condenser Cooling Water system forebay and intake structures;
- Reconditioning heat transport system;
- Overhaul fueling machines;
- Turbine generator major maintenance*;
- Feedwater heating system upgrades;
- Auxiliary boiler feed system availability improvement;
- Electrical maintenance;
- Transformer maintenance;
- Replacement of digital control computers;
- Condenser Cooling Water system pump overhaul; and
- Control room ventilation and air conditioning rehabilitation.

ANNEX 15.1
Dose to Personnel in Canadian Nuclear Power Plants

The review of occupational dose data is an important part in the assessment of the effectiveness of licensees' radiation protection programs. The dose data were provided by the National Dose Registry which is maintained by the Radiation Protection Bureau of Health Canada. Licensees of the Canadian Nuclear Safety Commission (CNSC) are required to submit individual worker's radiation dose to the National Dose Registry on a regular basis.

The doses received by workers in Canadian nuclear generating plants are maintained well below the levels where deterministic or stochastic effects have been shown to occur. In general, studies have not identified a higher incidence of radiation-associated diseases in Canadian nuclear workers. Consequently, it is scientifically challenging to provide measures of the benefit/detriment ratio for workers in Canadian plants. One approach that has traditionally been used is collective dose, where it is assumed that the net radiological detriment to society is considered to be the same whether one worker received, for example, 20 mSv or ten workers each received 2 mSv . This approach is highly controversial in the health-physics community because it is viewed as overly conservative and potentially misleading. While the CNSC regulations do not limit collective dose, it does believe that an effective radiological protection program should endeavour to reduce the collective dose. A summary of the effective dose trends for Canadian nuclear generating stations can be found in Table A15.1.

TABLE A15.1 EFFECTIVE DOSE TRENDS (person-Sv)

	1996	1997	1998
Bruce A	2.66	3.04	0.95
Bruce B	1.27	1.46	2.74
Darlington	1.11	0.96	0.93
Gentilly-2	1.33	1.98	1.72
Pickering A & B	4.58	3.45	2.63
Point Lepreau	0.92	1.32	0.81

ANNEX 15.2
Radiological Emmissions from Canadian Nuclear Power Plants

All nuclear generating stations release small quantities of radioactive materials in a controlled manner into both the atmosphere (as gaseous effluents) and adjoining water bodies (as liquid effluents). The purpose of this document is to report the magnitude of these releases for each operating nuclear generating station in Canada. The report also indicates how these releases compare with the limitations imposed by the Canadian Nuclear Safety Commission (CNSC) as part of its regulatory and licensing program. The data show that the levels of gaseous and liquid effluents from all currently operating nuclear generating stations are well below the values authorized by the CNSC. In fact, since 1987 no releases have exceeded 1% of those values.

Radioactive material released into the environment through gaseous and liquid effluents from nuclear generating stations can result in radiation doses to members of the public through direct irradiation, inhalation of contaminated air, or ingestion of contaminated food or water. The doses received by members of the public from routine releases from nuclear generating stations are too low to measure directly. Therefore, to ensure that the public dose limit is not exceeded, the *Radiation Protection Regulations* limit the amount of radioactive materials that may be released in effluents from nuclear generating stations. These effluent limits are derived from the public dose limit and are referred to as “derived release limits” or DRLs. In addition, the industry sets operating targets that are typically a small percentage of the derived release limits. These targets are based on the ALARA principle that doses be kept “as low as reasonably achievable.” These targets are unique to each facility depending on the factors that exist at each one.

When it approved the DRLs for each nuclear generating station, the CNSC considered the environmental pathways through which radioactive material could reach the most exposed members of the public after being released from the facility. The most exposed members of the public are called the “critical group.” They are defined as those individuals who are expected to receive the highest dose of radiation because of such considerations as their age, diet, lifestyle and location.

Since 1987, DRL calculations have been based on a method recommended by the Canadian Standards Association in document CAN/CSA-N288.1-M87. This approach takes into account many more environmental pathways than did previous methods of calculating DRLs, and it allows for the use of more site-specific data. More realistic assumptions were incorporated into the method, for example, the use of shielding factors and occupancy times. Environmental transfer parameters for individual radionuclides were also updated. In addition to the use of this standard, the CNSC may place additional requirements on the calculation of DRLs such as the use of certain site-specific information to enable better estimates of environmental transfer processes.

Both the derived release limit and the actual release data for each nuclear generating station can be found below. Table A15.2 shows the gaseous releases while Table A15.3 shows the liquid releases. Note that, in all cases, the releases are much less than the corresponding DRLs.

TABLE A15.2 Gaseous Effluent Release from Canadian Nuclear Generating Stations (1999)

		Bruce A	Bruce B	Darlington	Gentilly	Pickering A	Pickering B	Point Lepreau
Tritium Oxide (TBq)	Release	3.1×10^{2}	3.1×10^{2}	218 24*	131	2.0×10^{2}	2.7×10^{2}	1.1×10^{2}
	DRL	3.8×10^{5}	4.7×10^{5}	2.1×10^{5} 7.3×10^{6}*	4.4×10^{5}	3.4×10^{5}	3.4×10^{5}	4.3×10^{5}
Carbon-14 (TBq)	Release	0.2	***	3.5	0.25	0.32	***	0.28
	DRL	2.8×10^{3}	3.0×10^{3}	1.4×10^{3}	9.1×10^{2}	8.8×10^{3}	8.8×10^{3}	3.3×10^{3}
Noble Gases (TBq-MeV)	Release	12	79	344	3.8	2.6×10^{2}	2.1×10^{2}	3.8
	DRL	2.5×10^{5}	6.1×10^{5}	2.1×10^{5}	1.7×10^{5}	8.3×10^{4}	8.3×10^{4}	7.3×10^{4}
Iodine-131 (TBq)	Release	7.2×10^{-6}	3.5×10^{-5}	3.2×10^{-5}	ND**	7.2×10^{-5}	9.6×10^{-5}	ND**
	DRL	1.2	1.3	0.6	1.3	2.4	2.4	9.9
Particulates (TBq)	Release	3.4×10^{-6}	1.1×10^{-4}	8.2×10^{-5}	7.4×10^{-6}	3.6×10^{-4}	5.7×10^{-5}	3.5×10^{-6}
	DRL	2.7	4.8	4.4	1.9	5.0	5.0	5.2

* Elemental Tritium
** ND: not detected
*** Carbon-14 releases not reported till after 1999

TABLE A15.3 Liquid Effluent Release from Canadian Nuclear Generating Stations (1999)

		Bruce A	Bruce B	Darlington	Gentilly	Pickering A	Pickering B	Point Lepreau
Tritium (TBq)	Release	240×10^{1}	220×10^{2}	89	361	320×10^{2}	130	53×10^{1}
	DRL	1.7×10^{6}	3.0×10^{6}	5.3×10^{6}	1.2×10^{6}	8.3×10^{5}	8.3×10^{5}	1.6×10^{7}
Gross Beta-Gamma (TBq)	Release	9.7×10^{-3}	1.4×10^{-3}	1.4×10^{-2}	1.6×10^{-3}	4.8×10^{-3}	1.2×10^{-2}	3.3×10^{-3}
	DRL	20.0	23.0	130.0	5.3	9.7	9.7	16.0
Carbon-14 (TBq)	Release	0.86	3.6×10^{-2}	5.7×10^{-4}	1.5×10^{-2}	NA*	1.1×10^{-2}	2.6×10^{-3}
	DRL	4.5×10^{2}	4.8×10^{2}	3.2×10^{3}	1.0×10^{2}	1.4×10^{2}	1.4×10^{2}	3.0×10^{2}

*NA: not available

ANNEX 17.1
Siting

The initial stage of the licensing process in Canada is the site acceptance as described in Article 7.3 and Annex 7.3. The siting related regulatory requirements are summarized as follows:

- “Letter of intent” submitted by the applicant to Canadian Nuclear Safety Commission (CNSC) describing the following:
  - the type, size and major characteristics of the proposed nuclear power station;
  - the site and its location; and
  - the basic organization of personnel, including identifying contact persons with whom the CNSC staff will communicate.

- a 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*);

- a “Site Evaluation Report” submitted to CNSC for site acceptance. The report is to demonstrate that the site characteristics for the nuclear power station are suitable for design, construction, commissioning and operation of the facility. 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; and

- 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.

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, ease of access/egress from the site and populated areas. 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, and 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 and 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; and
- military test ranges with the possibility of stray missiles.

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 and 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.

THE IMPLEMENTING PROVISIONS FOR THE ABOVE-MENTIONED CRITERIA

The above-mentioned criteria are implemented through the siting regulatory requirements as summarized above. 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 CNSC 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.

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 above 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; and/or
- changes to man made neighbouring facilities such as a newly constructed oil refinery, rail corridor, airport flight path or chemical plant.

The above-mentioned 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:

- demographics;
- weather experience;
- seismicity;
- neighbouring facilities; and
- air and rail transport corridor activity, etc.

For radiation safety, environmental radiological monitoring programs have been instituted by each licensee to make sure there is continued safety acceptability at Canada's nuclear power stations. The four primary objectives of these environmental monitoring 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; and
- 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 CNSC 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 these stations. 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 and eat only local fish and produce.

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"; and
- to include in the notification "information on the proposed activity, including any available information on its possible transboundary impact".

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(1).

The CNSC 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.

[1] see the IJC (International Joint Commission) publication of February 1994.

ANNEX 18.1
Design and Construction

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 the process systems; and
- a “fail-safe” mode of operation should a component or a system failure occur.

One of the ways the CNSC ensures 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 specify the safety design standards for the special safety systems that include two shutdown systems, containment and the emergency core cooling system.

Using the plant design, licensees perform safety analyses to demonstrate that the reference dose limits for particular events are not exceeded. These reference dose limits are specified in CNSC documentation. The station design is also to be reviewed in a systematic and auditable manner to identify any other events of potential concern.

THE CONCEPT OF DEFENCE-IN-DEPTH

The application of the concept of defence-in-depth in the CANDU reactor safety philosophy was discussed in Article 6. 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 (e.g., 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; and
- provision of equipment and procedures to back up accident prevention measures in order to control the course and limit the consequences of accidents.

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 indicated:

- 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.

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

- a high level of quality during all aspects of a project;
- strict quality control during manufacturing and installation;
- use of proven components;
- well-trained staff;
- periodic inspection and testing of components and systems;
- safe and efficient operation within the operating envelope; and
- a high level of automation to reduce the risk of operator errors.

REDUNDANCY

Redundancy is the use of two or more components or systems that are each 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, each of which can maintain the station in a safe state supposing a failure of one of the groups occurs. This 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 station systems are divided into two basic groups as follows:

- **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; and
  - a set of safety support systems.

- **Group 2 Systems:** Systems that provide a safety function to mitigate an event and 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; and
  - 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.
Safety-related systems are separated in terms of the “two-group separation philosophy”. This concept divides selected safety-related systems into two groups, each capable of performing the essential safety functions of nuclear power station shutdown. If one of the groups becomes unavailable because of a localized event occurring outside the nuclear power station building, the other group will provide the safety function.

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.

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, and 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 above; and
- incorporating measures to protect these barriers from damage due to accidents.

Mitigation of accidents also includes building redundancy and diversity 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.

MEASURES FOR MAKING SURE THERE IS APPLICATION OF TECHNOLOGIES PROVEN BY EXPERIENCE OR QUALIFIED BY TESTING OR ANALYSIS

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 and Annex 7.3 for the licensing of new nuclear power stations, and in Article 7.4 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; and
- research and development findings.

As a result, many of the events in the Safety Report are often re-analysed in the updated version. The document “Requirements for the Safety Analysis of CANDU Nuclear Power Plants”, specifies the requirement for quality and validation for both analysis and computer codes to ensure there is adherence to current standards.

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. The overlapping functions of the control and safety systems make it easier to operate the nuclear power station within its operating envelope.

The CNSC 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 identifies responsibilities, the operating envelope and the principles to be applied for safe, easy, and well-controlled operation.

The OP&P is reviewed and approved by the CNSC 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 was given to device grouping, layout, labelling and 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 and informal briefings.

  Training is designed to make sure that employees perform the tasks required for their positions competently and 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.